



**Measurement of the  
Thermal Neutron Flux of  $^{241}\text{Am-Be}$  source  
and  
Thermal Neutron Capture Cross-Section for  $^{41}\text{K}$**

**A thesis presented to the school of Graduate Studies  
Addis Ababa University**

***In Partial Fulfillment of the Requirement for the Degree  
of  
Master of Science in Physics***

By

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## ABSTRACT

The thermal neutron flux of a  $^{241}\text{Am-Be}$  ( $\alpha, n$ ) source available in AAU (Science Faculty) is determined using neutron-activation method of KI target which is 99.99% pure. The flux is measured relative to  $^{127}\text{I}$ , whose cross-section is taken be equal to 6.2barns. So the thermal neutron flux of the neutron source for the reaction  $^{127}\text{I} (n, \gamma) ^{128}\text{I}$  has been determined to be  $1.0628 \times 10^4$  neutron/m<sup>2</sup> sec. In addition to this by measuring the  $^{42}\text{K}$  activity induced in potassium (within KI) by a thermal neutron irradiation of the experimentally determined flux, the thermal neutron capture cross-section for the reaction  $^{41}\text{K} (n, \gamma) ^{42}\text{K}$  has been determined to be 1.43barns.

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## **CHAPTER - I**

### **Introduction**

Enrico Fermi (1901-1954) won the 1938 physics Nobel Prize for the artificial production of new isotopes by neutron bombardment or neutron capture. These new “neutron activated” isotopes are often unstable.

Behavior of neutrons passing through matter is fundamentally different from that of energetic charged particles. Because of the absence of charge they interact hardly at all with the shell electrons and with the coulomb field of the nucleus. The chief process is scattering and absorption by the nucleus by virtue of the nuclear force alone. If a neutrons traverse an absorber, they do not, like charged particles, lose their energy gradually, but pass through unchanged until they are either captured or scattered away so, based on the energy of neutrons and the different techniques (methods) used in many nuclear reactions induced by neutrons many valuable information about a nucleus can be revealed.

One of these techniques is the one, which is used, in this experimental work, i.e. Neutron activation.

**Chapter 2** describes about neutron’s discovery, neutrons sources, nuclear reactions, and thermalization of neutrons using moderators, the concepts of neutron flux and nuclear cross-section, and neutron activation analysis. The purpose of this chapter is to give the nature, interaction and application of neutrons.

**Chapter 3** gives experimental set up in detail and focuses on the experiments such as the activation of KI and the measurement of the activities of  $^{128}\text{I}$  and  $^{42}\text{K}$ . In the observation section the data’s and the

complex decay curves are analyzed and the decay constant is determined so that the thermal neutron flux of the  $^{241}\text{Am}$ -Be neutron- source using  $^{127}\text{I}$  and the thermal neutron Capture Cross-Section of  $^{41}\text{K}$  using the calculated flux are determined specially the later is within an agreement of 2%.

**Chapter 4** discusses about different possible errors observed during the experiment and the two experimental results.

**Chapter 5** summarizes the whole work accomplished in the experiment.

## **2. NEUTRON PHYSICS**

### **2.1 The Neutron**

The existence of the neutron was first suggested by Rutherford in 1920. He thought that an electron could exist in a nucleus and could combine with a proton to form a neutron. Being electrically neutral, the neutron was very difficult to discover by methods of particle detection, which depend on the deflection of the particles in a magnetic or electric field or on their ionization of matter. In 1932, however, one of Rutherford's students, Chadwick, demonstrated the existence of the neutron, using an experiment first conducted by both and Becker in 1930 and later by Irene and Frederic Joliot- curve [1].

In the experiment, a beryllium plate Be was bombarded by  $\alpha$  particles from a polonium source S. this caused highly penetrating radiation to emanate from Be.

The reading on the detector (ionization chamber) was then noted. When a target containing large quantities of hydrogen (such as in the form of paraffin) P was inter posed between Be and the detector, the reading was found to increase, rather than decrease as it was thought that it would,

because of the expected absorption in P. Chadwick showed conclusively that the penetrating radiation could not be  $\gamma$ - ray photons, as was initially suspected, but was that of a neutral particle, roughly equal in mass to the nucleus of the hydrogen atom (proton). He thus demonstrated that Rutherford's concept of the existence of the neutron was correct.

The neutron's main characteristics are:

1. It is electrically neutral.
2. Having no orbital electrons to cause emission or absorption spectra, it cannot be seen on a spectrometer; i.e. it is invisible.
3. It doesn't interact with electrons to any large degree. Thus it ionizes matter mostly with nucleus in collision, causing some displacement of the nuclei or in absorption, causing the formation of other isotopes and subsequent induced radiations of various types.
4. It exists permanently only in nuclei. Otherwise it is slightly, having a 13- minute half- life. If it is in a free state, it rapidly ejects a  $\beta$ - particle and a neutrino and transforms in to a proton [1].

### **Energy Classification**

Neutrons are usually classified according to either energy or velocity, although the boundaries between the various divisions are ill defined. The common classification is [2]:

High-energy	> 10MeV
Fast	10Kev-10MeV
Intermediate	100ev-10KeV
Slow and Epithermal	0.3eV-100eV

Thermal	0.025ev
Resonance neutrons	> 1 eV.

## **2.2 Neutron Sources**

The possible choices for radioisotope neutron sources are much more limited and are based on either spontaneous fission or on nuclear reactions for which the incident particle is the product of a conventional decay process.

### **A. Spontaneous Fission**

Many of the transuranic heavy nuclides have an appreciable spontaneous fission decay probability. Several fast neutrons are promptly emitted in each fission event, so a sample of such a radionuclide can be a simple and convenient isotopic neutron source. Other products of the fission process are the heavy fission product, that is prompt fission gamma rays, and the beta and gamma activity of the fission products accumulated within the sample. When used as a neutron source, the isotope is generally encapsulated in a sufficiently thick container so that only the fast neutrons and gamma rays emerge from the source [3].

The neutrons emitted in fission process are of two types prompt and delayed neutrons.

#### **Prompt neutrons:**

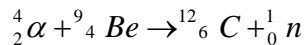
The prompt neutrons that are ejected within a period of  $10^{-14}$  to  $10^{-12}$  second from the fission fragments after the fission. Evident by these fragments are associated with excitation energy of the fragment. The emitted prompt neutrons are nearly 99% of the total neutrons ejected in the fission. Most of the prompt neutrons are having energies of the order of 1-2 MeV but some of them may also have energies beyond 10 MeV [4].

### **Delayed neutrons:**

After the fission has occurred the delayed neutrons are emitted for several minutes and the intensity decreases exponentially with time. As an example in  $^{235}\text{U}$  about 0.755% neutrons are the delayed neutrons, which are about three times than that of  $^{239}\text{Pu}$ . The decay of intensity is expressed in half-life [4].

### **B. Radioisotope ( $\alpha, n$ ) Sources.**

Because energetic alpha particles are available from the direct decay of a number of convenient radio nuclides, it is possible to fabricate a small self-contained neutron source by mixing an alpha-emitting isotope with a suitable target materials. Several different target materials can lead ( $\alpha, n$ ) reactions for the alpha particle energies that are readily available in a radioactive decay. The maximum neutron yield is obtained when beryllium is chosen as the target, and neutrons are produced through the reaction. [3]



This has a Q-value of +5.71 MeV.

Most of the alpha particles simply are stopped in the target, and only 1 in about  $10^4$  reacts with a beryllium nucleus. Virtually the same yield can be obtained from an intimate mixture of the alpha-particle emitter and beryllium, provided the alpha emitter is homogeneously distributed through out the beryllium in a small relative concentration. Several of the  $\alpha$  - emitters notably  $^{226}\text{Ra}$  and  $^{227}\text{Ac}$ , lead to long chain of daughters products that, although adding to the alpha particle yield, also contribute a large gamma-rays background. These sources are therefore inappropriate for some applications in which the intense gamma-ray

background interferes with measurement. Other isotopes involve simpler alpha decays and gamma ray background is much lower. The choice between these alternatives is made primarily on the basis of availability, cost, and half- life.

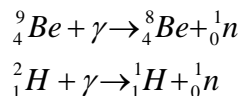
Because the physical size of the sources is no longer negligible, one would like the half- life to be as short as possible, consistent with the application, so that the specific activity of emitter is high.

In order to increase the neutron yield without increasing the physical source size alpha emitters, such as  $^{241}\text{Am}$  the one that is used in this experiment, with higher specific activities must be substituted.

In addition to this a number of other alpha- particle- induced reactions have occasionally been employed as neutron sources, but all have substantially lower neutron yield per unit alpha activity. This is because that all the Q- values of these reactions are less that what of the beryllium reaction [3].

### **C. Photo neutron Sources**

Some radioisotope gamma- ray emitters can also be used to produce neutrons when combined with an appropriate target material. The resulting photo neutron sources are based on supplying sufficient excitation energy to a target nucleus by absorption of a gamma- ray photon to allow the emission of a free neutron.  $^9\text{Be}$  and  $^2\text{H}$  are of any practical significance for radioisotope photo neutron sources [3].



## D. Reactions From Accelerated Charged Particles

Because alpha particles are the only heavy charged particles with low  $Z$  conveniently available from radioisotopes, reactions involving incident protons, deuterons, and so on must rely on artificially accelerated particles, two the most common reactions of this type used to produce neutrons are [3]:



### 2.3 Nuclear Reactions

This section shall briefly outline the fundamentals of nuclear reactions. A nuclear reaction is usually described by an equation similar to a chemical reaction equation thus the equation

$$a + X \rightarrow Y + b \text{ -----(1)}$$

describes the nuclear reaction in which an incident particle “a” strikes the target nucleus X and the products of the nuclear reaction are the outgoing particle “b” and recoil nucleus Y

The change in the mass of the particle during nuclear reaction represents the release or the absorption of energy. If the total mass of the particle after the reaction is reduced, the process releases energy. Consequently the increase in the mass of the resultant particle, will cause the absorption of energy.

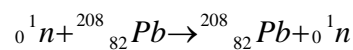
The equation of nuclear reaction is connected with the resettlement of proton and neutrons within the atom. In simple term, the equation shows the balance of neutron and proton.

Here only the five main types of reactions in which transformation may adopt are discussed. In this discussion what to be noted is that more collision of neutrons with atoms, and disposition of electrons, does not ionize the atoms, but the interaction with the nuclei gives significant effect.

### **1. Elastic Scattering**

The neutron interacts with the nucleus and after transformation the compound nucleus emits a particle, which is identical to the captured one. There is also no change in the resultant nucleus. The total internal kinetic energy of the bombarded nucleus and the restriking particle will not change at all.

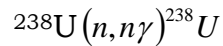
When the neutron strikes the nucleus, it imparts part of its initial kinetic energy and momentum to the nucleus, which causes the displacement of the nucleus in the crystal lattice by a significant distance and can change the structural properties of the material. In this process kinetic energy of neutron is reduced and is beneficial to slow down the neutrons in a reactor. Therefore in this transformation, there is neither release nor absorption of energy but as a result of collision, redistribution of kinetic energy takes place. As an example we can consider the following reaction [4].



### **2. Inelastic Scattering**

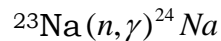
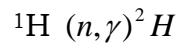
Hence the composition of the incident particle and ejected particle remain unchanged. When the particle interacts with nuclei it loses its kinetic energy and the target nucleus is excited. The energy is released in the form of gamma emission.

This process is limited to the condition that the neutron should have minimum energy sufficient to excite the target nucleus. The reaction is completed with the absorption or release of energy. As an example when a fast moving neutron hits the  $^{238}\text{U}$  nucleuses, then the nucleus is excited and there is an emission of gamma [4].



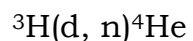
### 3. n- capture

In this process the incident particle may be captured or absorbed by the nucleus and may raise the mass number by unity. The nucleus is excited and the energy is emitted in the form of gamma. Artificial radioactive materials are produced by this process. In a reactor, Co- 60 isotope is produced by bombarding the natural Co- 59 with neutrons. The reaction has both the possibilities of producing the stable and unstable nucleus and may result in  $(n, \gamma)$  or  $(p, \gamma)$  reactions. As an example we can have the following reactions [4].



### 4. Disintegration

In this reaction, the impinging particle is trapped in the nucleus but the ejected particle is a different one. The composition of the resultant nucleus is also different from the parent nucleus as an example we can consider



## 5. Fission

Here in this reaction when the nucleus is excited highly, it splits in to two mostly equal masses. The produced two nucleuses are lighter nuclei and having more binding energies per nucleon. Hence this reaction always releases energy.

The fission of the nuclei is also possible by thermal neutrons and the process is called thermal neutron fission. Neutrons can be produced by  $(\alpha, n)$  reactions on light elements [5].

### **The compound Nucleus**

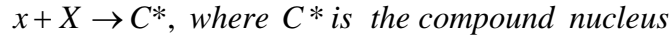
The actual nuclear process in a nuclear reaction dose not start before the two initial particles a and X have came near enough to one another, within the range of nuclear forces. The nuclear process has ceased when the two products have separated by more than the range. During the time of the interaction, a compound system is formed whose properties are decisive for the course of the nuclear reaction. It was N. Bohr who first pointed out that it is useful to divide the nuclear reaction in to two states [6].

- I. The formation of the compound nucleus C, and
- II. The disintegration of the compound system into the product of the reaction

So, Bohr's two assumptions are:

1. When a projectile enters a nucleus it interacts with nucleons present inside the target nucleus. Large number of collision may take place in which there is an exchange of energy and momentum. There is thorough mixing of incoming particle with the nucleons present in the nucleons

Entering particle makes now a new system such that



$$\text{Then, } E_c = \varepsilon_\alpha + E_B \text{ -----(2)}$$

Where

$\varepsilon_\alpha$  is energy of projectile in channel  $\alpha$ , and

$E_B$  is its binding energy in the nucleus.

This mixing of projectile with other nucleons takes too much time (about  $10^{-16}$  sec) so that compound nucleus formed forgets its history of formation. During this time which is much larger as compared with nuclear time. The life- time of the compound nucleus is about  $10^6$  times the nuclear time ( $10^{-21}$  sec– $10^{-23}$  sec).

2. Due to the large time taken by compound nucleus formation, its decay is independent of the mode of formation. The decay does not depend up on the nature of the incoming projectile that is if the same compound nucleus is formed by different channels its decay will be independent, the decay will depend only on the quantum mechanical parameters of the compound nucleus that is energy of excitation, angular momentum and parity [6].

### **Direct Interaction**

The direct interaction assumes that the incident particle collides with the target nucleus without significantly disturbing its over all structure; only one or at most a few nucleons are transformed from the projectile to the target or vice versa. That is, part of the incident particle is absorbed by

the nucleus while the remainders continue on after being deflected. This is known as a stripping reaction. The inverse of stripping, called a pick-up reaction, may also occur. In pick-up, an incident particle (e.g. a proton) interacts with a nucleon or group of nucleons in the target nucleus forming a complex particle (e.g. a deuteron), which leaves the target nucleus [6].

## **2.4 Neutron Moderation**

In thermal reactors, the neutrons that cause fission are at much lower energy than the energy level at which they were born from fission. In this type of reactor, specific materials must be included in the reactor design to reduce the energy level of the neutrons in an efficient manner.

### **Neutron Slowing Down and Thermalization**

Fission neutrons are produced at an average energy level of 2MeV and immediately begin to slow down as the result of numerous scattering reactions with a variety of target nuclei. After a number of collisions with nuclei, the speed of a neutron is reduced to such an extent that it has approximately the same average kinetic energy as the atoms (or molecules) of the medium in which the neutron is undergoing elastic scattering. This energy, which is only a small fraction of an electron volt at ordinary temperature (0.025 eV at 20°C), is frequently referred to as the thermal energy, since it depends upon the temperature. Neutrons whose energies have been reduced to values in this region (<1eV) are designated thermal neutrons. The process of reducing the energy of a neutron to the thermal region by elastic scattering is referred to as thermalization, slowing down or moderation. The material used for the purpose of thermalizing neutrons is called a moderator, A good moderator reduces the speed of neutrons in a small number of collisions, but does not absorb them to any great extent. Slowing the neutrons in as

few collisions as possible is desirable in order to reduce the amount of neutron leakage from the core and also to reduce the number of resonance absorptions in non-fuel materials [5].

The distance that a fast neutron will travel, between its introduction into the slowing-down medium (moderator) and thermalization, is dependent on the number of collisions and the distance between collisions. Though the actual path of the neutron slowing-down is tortuous because of collisions, the average straight line distance can be determined; this distance is called the fast diffusion length or slowing down length. The distance traveled, once thermalized, until the neutron is absorbed, is called the thermal diffusion length.

The ideal moderating material (moderator) should have the following nuclei properties:

1. Large scattering cross-section,
2. Small absorption cross-section,
3. Large energy loss per collision.

A convenient measure of energy loss per collision is the logarithmic energy decrement. The average logarithmic energy decrement is the average decrease per collision in the logarithm of the neutron energy. This quantity is represented by the symbol  $\xi$

$$\xi = \ln E_i - \ln E_f$$

$$\xi = \ln \left[ \frac{E_i}{E_f} \right] \text{-----(3)}$$

Where  $\xi$  = average logarithmic energy decrement

$E_i$  = average initial neutron energy

$E_f$  = average final neutron energy

Since the fraction of energy retained by a neutron in a single elastic collision is a constant for a given material,  $\xi$  is also a constant. Because it is a constant for each type of material and it does not depend up on the internal neutron energy,  $\xi$  is a convenient quantity for assessing the moderating ability of a material.

The values for the lighter nuclei are tabulated in a variety of sources. The following commonly used approximation may be used when a tabulated value is not available.

$$\xi = \frac{2}{A + \frac{2}{3}} \text{-----} (4)$$

This approximation is reactively accurate for mass numbers (A) grater than 10, but for some low values of A it may be in error by over 3%. Since  $\xi$  represents total number of collisions necessary for a neutron to lose a given a mount of energy may be determined by dividing  $\xi$  in to the difference of the natural logarithms of the energy range in question. The number of collision (N) to travel from any energy,  $E_{high}$ , to any lower energy,  $E_{low}$ , can be calculated as shown below [5].

$$N = \frac{\ln E_{high} - \ln E_{low}}{\xi}$$

$$N = \frac{\ln \left[ \frac{E_{high}}{E_{low}} \right]}{\xi} \text{-----} (5)$$

### **Macroscopic Slowing Down Power**

Although the logarithmic energy decrement is a convenient measure of the ability of a material to slow neutrons, it dose not measure all necessary properties of a moderator. A better measure of the capabilities

of a material is the macroscopic slowing down power (MSDP). MSDP is the product of the logarithmic energy decrement and the macroscopic cross section for scattering in the material [5]. That is

$$MSDP = \xi \sum_s \Sigma_s \quad \text{-----} \quad (6)$$

Where,  $\Sigma_s$  is macroscopic cross section for scattering.

### **Moderating Ratio**

The macroscopic slowing down power indicates how rapidly a neutron will slow down in the material, but it still does not fully explain the effectiveness of the material as a moderator. An element such as boron has a high logarithmic energy decrement and a good slowing down power, but it is a poor moderator because of its high probability of absorbing neutrons.

The most complete measure of the effectiveness of a moderator is the moderating ratio. The moderating ratio is the ratio of the macroscopic slowing down power to the macroscopic cross section for absorption. The higher the moderating ratio, the more effective the material performs as a moderator. So, the following equation shows how to calculate the moderating ratio of a material [5].

$$MR = (\text{Moderating Ratio}) = \frac{\xi \sum_s \Sigma_s}{\sum_a \Sigma_a} \quad \text{-----} \quad (7)$$

Where,  $\Sigma_a$  is macroscopic cross section for absorption.

Moderating properties of different materials are compared in table 1 [7].

<b>Table 1</b>			
<b>Moderating Properties of Materials</b>			
<b>Material</b>	$\xi$	<b>Number of Collisions to Thermalize</b>	<b>Moderating Ratio</b>
Hydrogen	1	18	66
H <sub>2</sub> O	0.927	19	67
Deuteron	0.725	25	>5820
D <sub>2</sub> O	0.51	35	~5820
Helium	0.425	43	94
Beryllium	0.209	86	146
Carbon	0.158	114	237
Oxygen	0.120	150	487

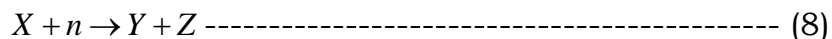
## **2.5 Neutron Flux and Nuclear Cross- section**

### **Neutron Flux**

The nuclei of certain naturally occurring isotopes can be transformed in to radioactive ones by exposing the material to neutron radiation, and the active products produced can be measured by means of appropriate counter system. In addition to the factors determined by the conditions of measurement, this activity is affected only by the neutron flux in the point of irradiation and by activation cross- section of the target material.

Provided that the activation cross- section is known, the neutron flux can be determined by measurement of the activity of the target irradiated.

The basic idea is that the target isotope (X) is converted in to an unstable isotope (Y) by capture of a neutron [8].



The activity of Y is proportional to the amount of X present in the target. This amount can be calculated after measurement of the activity of Y provided that the counting efficiency is known. Measurement of the activity allows calculation of the mass of the element in question. According to theory the activity of Y after irradiation in a neutron source with flux density  $\phi$  for a time  $t_i$  is

$$A_y(t) = \sigma \cdot \phi \cdot N_x (1 - e^{-\lambda t_i}) \text{-----} (9)$$

The cross-section of the process is  $\sigma$ , the decay constant of nuclide Y is  $\lambda$  and the number of target nuclei is  $N_x$ . After the target is removed from the neutron source, a time  $t_w$  elapses before measurement (counting) of the sample can be initiated. Considering correction for decay during measurement the number of counts collected in the time  $t_c$  is.

$$n = \varepsilon_p \cdot P_\gamma \cdot \frac{\sigma \cdot \phi \cdot N_x}{\lambda} \cdot (1 - e^{-\lambda t_i}) \cdot e^{-\lambda t_w} \cdot (1 - e^{-\lambda t_c}) \text{-----} (10)$$

Here the yield of the  $\gamma$  - radiation considered is  $P_\gamma$ , the counting efficiency of the actual  $\gamma$  - energy in the photo peak is denoted  $\varepsilon_p$ . The time dependent part of the expression may be denoted  $K_D(\lambda, t_i, t_w, t_c)$ :

$$K_D(\lambda, t_i, t_w, t_c) = (1 - e^{-\lambda t_i}) \cdot e^{-\lambda t_w} \cdot (1 - e^{-\lambda t_c}) \text{-----} (11)$$

If the target contains a mass of the element with molar mass  $M$  and the abundance of the isotope X to be activated is  $P_x$  then the numbers of target atoms are:

$$N_x = \frac{m}{M} N_A \cdot P_x \text{-----} (12)$$

Combining this with (5)

$$n = \frac{\varepsilon_p \cdot P_\gamma \cdot \sigma \cdot \phi \cdot N_A \cdot P_x \cdot K_D \cdot T_{\frac{1}{2}} \cdot m}{M \ln 2} \text{----- (13)}$$

Here the decay constant  $\lambda$  has been replaced by  $\left(\frac{\ln 2}{T_{\frac{1}{2}}}\right)$ . Finally we obtain the following expression for the neutron flux of the neutron source:

$$\phi = \frac{nM \cdot \ln 2}{\varepsilon_p P_\gamma \cdot \sigma \cdot N_A \cdot P_x \cdot m \cdot K_D \cdot T_{\frac{1}{2}}} \quad \text{or}$$

$$\phi = \frac{\lambda \cdot M \cdot n \cdot e^{\lambda \cdot t_d}}{\varepsilon \cdot p_\gamma \cdot \sigma \cdot m \cdot N_A \cdot p_x \cdot (1 - e^{-\lambda \cdot t_i}) \cdot (1 - e^{-\lambda \cdot t_{ci}})} \text{----- (14)}$$

So, the neutron flux can be determined experimentally by the above relationship.

### **Cross Section**

In fact, nuclear cross-section is effective area of a nucleus as shown to a projectile or as seen by a projectile. If the projectile passes through this area, interaction will take place. It is measured in terms of number of events produced by a definite number of projectiles per nuclei. It can also be measured in terms of number of projectile absorbed in the target, while producing a definite kind of event. Total cross-section can be calculated by the estimation of all the types events produced [4].

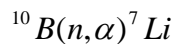
It has been seen that the cross-sections of various nuclei are in the range of  $10^{-24} \text{cm}^2$ . So, the unit of cross-section is barn and is equal to  $10^{-24} \text{cm}^2$ . It is to be considered that the cross-section to target material dose not mean the actual geometrical dimension of nucleus. It not only depends on the nuclei but also on the velocity of the neutrons i.e., on the

associated neutron energy. The process of interaction has been given various names to the cross-section i.e. scattering cross-section (s), fission cross-section (f) absorption cross-section (a), and captures cross-section (c)

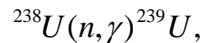
The total cross-section is the sum of the cross-section of various processes during nuclear reaction.

On the basis of energy range the cross-section variation can be classified in to three regions.

- a) In the low energy range (0.025eV) cross-section is inversely proportional to the neutron velocity ( $\frac{1}{v}$  law). This region covers, low energy reactions such as  $(n, \gamma)$  and exothermic  $(n, \alpha)$  reactions, as an example we can consider (Fig 1(a) )



- b) In some of the reactions, in near thermal region, the cross-section has one resonance peak and in the rest of the region, it follows  $\frac{1}{v}$  law. As an example  ${}^{113}\text{Cd}$  has only one peak in the low energy thermal region. Similarly in



reaction, there is only one resonance peak (Fig 1(b))

- c) In this type of nuclear reaction, the cross-section follows  $1/v$  law for a narrow interval of low energies, but the remaining range of energy, it forms many peaks. This odd behavior phenomenon is known as resonance.

The neutron energy associated with resonance cross-section is called resonance energy.

The resonance peak width is affected by the life of the excited compound nucleus. Large life of the excited nucleus will give a narrow peak of high cross-section and vice versa (Fig 1(c)) [4].

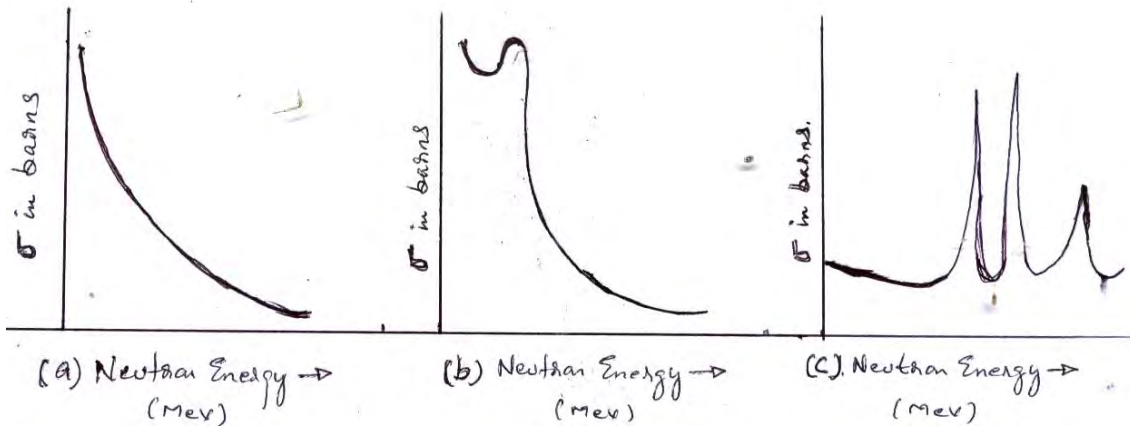


Fig. 1 Variation of cross-section with the energy in different type of nuclear reactions [4]

**Macroscopic cross-section**

Whether a neutron will interact with a certain volume of material depends not only on the microscopic cross-section of the individual nuclei but also on the number of nuclei within that volume. Therefore, it is necessary to define another kind of cross-section ( $\Sigma$ ). The macroscopic cross-section is the probability of a given reaction occurring per unit travel of the neutron.  $\Sigma$  is related to the microscopic cross-section ( $\sigma$ ) by the relationship shown below.

$$\Sigma = N\sigma \text{ ----- (15)}$$

Where,

$\Sigma$  = macroscopic cross-section ( $cm^{-1}$ )

$N$  = atom density of material  $\left(\frac{atoms}{cm^3}\right)$

$\sigma$  = Microscopic cross-section ( $cm^2$ )

## 2.6 Neutron Activation Analysis [NAA]

The sequence of events occurring during the most common type of nuclear reaction used for NAA, namely the neutron capture or ( $n, \gamma$ ) reactions is illustrated in figure 2.

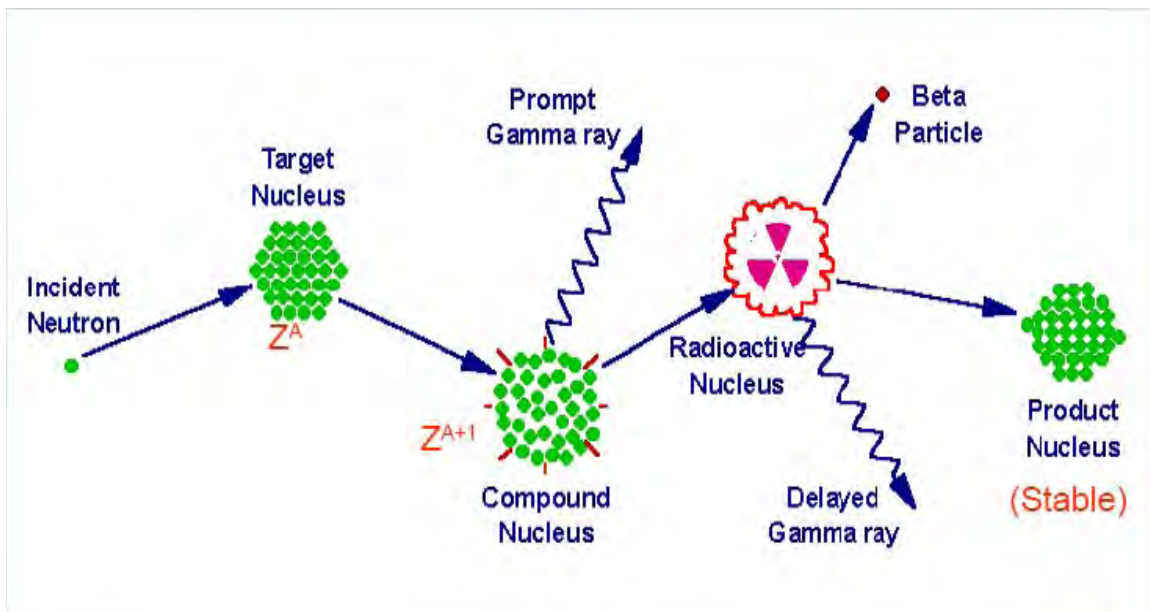


Fig. 2 the process of neutron capture by a target nucleus followed by the emission of gamma rays [9].

When a neutron interacts with the target nucleus via a non-elastic collision, a compound nucleus forms in an excited state. The excitation energy of the compound nucleus is due to the binding energy of the neutron with nucleus. The compound nucleus will almost

instantaneously de-excite into a more stable configuration through emission of one or more characteristic prompt gamma rays. In many cases, this new configuration yields a radioactive nucleus which also de-excites (or delays) by emission of one or more characteristic delayed gamma rays, but at a much slower rate according to the unique half-life of the radioactive nucleus. Depending upon the particular radioactive species, half-life can range from fractions of a second to several life can range from fractions of a second to several years.

In principle, therefore, with respect to the time to measurement, NAA falls into the categories.

1. Prompt gamma- ray neutron activation analysis (PGNAA), where measurements take place during irradiation, or
2. Delayed gamma-ray neutron activation analysis (DGNAA), where the measurements follow radioactive decay.

The later operational mode is more common; thus, when one mention NAA it is assumed that measurement of the delayed gamma rays is intended. About 70% of the elements have properties suitable for measurement by NAA [9].

## **Neutrons**

Although there are several types of neutron sources (reactors, accelerators, and radio isotopic neutron emitters as it was observed earlier) one can use for NAA, nuclear reactors with their high fluxes of neutrons from uranium fission offer the highest available sensitivities for most elements. Different types of reactors and different positions within a reactor can vary considerably with regard to their neutron energy

distributions and fluxes due to the materials used to moderate (or reduce the energies of) the primary fission neutrons.

The thermal neutron component consists of low- energy neutron (energies below 0.5 eV) in thermal equilibrium with atoms in the reactors. At room temperature the energy spectrum of thermal neutrons is best described by a Maxwell- Boltzmann distribution with a mean energy of 0.025 eV and a most probable velocity of 2200 m/s. In most reactor irradiation positions, 90-95% of the neutrons that bombard a sample are thermal neutrons. In general, a one-megawatt reactor has a peak thermal neutron flux of approximately  $1 \times 10^{13}$  neutrons per square centimeter per second [9].

The epithermal neutron component consists of neutrons (energies from 0.5 eV to about 0.5 MeV), which have been only partially moderated. A cadmium foil 1 mm thick absorbs all the thermal neutrons but will allow epithermal and fast neutrons above 0.5 eV energy to pass through. In a typical unshielded reactor irradiation position, the epithermal neutron flux represents about 2% of the total neutron flux. Both thermal and epithermal neutrons induce (n, gamma) reactions on target nuclei. An NAA technique that employs only epithermal neutrons to induce (n, gamma) reactions by irradiating the samples being analyzed inside either cadmium or boron shields is called epithermal neutron activation analysis (ENAA) [9].

The fast neutron component of the neutron spectrum (energy above 0.5 MeV) consists of the primary fission neutrons, which still have much of their original energy following fission. Fast neutrons contribute very little to the (n, gamma) reaction, but instead induce nuclear reactions where the ejection of one or more nuclear particles (n,p), (n,n), and (n,2n)- are prevalent. In a typical reactor irradiation position, about 5% of the total

flux consists of fast neutrons. An NAA technique that employs nuclear reactions induced by fast neutrons is called fast neutron activation analysis (FNAA).

### **Prompt Vs Delayed NAA**

As mentioned earlier, the NAA technique can be categorized according to where gamma rays are measured during neutron irradiation (PGNAA) or at some time after the end of the irradiation (DGNAA). The PGNAA technique is generally performed by using a beam of neutrons extracted through a reactor beam port. Fluxes on samples irradiated in beams are on the order of one million time lower than on samples inside a reactor but detectors can be placed very close to the sample compensating for much of the loss in sensitivity due to flux. The PGNAA technique is most applicable to elements with extremely high neutron capture cross-sections (B Cd, Sm, and Gd); elements which decay too rapidly to be measured by DGNAA; elements that produce only stable isotopes; or elements with weak decay gamma- ray intensities.

DGNAA (sometimes called conventional NAA) is useful for the vast majority of elements that produce radioactive nuclides. The technique is flexible with respect to time such that the sensitivity for a long – lived radionuclide can be improved by waiting for the short- lived radionuclide to decay. This selectivity is a key advantage of DGNAA over other analytical methods.

### **Instrumental vs. Radiochemical NAA**

With the use of automated sample handling gamma-ray measurement with solid-state detectors and computerized data processing it is generally possible to simultaneously measure more than thirty elements in most in most sample types with out processing. The application of

purely instrumental procedures is commonly called instrumental neutron activation analysis (INAA) and is one of NAA's most important advantages over other analytical techniques. If chemical separations are done to samples after irradiation to remove interferences or to concentrate the radioisotope of interest, the technique is called radiochemical neutron activation analysis (RNAA). The latter technique is performed infrequently due to its high labor cost [9].

### **Measurement of Gamma Rays**

The instrumentation used to measure gamma rays from radioactive samples generally consists of a semiconductor detector, associated electronics, and a computer-based multi-channel analyzer (MCA/computer). Most NAA labs operate one or more hyper pure or intrinsic germanium HPGe detectors, which operate at liquid nitrogen temperatures (77 degrees K) by mounting crystal in a vacuum cryostat, thermally connected to a copper rod or "cold finger". Although HPGe detectors come in many different designs and sizes, the most common type of detector is the coaxial detector with in NAA is useful for measurement of gamma rays with energies over the range from about 60 keV to 3.0 MeV [9].

The two most important performance characteristics requiring consideration when purchasing a new HPGe detector are resolution and efficiency. Other characteristics to consider are peak shape, peak to Compton ratio, crystal dimensions or shape, and price.

The detector's resolution is a measure of its ability to separate closely spaced peaks in a spectrum. In general, detector resolution is specified in terms of the full width at half maximum (FWHM) of the 122-keV photopeak of Co-57 and the 1332- KeV photopeak of Co- 60. For most

NAA applications, a detector with 1.0-keV resolution or below at 122 KeV and 1.8 KeV or below at 1332 keV is sufficient.

Detector efficiency depends on the energy of the measured, the solid angle between sample and detector crystal, and the active volume of the crystal. A large volume detector will have a higher efficiency. In general, detector efficiency is measured relative to a 3- inch by 3 inch sodium iodide using a Co -60 source (1332 keV gamma ray) at a distance of 25 cm from the crystal face. A general rule of thumb for germanium detectors is 1 percent efficiency per each 5cc of active volume. As detector volume increase, the detector resolution gradually decreases. For most NAA applications, an HPGe detector of 15-30 percent efficiency is adequate [10].

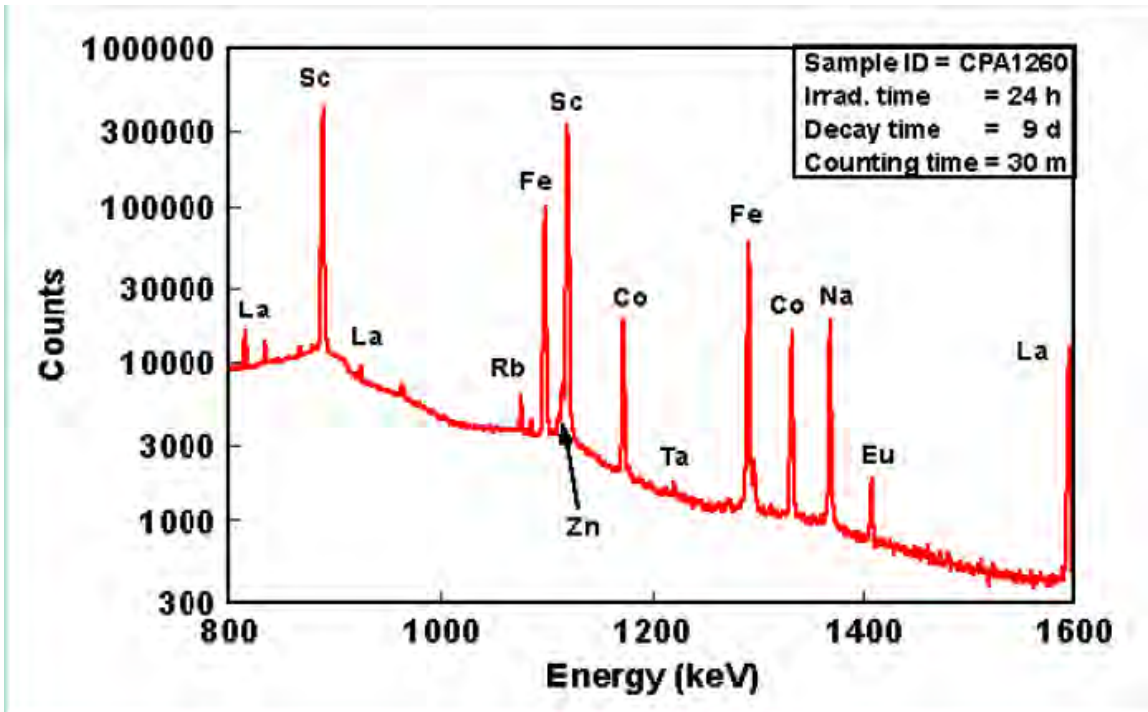


Fig. 3 Gamma ray spectrum showing several short lived elements measured in a target. [9]

## Using Gamma-ray Counts to calculate Element Concentration

The procedure generally used to calculate concentration (i.e., ppm of element) in the unknown sample is to irradiate the unknown sample and a comparator standard containing a known amount of the element of interest together in the reactor. If the unknown sample and the comparator are both measured on the same detector, then one needs to correct the difference in decay between the two. One usually decay corrects the measured counts (or activity) for both samples back to the end of irradiation using the half-life of the measured isotope. The equation used to calculate the mass of an element in the unknown sample relative to the comparator standard is [9]

$$\frac{A_{sam}}{A_{std}} = \frac{m_{sam} (e^{-\lambda t_d})_{sam}}{m_{std} (e^{-\lambda t_d})_{std}}$$

Where,

A= activity of the sample (sam) and standard (std),

m = mass of the element

$\lambda$  = decay constant for the isotope and

$t_d$  = decay time.

When performing short irradiation, decay and counting time are normally fixed the same for all samples and standards such that the time dependent factors cancel [9]. Thus the above equation simplifies in to

$$C_{sam} = C_{std} \frac{W_{std} A_{sam}}{W_{sam} A_{std}}$$

Where

C= concentration of the element and

W= weight of the sample and standard

## Sensitivities Available by NAA

The sensitivities for NAA are dependent upon the irradiation parameters (i.e., neutron flux, irradiation and decay times), measurement conditions (I.e. measurement time detector efficiency), nuclear parameter of the elements being measured (i.e. isotope abundance, neutron cross- section, half-life and gamma – ray abundance). The accuracy of an individual NAA determination usually ranges between 1 to 10 percent of the reported value assuming interference free spectra.

### 3. EXPERIMENT

#### 3.1 Experimental Setup

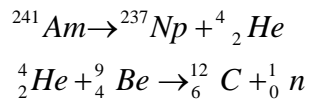
The AAU nuclear physics department has two facilities for the off-beam experiments, which could be performed under the supervision of the department. One is the nuclear laboratory and the other is the  $^{241}\text{Am}-\text{Be}$  neutron source, which is found about 500m from the nuclear lab.

#### The $^{241}\text{Am}$ Be neutron source

As it is already observed in the chapter 1,  $^{241}\text{Am}-\text{Be}$  is one of alpha-initiated radio isotopic sealed neutron source. The main advantage of using alpha initiated radioisotope neutron sources are:

1. Physically small size and rugged construction, (i.e., a double-walled stainless steel capsule),
2. Neutron flux stability, and
3. Long useful life-time (more than 20 years).

In this experiment the 2Ci Am-Be neutron source is used. In the  $^{241}\text{Am}-\text{Be}$  neutron source, the americium-241 alloyed with Beryllium, the alpha particles from americium have sufficient energy to react with Beryllium, which causes emission of neutrons according to the reaction.



The neutron yield of the source is approximately 77 neutrons per  $10^6$  alpha particles with an average of 4.5 MeV [11].

The  $^{241}\text{Am}-\text{Be}$  source is surrounded with paraffin wax as a moderator to thermalize the neutrons and insulated in the center of the assembly as

shown in the figure 4. The holder fitted over the irradiation channels during irradiation.

### **Description for the Neutron Source Shielding Tank**

Where:

- 0- Central channel (for  $^{241}\text{Am}$ -Be neutron source,  $\phi 30 \times 30 \text{ mm } 2 \text{ Ci}$ )
- 1-  $\text{BF}_3$  channel, there is a red marked Number 1 on the surface of the tank.
- 2, 3, 4- Measurement channels, there are wax rods inside, while any of channels is to be used, raise out the wax-rod and drop in a positioning holder there is a tong, sample of  $\phi 10$  mm can be clipped or put on the top of tong for irradiation.
5. Lock, preventing neutron source missing
6. Wax rods, neutron protecting- aluminum cylinder filled with wax. Total four rods: three for measurement channels and one for  $\text{BF}_3$  channel.
7. Cables connect  $\text{BF}_3$  tube to Preamplifier FH1048.
9. Outer Casing,  $\phi 600 \times 600$  mm
10.  $^{241}\text{Am}$ -Be neutron source  $\phi 30 \times 30 \text{ mm } 2 \text{ Ci}$
11. Pure wax filled
12. Aluminum partition
13. Wax containing Boron
14. Positioning sample-holder (Plexiglass), total three.
15. Handles. [12]

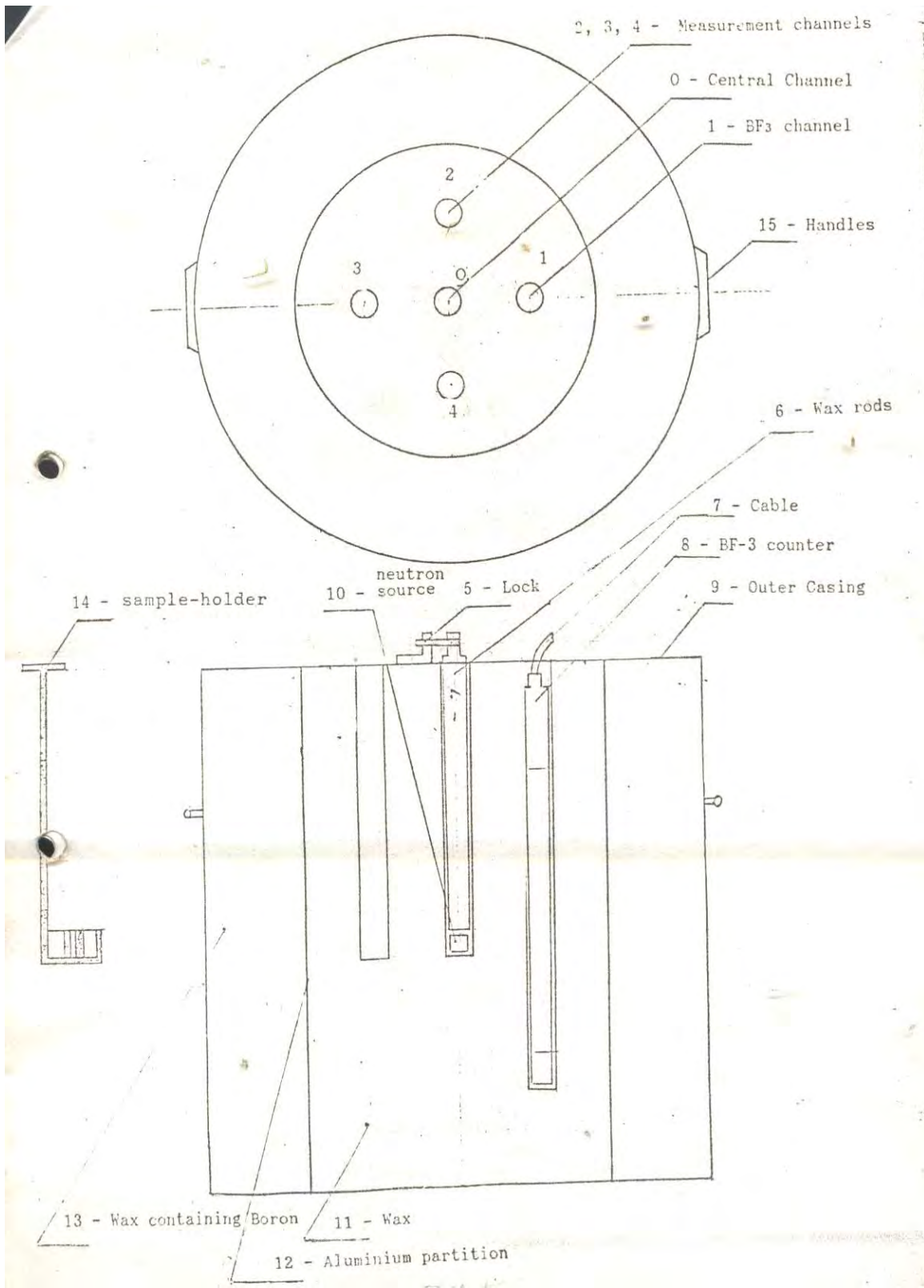


Figure 4 the neutron shield tank

## **GM-Counter**

Basically, the Geiger- Muller Counter Consists of two electrodes with a gas at reduced pressure between the electrodes. The outer electrode is usually a cylinder, while the inner (positive) electrode is a thin wire positioned in the center of the cylinder. The voltage between these two electrodes is maintained at such a value that virtually any ionizing particle entering the Geiger tube will cause an electrical avalanche within the tube. The Geiger Muller tube used in this experiment is called end window tube because it has a thin window at one end through which the ionizing radiation enters.

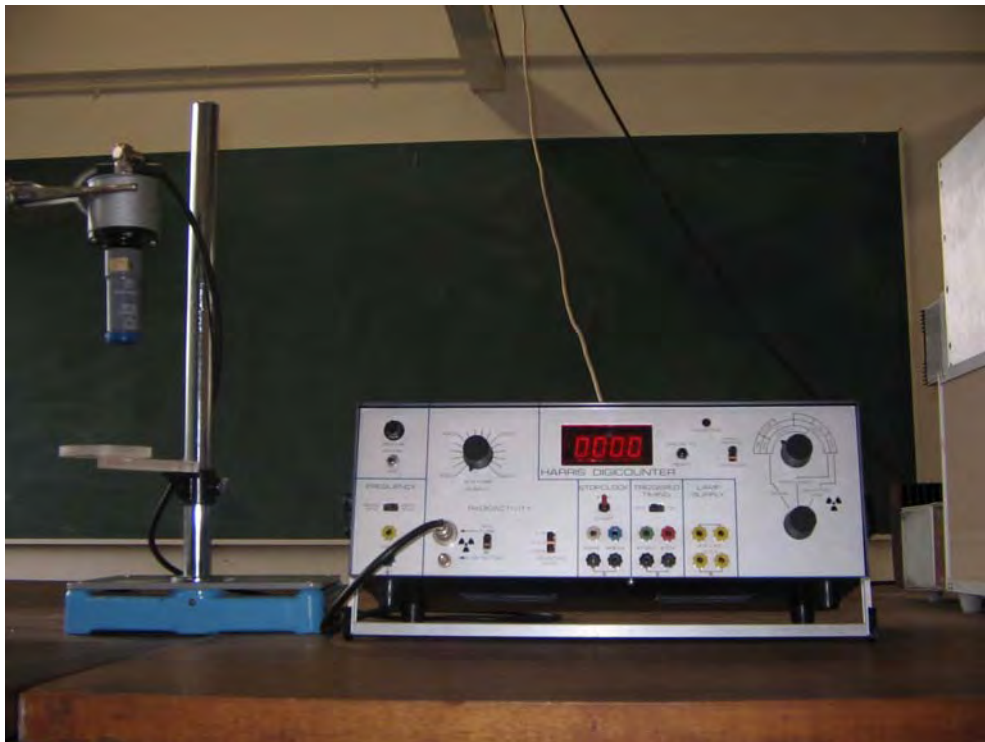
The Geiger Muller Counter does not differentiate between kinds of particles or energies; it tells only that a certain number of particles (betas and gammas) entered the detector during its operation. The voltage pulse from the avalanche is typically large in amplitude.

For proper use of the GM Counter, one must have the appropriate voltage across the electrodes. If the voltage is too low, the electric field in the tube is too weak to cause a current pulse. If the voltage is too high, the tube will undergo continuous discharge, and the tube can be damaged.

For low voltages, no counts are recorded. This is because the electric field is too weak for even one pulse to be recorded. As the voltage is increased, eventually one obtains a counting rate. The voltage at which the GM tube just begins to count is called the starting potential. The counting rate quickly rises as the voltage is increased, then the rise becomes so fast, that the graph looks like a “step” potential. After the quick rise, the counting rate levels off. This range of voltages is termed the “plateau region”.

Proper operation is when the voltage is in the plateau region of the curve. For best operation the voltage should be selected fairly close to the threshold voltage, and about the middle of the way into the plateau region.

In the plateau region the graph of counting rate versus voltage is in general not completely flat. Infact, the slope of the curve in the plateau region is a measure of the quality of the GM-tube. For a good GM- tube, the plateau region should rise at a rate less than 10% per 100 volts. An excellent tube could have the plateau slope as low as 3% per 100 volts. The GM tube set up which is used in the experiment is shown in the figure below [13].



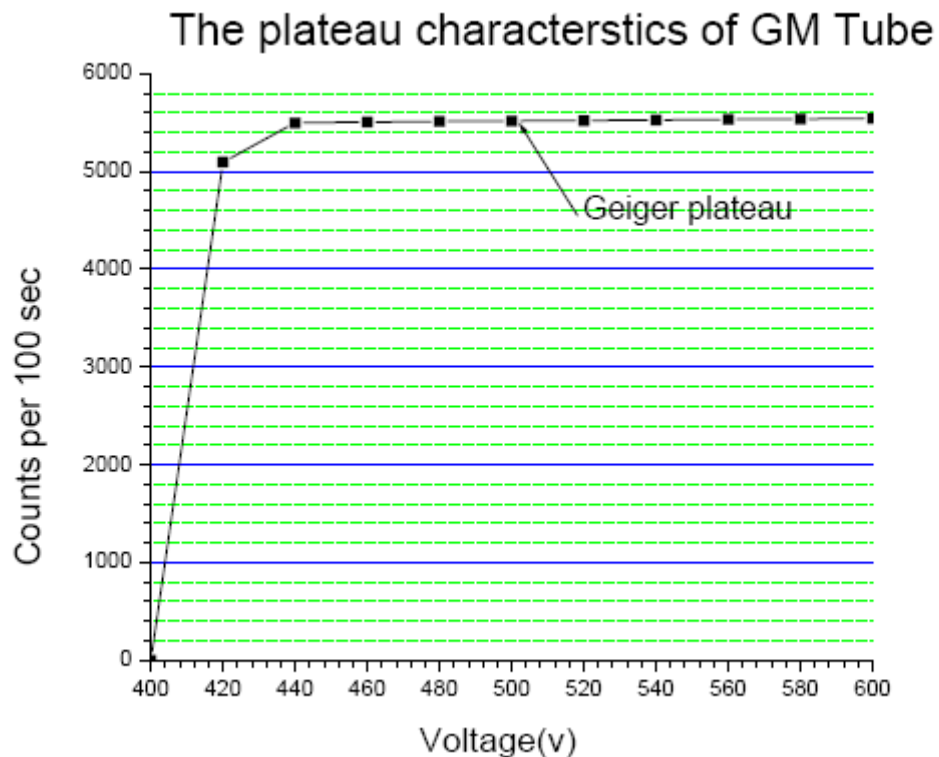
**Figure 5 GM tube set up**

## Determining the operating voltage of the GM tube

In the determination of the operating voltage of the apparatus; GM tube, digital counter, sample holder, stand, electronic watch, and Cs-137 are used. The digital counter is set at single reading, 100-second Counting time, and the counter display at X1. Putting the Cs-137 on the sample holder at 2mm from the window of the counter, the counter is switched on and the voltage is slowly increased until Counts just began to be recorded on the scaler. Beginning from the starting voltage, which is 420V one count in every additional 20 volt is recorded. Finally by making tabulation of the counts rate versus voltage, the Geiger plateau is plotted as shown in the table 2 below.

Voltage (V)	Count (counts per 100 sec)	$\pm\sqrt{N}$
400	0	0
420	5094	71.37
440	5497	74.14
460	5504	74.18
480	5510	74.22
500	5515	74.26
520	5519	74.28
540	5525	74.33
560	5531	74.37
580	5535	74.39
600	5542	74.44

Table 2 Voltage and count values for the determination of Geiger plateau



**Figure 6 the plateau characteristics of GM tube**

From the figure 6 indicated above, the slope of the plateau is less than unity, so this shows that the GM tube is an excellent one. For best operation the voltage should be selected fairly close to the threshold voltage and that is about the middle of the way into the plateau region. Therefore, the operating voltage of the GM tube is taken at 500V.

### **3.2 Irradiation of KI**

Iodine is a very volatile and reactive element at room temperature, due to this it can evaporate and show some variation on its mass, and this is also conformed in the experiment. For this reason, a powdered form metal iodide salt, potassium iodide (KI) is used.

99.99% of this metal Iodide salt is the natural Iodine ( $^{127}\text{I}(100\%)$ ), and the natural potassium ( $^{39}\text{K}(93.3\%)$ ,  $^{40}\text{K}(0.012\%)$ ,  $^{41}\text{K}(6.7\%)$ ) [9].

In the process of irradiating the target other than that used in the operating voltage determination of GM tube, the apparatus used are: a bottle of powder KI, microbalance, plastic rings, sticky tape, scissors, micro gauge, target holder, and the  $^{241}\text{Am}$  Be neutron source. Then, two different target masses of KI are measured. It is also checked that whether an epithermal neutrons are there. This has been done by putting a cadmium-covered sample in the neutron source, and it is shown that there are some few epithermal neutrons.

Then the two measured KI targets

$M_1=0.4810\text{g}$  (labeled front), and

$M_2=0.4819\text{g}$  (labeled back)

together with indium metal (putting indium target at the middle of them) are fixed on the target holder. Finally, positioning them according to their label and registering the time of start of irradiation, the targets are inserted in the neutron source.

### **3.3. Measurement of the activities of KI target**

In order to omit some fluctuations of the electronic components of the already set up apparatus the electrical system was already switched on, and everything was ready for experimental reading. In addition to those being measured from a sample, extraneous radiation like gamma rays emitted by certain radioisotopes in the ground, the air, and various building materials as well as cosmic radiation can all provide counts in the GM tube. This background counting rate should always be subtracted from the sample's counting rate in order to obtain the rate

only from the sample. Therefore readings of the background counts are taken.

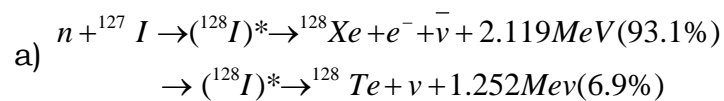
The back KI target, just by registering the time at which the irradiation is stopped, is taken from the neutron source and put on the sample holder at zero distance from the GM tube window. Finally, the counts per 100 second and the corresponding counting time are read and recorded to about 25 hours.

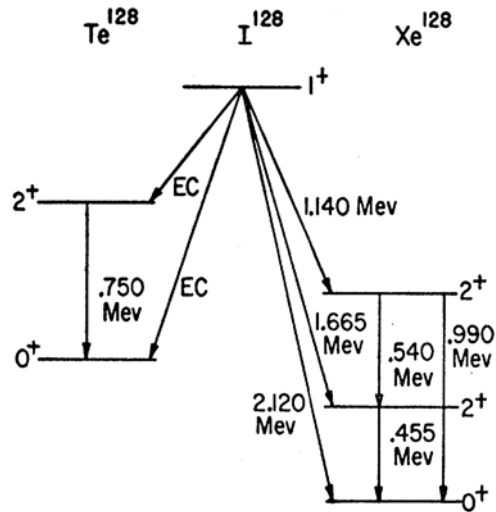
Similarly, after 4 days, measurement of the counts per 100 sec and the corresponding time at each count of the front KI target are read and recorded for about 25 hours.

### 3.4 Data and Data Analysis

As the technique that is used in this particular experiment is neutron activation of KI target, the radioactive nucleus  $^{128}\text{I}$  is produced after a neutron is captured in a  $^{127}\text{I}$  and a radioactive nucleus  $^{42}\text{K}$  is produced after a neutron is captured in  $^{41}\text{K}$ .

The activation of both  $^{128}\text{I}$ , and  $^{42}\text{K}$  which are occurred through  $(n, \gamma)$  reactions are as follows [9] :





**Figure 7 decay scheme of  $^{128}\text{I}$  [9].**

According to the decay scheme of  $^{128}\text{I}$  illustrated above electrons are released in 93.1% of the  $^{128}\text{I}$  decays through three channels to three different energy levels of  $^{128}\text{Xe}$ . The energy and abundance of the three  $\beta^-$  radiations are shown in table 3.

**Table 3 [9]**

<b>Beta</b>	<b>Radiation (Mev)</b>	<b>Branching ratio (%)</b>
$\beta_1$	2.12	76
$\beta_2$	1.665	15.5
$\beta_3$	1.125	2



**Part I (back target)****Table 5 (back)****Ave. Background=60  $\pm$ 7.7 c/s**

<b>S.No</b>	<b>Time (sec)</b>	<b>Background Subtracted Count rate (per sec)</b>	$\pm\sqrt{N}$	<b>S.No</b>	<b>Time (sec)</b>	<b>Background Subtracted Count rate (per sec)</b>	$\pm\sqrt{N}$
1	639	774	27.8	35	7659	41	6.4
2	839	740	27.2	36	7859	69	8.3
3	1039	634	25.1	37	8059	67	8.1
4	1239	605	24.5	38	8259	57	7.5
5	1439	544	23.3	39	8459	67	8.1
6	1639	489	22.1	40	8859	33	5.7
7	1839	463	21.5	41	9059	53	7.2
8	2039	448	21.1	42	9459	55	7.4
9	2239	385	19.6	43	9859	16	4
10	2439	332	18.2	44	10259	38	6.1
11	2639	305	17.4	45	10659	44	6.6
12	2839	302	17.3	46	10859	32	5.6
13	3039	280	16.7	47	11459	46	6.7
14	3239	240	15.4	48	11859	34	5.8
15	3439	252	15.8	49	12059	32	5.6
16	3639	233	15.2	50	12459	35	5.9
17	3839	208	14.4	51	12659	19	4.3
18	4039	170	13.0	52	12859	37	6.0
19	4239	174	13.1	53	12859	34	5.8
20	4439	159	12.6	54	13059	35	5.9
21	4799	139	11.7	55	13259	32	5.6
22	4999	119	10.9	56	13459	45	6.7
23	5199	118	10.8	57	13659	45	6.7
24	5399	112	10.5	58	13859	25	5
25	5599	98	9.8	59	14059	46	6.7
26	5799	109	10.4	60	14459	30	5.4
27	5999	114	10.6	61	14659	25	5
28	6199	98	9.89	62	14859	26	5.0
29	6399	88	9.3	63	15059	30	5.4
30	6699	64	8	64	15259	23	4.7
31	6889	62	7.8	65	15459	19	4.3
32	7059	78	8.8	66	15459	32	5.6
33	7259	60	7.7	67	15659	30	5.4
34	7459	69	8.3	68	15859		

a) Exponential curve 1

## Exponential Curve for Irradiated KI (back) target

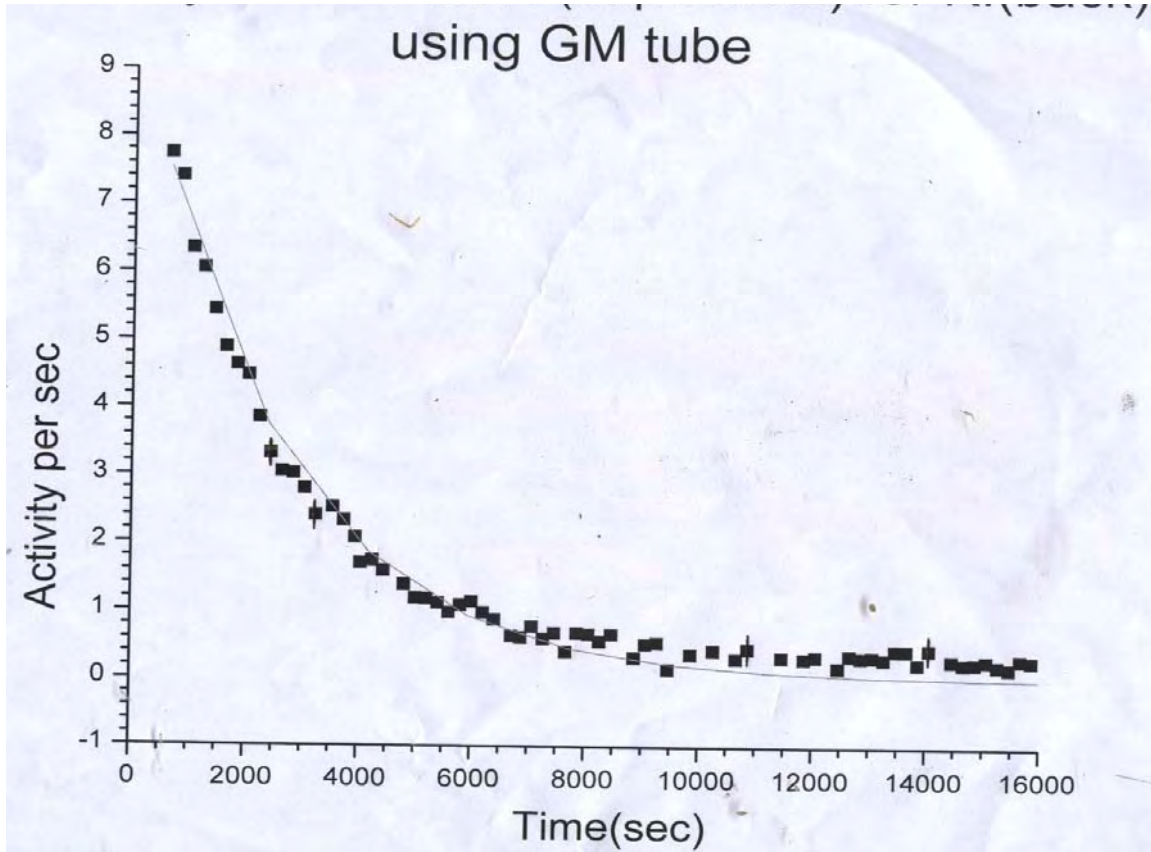


Figure 9 Experimental curve of Activity Vs time for KI (back) target.

From the above experimentally plotted exponential curve, it is seen that the activities of different half-life are present, so the decay curve is a superposition of the exponentials. That is

$$I = I_1^0 e^{-\lambda_1 t} + I_2^0 e^{-\lambda_2 t} \text{-----(16)}$$

The determination of the half-life from the graph shouldn't be made by reading only the time during which the intensity drops from  $I_0$  to  $I_0/2$ . A more accurate result is obtained by choosing the points on the straight decay curve which lie as far from each other as possible [13].

**b) Logarithmic Curve 1**

Using the same data but converting the vertical axis parameter (count rate) to the natural logarithm of the curve is plotted as follows:

**Logarithmic curve for irradiated KI (back) target**

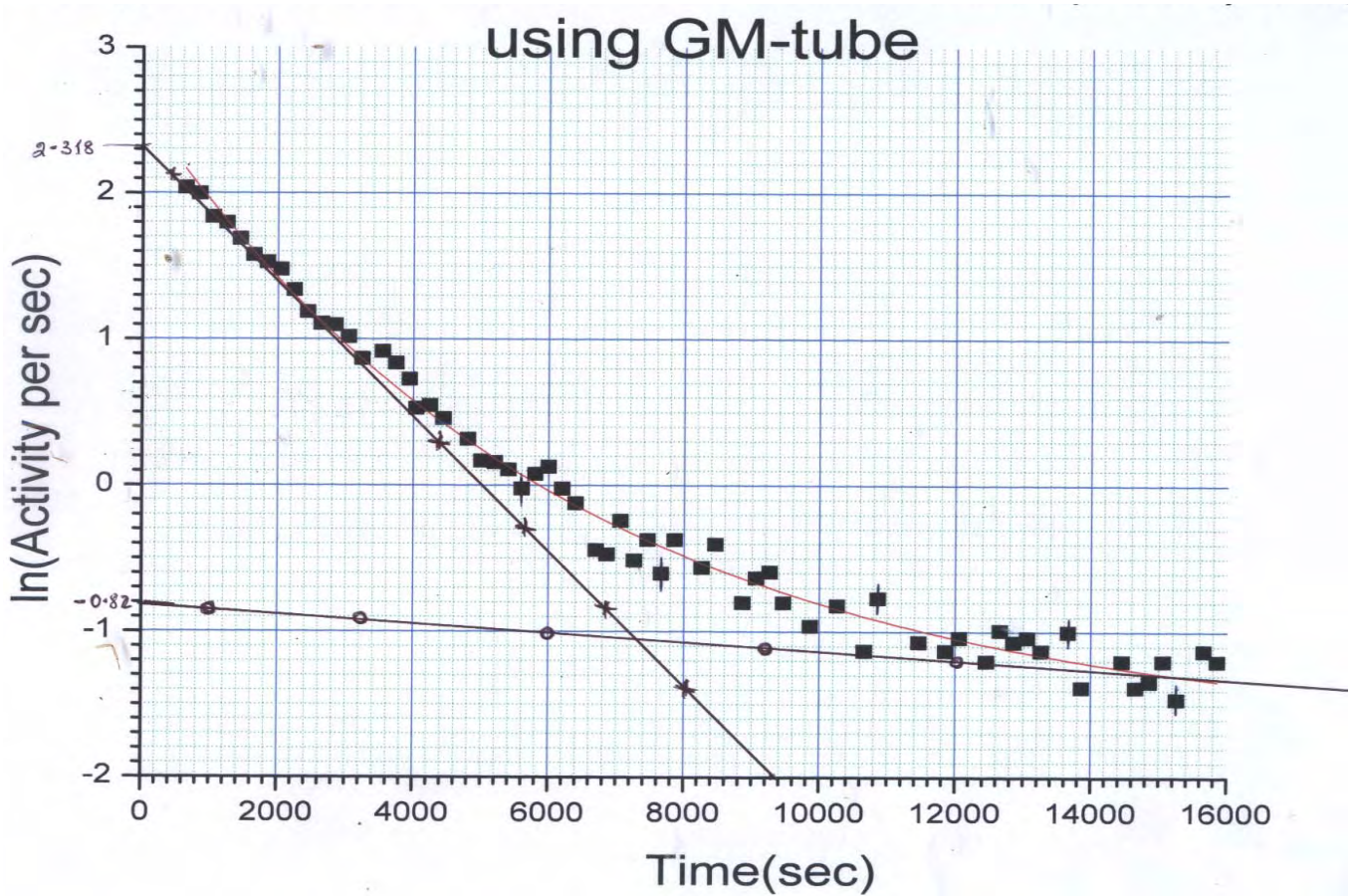


Figure 10 Logarithmic Curve of activity Vs time for KI (back) target.

Part II (front target)

Table 6(front)

Ave. background=68 ±8.2 c/s

S.No	Time (sec)	Background Subtracted Count rate (per sec)	$\pm\sqrt{N}$	S.No	Time (sec)	Background Subtracted Count rate (per sec)	$\pm\sqrt{N}$
1	190	1030	32.0	35	11450	28	5.2
2	390	917	30.2	36	11950	22	4.6
3	590	843	29.0	37	12450	31	5.5
4	790	753	27.4	38	12950	26	5.0
5	990	716	26.7	39	13450	15	3.8
6	1190	669	25.8	40	13950	23	4.7
7	1390	609	24.6	41	14450	24	4.8
8	1790	493	22.2	42	14950	27	5.1
9	1990	430	20.7	43	15450	13	3.6
10	2190	438	20.9	44	15950	18	4.2
11	2390	382	19.5	45	16450	29	5.3
12	2590	352	18.7	46	16950	20	4.4
13	2790	347	18.6	47	17450	12	3.4
14	2990	302	17.3	48	17950	27	5.1
15	3390	258	15.9	49	18450	26	5.0
16	3590	200	14.1	50	18950	30	5.4
17	3790	208	14.4	51	18650	24	4.8
18	4190	193	13.8	52	20350	17	4.1
19	4590	171	13.0	53	21050	24	4.8
20	4990	124	11.1	54	21750	35	5.9
21	5390	130	11.4	55	22450	16	4
22	5670	94	9.6	56	23150	22	4.6
23	6070	96	9.7	57	23850	24	4.8
24	6470	80	8.9	58	24550	12	3.4
25	6870	87	9.3	59	25250	22	4.6
26	7270	83	9.1	60	25950	21	4.5
27	7560	65	8.0	61	26650	21	4.5
28	8050	66	8.1	62	27350	23	4.7
29	8450	53	7.2	63	28050	16	4
30	8950	44	6.6	64	28750	24	4.8
31	9450	38	6.1	65	29450	21	4.5
32	9950	36	6.0	66	30150	20	4.4
33	10450	33	5.7	67	31550	19	4.3
34	10950	46	6.7	68	33250	16	4

a) **Exponential Curve 2**

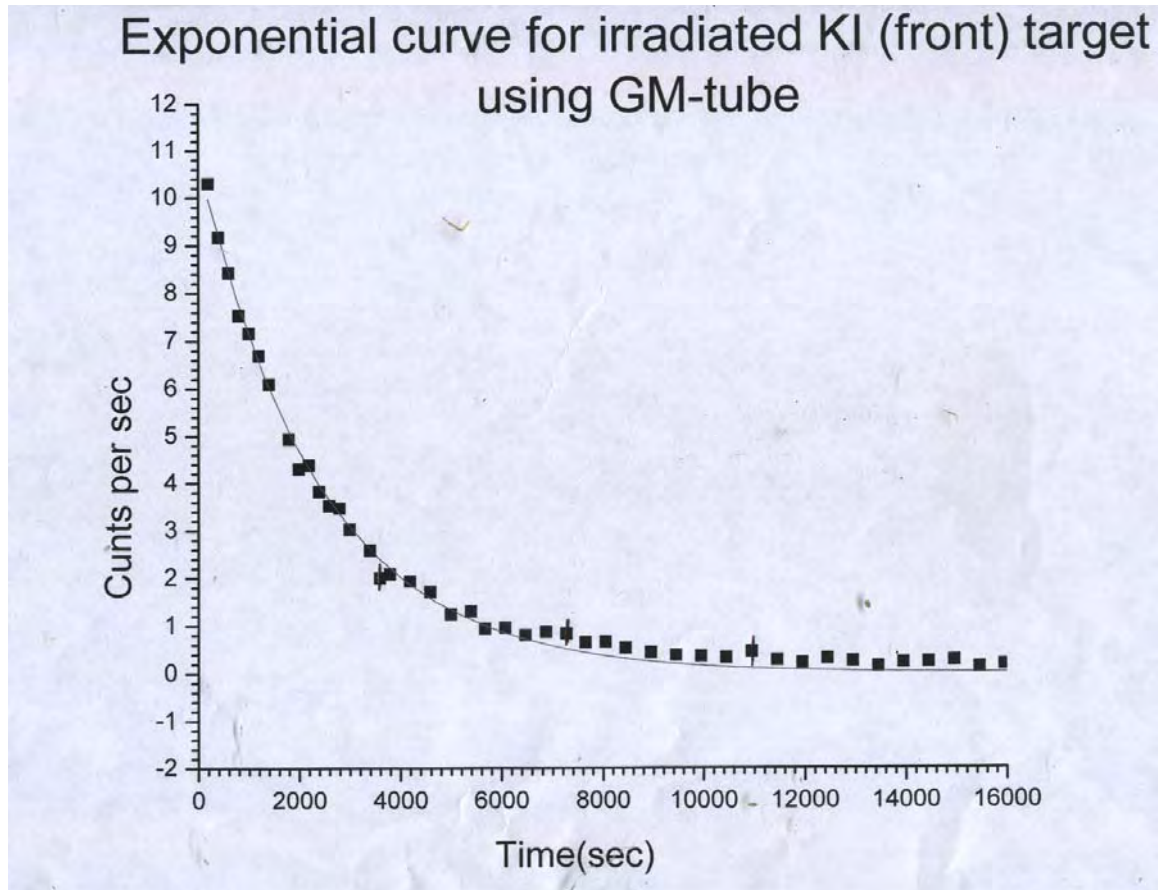


Figure 11. Experimental curve of activity Vs time for KI (front) target.

Similar analysis is made and the results are:

b) Logarithmic Curve 2

Logarithmic curve for irradiated KI (front) target

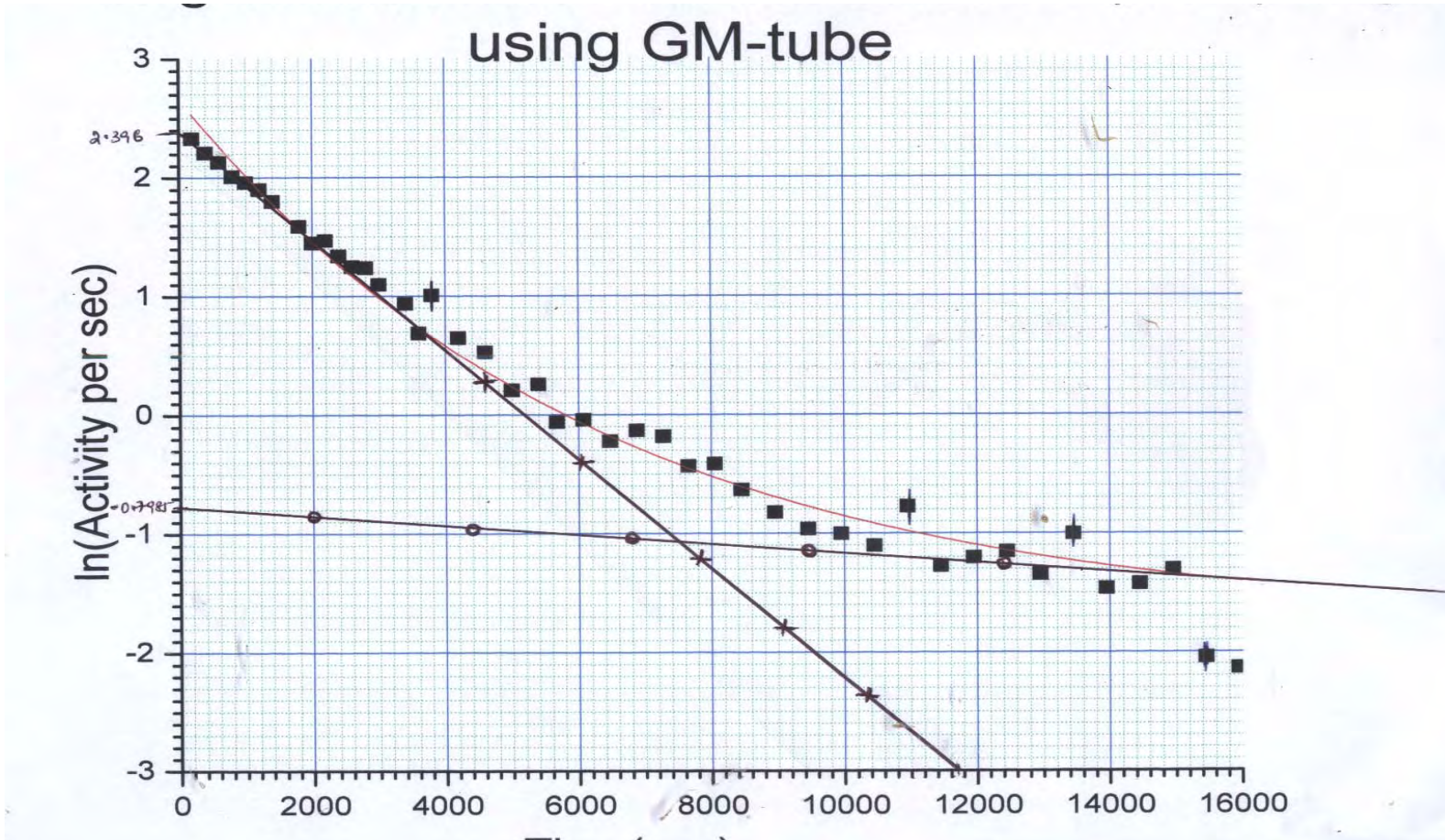


Figure 12 Logarithmic curve of activity vs. time for KI (front) target.

### Decay Constants of Iodine-128 and Potassium-42

The targets consist of a mixture of independently decaying radioisotopes  $^{128}\text{I}$  and  $^{42}\text{K}$ . The presence of one has no effect on the decay rate of the other and the total activities of the targets are the sum of the individual activities. Thus, the total activity is of the mixture of the two.

For a simple source consisting of a single species of nuclide, the logarithm of the activity plotted against time results in a straight-line plot.

But the activated target KI consists of two radioisotopes  $^{128}\text{I}$  and  $^{42}\text{K}$ , each decaying independently and this is shown in figures 10 and 12.

From the figures, the solid curve represents the sum of the two activities whereas the (X) marked line represents the short half- life of nuclide  $^{128}\text{I}$ , and the (o) marked straight line represent the long half life nuclide  $^{42}\text{K}$ .

When the slope (or decay constant) of the (X) marked line is calculated it is found to be  $4.620 \times 10^{-4} \text{s}^{-1}$  and the slope (or the decay constant) of the (o) marked line is calculated and it is found to be  $1.558 \times 10^{-5} \text{s}^{-1}$ .

Therefore, the experimentally determined decay constants of the two nuclides Iodine-128 and potassium-42 are:

#### **Using front target semi log curve**

$$\lambda \text{ for } ^{128}\text{I} = \frac{5.396}{11680} = 4.620 \times 10^{-4} \text{s}^{-1}$$

$$\lambda \text{ for } ^{42}\text{K} = \frac{2.201}{141300} = 1.558 \times 10^{-5} \text{s}^{-1}$$

#### **Using back target semi log curve**

$$\lambda \text{ for } ^{128}\text{I} = \frac{4.32}{9345} = 4.620 \times 10^{-4} \text{s}^{-1}$$

$$\lambda \text{ for } ^{42}\text{K} = \frac{1.18}{75740} = 1.558 \times 10^{-5} \text{s}^{-1}$$

### 3.5 The Determination of thermal neutron flux of the <sup>241</sup>Am- Be neutron Source.

#### Efficiency of the GM-tube

Though the Geiger counter can detect all the three types of radiation, it is most efficient for beta particles.

Since beta particles have a continuous energy distribution, ranging from no energy up to their maximum energy, there is a continuum in the distance they can penetrate without being absorbed. An empirical approximation for the intensity of beta particles is given [3].

$$\frac{I}{I_0} = e^{-\mu d} \text{-----(17)}$$

Where I= is the intensity after penetrating the absorber

I<sub>0</sub>= is the intensity with no absorber

d= the thickness of the absorber in g/cm<sup>2</sup>

μ = the mass absorption coefficient in cm<sup>2</sup>/g

So the geometry dependent efficiency of the GM-tube is:

$$\varepsilon = \frac{I}{I_0} = e^{-\mu d} \text{-----(18)}$$

But specific to this experiment it can be modified as:

$$\varepsilon = \frac{1}{100} (ae^{-\mu_1 d} + be^{-\mu_2 d} + ce^{-\mu_3 d}) \text{-----(19)}$$

Where,

a, b, and c are the relative abundance of the beta in that specific energy.

μ is also can be given by an empirical relation [15]

$$\mu = 17E^{-1.14} \text{-----(20)}$$

Where,

E is given in MeV

$\mu$  is in  $\text{cm}^2/\text{g}$

Finally, the efficiency ( $\varepsilon$ ) of the GM-tube for the three Beta particles of  $^{128}\text{I}$  is calculated

### Using the back KI target

**Given:** for  $\beta_1, E_1 = 2.12\text{MeV}, a = 76$

For  $\beta_2, E_2 = 1.665\text{MeV}, b = 15.5$

For  $\beta_3, E_3 = 1.125\text{MeV}, C = 2$

$$\mu_1 = 17(E_1)^{-1.14} = 17(2.12)^{-1.14} = 7.218\text{Cm}^2/\text{g}$$

$$\mu_2 = 17(E_2)^{-1.14} = 17(1.665)^{-1.14} = 9.506\text{Cm}^2/\text{g}$$

$$\mu_3 = 17(E_3)^{-1.14} = 17(1.125)^{-1.14} = 14.863\text{Cm}^2/\text{g}$$

$d = d_1 + d_2 + d_3$ , where,

$d_1$  = window thickness ( $2.5\text{mg}/\text{cm}^2$ )

$d_2$  = target thickness ( $0.3068\text{g}/\text{cm}^2$ )

$d_3$  = tape thickness ( $2.5\text{mg}/\text{cm}^2$ )

This implies,  $d = 0.3118\text{g}/\text{cm}^2$

Therefore,

$$\begin{aligned}\varepsilon &= \frac{1}{93.5}(8.006 + 0.800 + 0.019) \\ &= \underline{\underline{0.094}}\end{aligned}$$

The value of the thermal neutron flux of the  $^{241}\text{Am}$ -Be neutron source of Department of Physics, AAU (Science Faculty) is determined experimentally using equation 14.

$$\phi = \frac{\lambda_b M n e^{\lambda_b t_d}}{\varepsilon \cdot P \sigma \cdot m_b \cdot N_A \cdot (1 - e^{-\lambda_b t_i}) \cdot (1 - e^{-\lambda_b t_c})}$$

Where,  $\lambda_b = 4.620 \times 10^{-4} \text{s}^{-1}$  (figure 9)

$$M=127 \text{ g/mole}$$

$$n_b=10.16 \text{ (at 0 time)}$$

$$t_d=0 \text{ (figure 10)}$$

$$\varepsilon = 0.094$$

$$P=100\% [16]$$

$$M_b = \frac{127}{166} \times 0.4819 \text{ g} = 0.3687 \text{ gm}$$

$$N_A = 6.02 \times 10^{23} \text{ atoms/mole}$$

$$t_c = 100 \text{ sec}$$

$$\sigma = 6.2 \times 10^{-28} \text{ m}^2 [16]$$

$$t_i = 432546 \text{ seconds}$$

$$\phi_{back} = \frac{4.62 \times 10^{-4} \text{ S}^{-1} \times 127 \text{ g/mole} \times 10.16}{0.094 \times 100 \times 6.2 \times 10^{-28} \text{ m}^2 \times 0.3687 \text{ gm} \times 6.02 \times 10^{23} \frac{\text{atom}}{\text{mole}} (1 - e^{-\lambda_b t_i}) (1 - e^{-\lambda_b t_c})}$$

$$\phi_{back} = \frac{4.62 \times 10^{-4} \times 127 \times 10.16}{0.094 \times 100 \times 6.2 \times 10^{-28} \times 0.3687 \times 6.02 \times 10^{23} \times (1 - e^{-4.62 \times 10^{-4} \times 432546}) (1 - e^{-4.62 \times 10^{-4} \times 100})}$$

$$\phi_{back} = \frac{0.59612784}{5.840015629 \times 10^{-5}} \text{ neutron/m}^2 \text{ s}$$

$$\phi_{back} = (1.0207 \times 10^4 \pm 0.10207 \times 10^4) \text{ neutrons/m}^2 \text{ s}$$

### **From the Front KI target**

Every parameter except the following is the same

$$n_f = 10.98$$

$$t_i = 864153 \text{ seconds}$$

$$m_f = \frac{127}{166} \times 0.4810 \text{ g} = 0.3679 \text{ gm}$$

The flux,  $\phi_{front} = \frac{\lambda_f M_{nf} e^{\lambda_f t_d}}{\varepsilon.P.\sigma.m.N_A (1 - e^{-\lambda_f t_i}) (1 - e^{-\lambda_f t_e})}$

$$\Rightarrow \phi_{front} = \frac{4.620 \times 10^{-4} \times 127 \times 10.98}{0.094 \times 100 \times 6.2 \times 10^{-28} \times 0.3679 \times 6.02 \times 10^{23} \left(1 - e^{-4.62 \times 10^{-4} \times 864153}\right) \left(1 - e^{-4.62 \times 10^{-4} \times 100}\right)}$$

$$\Rightarrow \phi_{front} = \frac{0.6442052}{5.829085796 \times 10^{-5}}$$

$$\Rightarrow \phi_{front} = 1.1051.5649 \text{ neutron / m}^2 \text{ s}$$

$$\Rightarrow \phi_{front} = \underline{\underline{(1.1052 \times 10^4 \pm 0.11052 \times 10^4) \text{ neutron / m}^2 \text{ s}}}$$

Therefore, the average flux is:

$$\phi_{ave} = \frac{\phi_{front} + \phi_{back}}{2}$$

$$\phi_{ave} = \frac{(1.1052 \times 10^4 + 1.0207 \times 10^4) \text{ neutrons / m}^2 \text{ s}}{2}$$

$$\phi_{ave} = \underline{\underline{(1.0629 \times 10^4 \pm 0.10629 \times 10^4) \text{ neutrons / m}^2 \text{ s}}}$$

### 3.6 Determination of the thermal neutron capture cross-section for <sup>41</sup>K.

The efficiency of the GM-tube for the two beta particles from <sup>42</sup>K is determined by the same relationship used for <sup>128</sup>I.

Given: - for  $\beta_1, E_1 = 3.55 \text{ MeV}, a = 82$

For  $\beta_2, E_2 = 1.99 \text{ MeV}, b = 18$

$$\mu_1 = 17(E_1)^{-1.14} = 17(3.55)^{-1.44} = 4.010 \text{ cm}^2 / \text{g}$$

$$\mu_2 = 17(E_2)^{-1.14} = 17(1.99)^{-1.44} = 7.758 \text{ cm}^2 / \text{g}$$

$d$  is the same ( $0.3118 \text{ g / cm}^2$ )

Then

$$\begin{aligned}\varepsilon &= \frac{1}{100} (82e^{-\mu \cdot d} + 18e^{-\mu^2 d}) \\ &= \frac{1}{100} (82e^{-4.01 \times 0.31179} + 18e^{-7.758 \times 0.311797}) \\ &= \underline{\underline{0.025}}\end{aligned}$$

Then the calculation of the thermal neutron capture cross-section for  $^{41}\text{K}$  is done using an equation used for the determination of the thermal neutron flux.

### **From the back KI target**

Given

$$\lambda_k = 1.558 \times 10^{-5} \text{ s}^{-1} \text{ (figure 9)}$$

$$M = 41 \text{ g / mole}$$

$$n_b = 0.44 \text{ (figure 9)}$$

$$t_d = 0$$

$$\varepsilon = 0.25$$

$$P = 6.7\% [16]$$

since natural potassium has [93.3% of  $\text{K}^{39}$ ,

0.012% of  $\text{K}^{40}$ , and

6.7% of  $\text{K}^{41}$ ]

$$m_k = \frac{41}{166} \times 0.4819 \text{ g} = 0.1190 \text{ g}$$

$$N_A = 6.02 \times 10^{23} \text{ atoms / mole}$$

$$t_c = 100 \text{ sec}$$

$$t_i = 432546$$

Given: -  $\phi_b = 1.0207 \times 10^4 \text{ neutrons / m}^2 \text{ s}$

Then the cross-section for  $^{41}\text{K}$  from the back KI target is:

$$\begin{aligned}\sigma_{back} &= \frac{\lambda_k Mn_k e^{\lambda_k t_d}}{\varepsilon_K P \Phi_b m_k N_A (1 - e^{-\lambda_k t_i})(1 - e^{-\lambda_k t_i})} \\ \sigma_{back} &= \frac{1.558 \times 10^{-5} \text{S}^{-1} \times 41 \times 0.44}{0.25 \times 6.7 \times 1.0207 \times 10^4 \times 0.1190 \text{g} \times 6.02 \times 10^{23} (1 - e^{-1.588 \times 10^{-5} \times 432546}) \times (1 - e^{-1.558 \times 10^{-5} \times 100})} \\ \sigma_{back} &= \frac{2.810632 \times 10^{-4}}{1.906714008 \times 10^{24}} \\ \sigma_{back} &= 1.474071092 \times 10^{-28} \\ \sigma_{back} &= \underline{\underline{(1.474 \pm 0.1474) \text{ barns}}}\end{aligned}$$

Similarly from the front KI target it is determined the parameters in which there values changed one:

$$\begin{aligned}n_f &= 0.45 (\text{figure 12}) \\ \phi_f &= 1.1052 \times 10^4 \text{ neutron/m}^2 \text{s} \\ m_f &= 0.1188 \text{g} \\ t_i &= 864153\end{aligned}$$

Then the cross section for  $^{41}\text{K}$  from the front KI target is:

$$\begin{aligned}\sigma_{front} &= \frac{\lambda_f Mn_f e^{\lambda_k t_d}}{\varepsilon P \phi_f m N_A (1 - e^{-\lambda_k t_c})(1 - e^{-\lambda_k t_c})} \\ \sigma_{front} &= \frac{1.558 \times 10^{-5} \times 41 \times 0.45}{0.25 \times 6.7 \times 1.1052 \times 10^4 \times 0.1188 \times 6.02 \times 10^{23} (1 - e^{-1.558 \times 10^{-5} \times 864153}) (1 - e^{-1.558 \times 10^{-5} \times 100})} \\ \sigma_{front} &= \frac{2.87451 \times 10^{-4}}{2.061093995 \times 10^{24}} \\ \sigma_{front} &= 1.394652552 \times 10^{-28} \\ \sigma_{front} &= \underline{\underline{(1.395 \pm 0.1395) \text{ barns}}}\end{aligned}$$

$$\sigma_{ave} = \frac{\sigma_{front} + \sigma_{back}}{2} = \frac{1.474barn + 1.395barn}{2}$$

$$\sigma_{ave} = (1.43 \pm 0.143)barns$$

## 4. Results and Discussion

### 4.1 Error Analysis

In the measurements of the needed parameters and collection of data's experimental errors are inevitable.

In the determination of the thermal neutron capture cross-section of  $^{41}\text{K}$  the previously determined value is 1.46 barns [16] while the experimentally determined value is 1.43 barns. The error calculated, therefore, is:

$$\text{Error (\%)} = \frac{\text{previously determined value} - \text{experimental value}}{\text{previously determined value}} \times 100\%$$

$$= \frac{1.46 - 1.43}{1.46} \times 100\%$$

$$= 2.05\%$$

The possible systematic and random sources of errors for this are:

- The presence of epithermal neutrons, which is already observed by irradiating, samples one without cadmium cover and the other with cadmium cover.
- Instrumental errors due to GM-tube and the digital counter since they have been using for a long time, and the microbalance used to measure the mass of the target.

- Personal errors in measuring
  - i) The masses of the KI targets
  - ii) The time from the start of irradiation to the counting rate
  - iii) The diameter of the ring in which the KI target is filled.
  - iv) Thickness of the sticking tape.
- Errors due to the rounding off decimals numbers.
  
- Unpredictable fluctuations of line voltage, temperature, and mechanical vibrations of the equipments used.
  
- Due to the probability of incident gamma rays interaction with the GM-tube wall, and production of secondary charge electrons that probably reach the fill gas in the tube, there could be < 1% error. The total error due to the above factors could be estimated around < 10%.

## **4.2 Discussion on Results**

Though up-to-date and more efficient detecting equipments such as HPGe and MCA are not used the decay constants, of the two nuclides I-128 and K-42; which are found experimentally using the end window GM tube are beyond expectation, because the GM-tube's age and other factors which could be observed in the experiment.

Of course to get this result very careful manual work has been done, since the activities measured in both targets are mixture of the independent activities of the nuclides. Therefore, the experimentally determined decay constants from fig. 10 and figure 12 are almost equal to theoretically given values found [16].

The average thermal neutron flux of the  $^{241}\text{Am}$ -Be neutron used in the experiment is determined to be  $1.0629 \times 10^4$  neutron /m<sup>2</sup> sec. Though

there are no previous experimental works done on it, from its activity (2Ci) given in reference [12] and the neutron yield of the  $^{241}\text{Am}$ -Be source approximated in literature [11] the result can be taken as an acceptable one.

The thermal neutron capture of  $^{41}\text{K}$  is determined to be 1.43 barns. This result, when compared with the previously determined value 1.46 barns [16], is in a good agreement. In addition to this in the calculation of the cross section of the  $^{41}\text{K}$  since almost many of the parameters in the formula are constant, and its calculated value is also very nearer to the theoretical value then the result of the flux can indirectly be confirmed as an acceptable one.

## **5. Conclusion**

Through the activation of the chemical compound KI and measurement of Beta particles of the activated KI target; by measuring the subsequent decay rate of the radioactive nuclides ( $^{128}\text{I}$  and  $^{42}\text{K}$ ), the thermal neutron flux of the  $^{241}\text{Am}$ -Be neutron source which is available in Physics Department AAU (Science Faculty) is determined successfully. In addition to this the thermal neutron capture cross-section of the  $^{41}\text{K}$  is determined, the result of which is in a good agreement with the theoretical value, 1.46 barns [16]. Again, these results show that the neutron activation method, other than its wide application in the qualitative and quantitative elemental analysis, it can also be used in the determination of the neutron flux of a neutron source and the cross section of an irradiated target nucleus.

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## **DECLARATION**

I hereby declare that this thesis is my original work and has not been presented for degree in any other university. All sources of materials used for the thesis have been duly acknowledged.

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