THERMAL NEUTRON INDUCED REACTION IN AS-75

By

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The undersigned hereby certify that they have read and recommend to the School of Graduate Studies for acceptance a thesis entitled "**Thermal neutron induced reaction in As-75**" by **Tamene Hailu Melkegna** in partial fulfillment of the requirements for the degree of **Master of Science in Physics**.

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Abstract

The build-up and decay of radioactivity in arsenic caused by reaction with slow neutrons will be studied. In particular, we will be able to measure the radioisotope capture cross section of the sample, using a GM-counter on the decay of radioactivity data taken. We will also a measurement of the half life of the sample and we will show that the activity level reaches an asymptotic value using slow neutron activation analysis technique.

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Introduction

Discovery of Neutron

German physicist Bothe and Becker continuing Rutherford's experiment on alpha particle bombardment of light atoms and observation of nuclear reaction discovered that when some elements such as ,lithium ,beryllium, or boron are exposed to alpha radiation there appears a strongly penetrating radiation which pass through lead easier than the hardest gamma ray emitted by natural radioactivity substance[1]. The new radiation interested French physicist Irene and Frederick Joliot-Curie who discovered its ability of knocking out protons from paraffin and other substance containing hydrogen. The recoil proton has high kinetic energy and penetrating power .If the recoil nuclei are considered to appear under the influence of gamma ray, then, the gamma ray energy must be so high that it is impossible to fit it to energy balance of the nuclear reaction activated by alpha particle in the light nuclei. Gamma ray assumption were entirely inconsistent with the results from experiment on the absorption of these rays in the lead The nature of this radiation was discovered by Chadwick (2).He performed several experiments on the recoil of hydrogen and nitrogen nuclei that were struck by the rays coming from beryllium bombarded by alpha particles and he suggested that, if these rays are taken to be gamma rays then the result of experiments led to values for the energies of these rays that depended on the nature

of recoil nuclei. But his experimental result show that ,if the recoil atoms are to be attributed to the collision with gamma rays ,the amount of energy that has to be assigned to the gamma ray will increase with the increase the mass of recoil atom this is contrary to the principle of conservation of energy and momentum during collision .

Chadwick supposed that, these rays are not gamma rays but consist of particles of mass very nearly the mass of proton but having no charge. These particles are called neutrons (n). He was able to produce such particles (neutrons) in a free state by bombarding aluminum atomic nuclei with alpha-particles[1].

$$^{27}Al +^{4} He \rightarrow (^{31}P^{*}) \rightarrow^{30} p +^{1} n + Q \tag{0.0.1}$$

In addition, having measured the energy of recoil nuclei, the mass these particles (neutrons) approximately, m=1.008665amu.The analysis the properties of the new particles led to the conclusion that, neutron, like proton has a spin of half ($\frac{1}{2}$) but a magnetic moment of $1.9135\mu_N$[1]. This indicates that neutron is a composite particle. It is in fact composed of 2d quarks and 1u quark. It was for this work that Chadwick was awarded the Nobel Prize for physics in 1935.

Radioactive Decay of Neutron

The first determination of the mass of the neutron by Chadwick and Goldhaber shows that, its mass was greater than that of proton. Depending on this fact they suggest that free neutrons should be unstable and they should decay into a the proton, electron and antineutrino.

$$n \rightarrow p + e + \nu \tag{0.0.2}$$

From the radioactive decay of neutron there is no physical law violated .In 1948, A.H Snell and his workers were able to detect protons and electrons coming out

from a thermal neutron source with an estimated half life abut 10 to 30 minute. J.M.Robinson [1] was able to determine accurate half-life of neutron and its beta-ray (electron) spectrum using intense thermal neutron source. The end point energy spectrum of the experiment was 784kev[1],which was in excellent agreement with the mass difference of 784kev between a neutron and hydrogen atom. The value of half-life of neutron was determined from the value of the density of neutron beam and the number of neutrons decaying per unit time per volume. The accepted value of the mean lifetime of free neutron is 10.26 ± 0.04 minute; while neutrons bound in a nucleus apparently are stable[2].

Chapter 1

Neutron Sources

Neutron source is a general term referring to the Variety of device that emit neutrons. These devices are diverse in size, complexity and applications in areas of physics, engineering, medicine and research. Depending up on the mechanisms used to produce neutrons; neutron sources are divided in to two broad categories:

1, Radioactive sources

2, Accelerator based sources

1.1 Radioactive Sources

1.1.1 Radioisotope(α, n)Sources

These neutron sources uses (α, n) reaction in order to produce neutron, those light nuclei like; Li, Be, B, N, O, F, are used as a targets and radio nuclide like; Pu, Ra, Am, Cm, Th, and U are used for sources[3].

1,*(α, n) Reactions*

One of the most known alpha particle nuclear reaction is reaction of α-particle with light nuclei and which results in compound nucleus and it ejects a neutron when

returning to the ground state.

2,*(α, n) Targets*

A number of light nuclei are good to undergoing(α, n) reaction relatively with low energy $\alpha - particle$. This reaction has higher probability in light nuclei than heavy nuclei, because the threshold energy or coulomb barrier is smaller in light nuclei. The nuclei are then taken to as targets in table(1.1) along with the reaction energy (Q-value), threshold energy and neutron energy for 5.5Mev[3] .

Target	Q (Mev)	Threshold energy (Mev)	Mean En for E_α =5.5Mev	Yield ($n/10^6\alpha$)
^{6}Li	- 3.975	6.620	NA	NA
^{7}Li	-2.79	4.382	0.5883	3.156
^{10}Be	5.702	exothermic	5.005	80.073
^{11}B	1.06	"	2.243	5.72
^{13}B	0.157	"	2.993	23.724
^{13}C	2.215	"	4.72	9.904
^{17}O	0.587	"	2.523	0.152
^{18}O	-0.697	0.852	2.374	0.333
^{19}F	-1.95	2.361	1.304	0.106

Figure 1.1: Traits of common(α, n)source[3],[24]

As shown from fig (1.1), the greatest yield achieved by a beryllium target for (α, n) reaction. Therefore, it has usually been used as a target material for a neutron source. The reaction is:

$$^{9}_{4}Be(\alpha, n)^{12}_{6}C \tag{1.1.1}$$

Even though, those nuclei in the table undergo (α, n) reaction, only those isotopes which can undergo exothermic or low energy threshold are considered to be useful

Source	$T_{½}$	Most common E_{α} (Mev)	Yield (n /10^5 α)
^{239}Pu-Be	24110yr	5.156	57.2
^{210}Po-Be	138.38d	5.304	63.7
^{241}Am-Be	432.2yr	5.486	71.5
^{238}Pu-Be	87.7yr	5.499	72.1
^{238}Cm-Be	18.1yr	5.805	105.3
^{244}Cm-Be	162.8d	6.113	88.1
^{226}Ra-Be	1600yr	4.784,5.490,6.002 7.687, 5.304	736.8
^{227}Ac-Be	21.8yr	6.038,5.726, 6.819 7.386, 6.623	707.2
^{228}Ra-Be	5.75yr	5.4223,5.686,6.288 6.779,6.051(36%) 8.784	707
^{228}Th-Be	1.911yr	64% ^{228}Ra	707.2
^{232}U-Be	68.9yr	5.320 +^{228}Ra	755.5

Figure 1.2: (α, n)source properties[3],[24]

neutron source[3].

3,(α, n) ***Radionuclide***

Alpha emitting radionuclide are also very important components of (α, n) reaction neutron sources. Uranium and transuranic elements are usually alpha emitters since the probability of (α, n)reaction increases with alpha -particle energy and the number of alpha-particle depends on half-life; those alpha -emitters having high energy and

long half life on the order of 1 up to 2500 years are preferable.

Radioactive neutron sources mentioned in the fig (1.2) can be taken as smaller device neutron sources .^{226}Ra-Be neutron source has long half-life and high neutron yield, therefore it can be considered as a good neutron source.

1.1.2 Spontaneous Fission Source

Certain transuranic heavy nuclide undergo spontaneous fission at a rate sufficient to give a useful neutron source. Each such fission may produce several neutron as well as beta and gamma rays. The most commonly used spontaneous fission neutron source is Californium-252.It undergos an alpha decay with 97percent and half-life 2.65 year. For every 32 alpha emissions, there is a spontaneous fission with an average 3.8 neutrons and half-life of 85.5 year. The neutron yield is 2.3 10^6n / s microgram. Hence, a small-encapsulated source provide significant yield. The peak of relative number of neutrons lies between 0.5Mev and 1Mev with the higher energy neutrons being 8Mev to 10Mev[6].

1.1.3 Nuclear Reactors Sources

Nuclear fission which takes place within the reactors produce very large quantities of neutrons and can be used for a Variety of purposes including power generation and experiments. A nuclear reactor is a source of products of fission process, such as, energy, neutron and some useful radioactive isotopes. The fission reaction which takes place in the reactors using slow or thermal neutrons having energy 0.0253ev and velocity about 2200m/s[2].These neutrons react with natural uranium producing several prompt neutrons. The average number of neutrons produced per fission of

uranium-235 by a thermal neutron is:υ=2.47 $\pm$0.03 neutron/fission[1]. The energy of neutrons from thermal fission of ^{235}U is different value extending from about 0.04Mev to about 17Mev with a distribution maximum average value about 0.75Mev. In order to use these neutrons having different energy value for fission process in the reactor and for experiments they have to be slow down in the reactor by using a substance moderator. The energy released and the number of neutrons per fission of uranium-235 is given by;

$$ {}^{1}_{0}n + {}^{235}_{92}U \rightarrow ({}^{236}_{92}U^{*}) \rightarrow {}^{141}_{56}Ba + {}^{92}_{36}Kr + 3n + Q \tag{1.1.2} $$

Where Q represent the energy released in the reaction which is about 175Mev.Such types of nuclear reactors may produce a large number of thermal neutron flux at about $5.5 \times 10^{6} n/m^{2}.s$ and ,that of fast neutron flux at about $7.1 \times 10^{16} n/m^{2}.s$ [2]

1.1.4 Photo-neutron Source

In some nuclei, gamma radiation from radioactive isotopes with an energy exceeding the neutron binding energy of a nucleus can eject a neutron.The minimum energy needed by the gamma radiation to eject a neutron from a given nuclei is called threshold energy. Beryllium and Deuteron have threshold energy about 1.666Mev and 2.225Mev [1] respectively, which have smaller threshold value. Those radioisotopes, which decay with gamma ray greater than the threshold of these nuclei, co-located with Beryllium and Deuteron, used as neutron(α, n) source. The reaction in photo neutron source is given by ;

$$ \gamma + {}^{9}_{4}Be \rightarrow {}^{8}_{4}Be + {}^{1}_{0}n + Q \tag{1.1.3} $$

$$ \gamma + {}^{2}_{1}H \rightarrow {}^{1}_{1}H + {}^{1}_{0}n + Q \tag{1.1.4} $$

ϒ Source	$T_{1/2}$	E_γ(Mev)	Target	E_n	Yield(n/10^{10}Bq)
^{24}Na	15hr	2.7541 2.7541	Be D	967 263	340000 330000
^{28}Al	2.24m	1.7787	Be	101	32600
^{38}Cl	37.3m	2.1676	Be	446	43100
^{56}Mn	2.58hr	1.8107 2.1131 2.9598 2.9598	Be Be Be D	129 98 1149 365	91500 91500 91500 162
^{72}Ga	14.1m	1.8611 2.2016 2.5077 2.5077	Be Be Be D	174 476 748 140	64900 64900 64900 25100
^{76}As	26.3hr	1.7877 2.0963	Be Be	109 383	3050 3050
^{88}Y	107d	1.8361 2.734 2.734	Be Be D	152 049 253	229000 22900 160
^{116m}In	54.1m	2.1121	Be	397	15600
^{124}Sb	60.2d	1.691	Be	23	210000
^{140}La	40.3hr	2.5217 2.5217	Be D	760 147	10200 6600
^{44}Pr	17.3m	2.1856	Be	462	690

Figure 1.3: (γ, n)source properties[24]

In table(1.3), some of the photo neutron sources with their properties listed [3].Target materials which have large threshold energy needs to have gamma-radiation of several Mev. But these radiations are dangerous which requires larger radiation shielding.

1.2 Accelerator Source

Accelerator neutron sources are neutron sources /devices/ which used proton, tritium or deuteron as a projectile to hit a target material in order to emit neutrons. Theses systems vary in size and diversity, and they include large installations such as the Spallation neutron sources. Those particle accelerators have three basic parts in general. It consists of a source to generate positively charged ions, one or more structures to accelerate the ions in kev up to Mev and a target material. Usually these accelerators used proton and deuteron as a projectile. The energy and intensity of projectiles can be controlled by the system in order to get the desired amount and energy of neutron. Those materials made from light atoms, such as, deuterium, tritium, lithium, and sodium, are taken as a target material. These devices are relatively effective in producing, approximately, mono energetic neutrons with a variety of intensity. Due to this reason, they have many industrial application.

1.2.1 Neutron Producing Reaction

No neutron source is perfectly mono energetic but certain reactions can be used to produce a neutron energy spectrum with a reasonably small energy spread. Because, the energy of neutrons depends on certain factors, such as, energy spread of accelerated particles, the reaction cross-section, exited level of the residual nucleus, etc.

Reaction	Q Value	Best Neutron Energy
$^{7}Li(p,n)^{7}Be$	-1.64Mev	0.1--------0.4Mev
$^{3}H(p,n)^{3}He$	-0.76Mev	0.1--------3Mev
$^{2}H(^{2}H,n)^{3}He$	3.27Mev	4----------10Mev
$^{3}H(^{2}H,n)^{4}He$	17.59Mev	13---------16Mev

Figure 1.4: neutron producing reactions with its Best neutron Energy[24]

Unfortunately, there is no single reaction suitable for producing neutrons across the entire energy range. table(1.4) list several reactions that could be used to make neutrons and energy range in which it works the best. Notice that all these reactions involve either beams of protons or deuterons with small energy range[4].

Apart from their usage interims of the energy of neutrons produced, these particle accelerators are very important for experiments and industrial applications due to their size, diversity, durability and their consistency in number of neutrons produced per second. Depending up on the energy of particles accelerated, target nucleus and the number of neutrons produced; particle accelerators can be divided in to two broad categories.

1.2.2 Light -Ion Accelerators

Compact light-ion accelerators with hydrogen (H), deuterium (D), or tritium (T) ion sources may be used to produce using targets of deuterium, tritium, lithium, beryllium and other low-z metals. Typically these accelerators operate with a voltage in the range of 1Mev and above. The number of neutrons produced is smaller as compared to the larger ones.

1.2.3 High Energy Particle Accelerator

A spallation source is one of the most high-flux neutron source .In the neutron spallation source , a high energy proton beam is used in order to bombard the heavy metal(like mercury) target to yield neutrons and and sometimes used to generate heat. Spallation sources may generate an average neutron flux at about $5\times\ 10^{10}\mathrm{n}/(cm^2.s)$ and above[24].

—-

Chapter 2

Nuclear Reaction

The process of the production of new nuclei and particles or beta and gamma ray in collision of particles/ray and a target nucleus is called nuclear reaction. If upon collision of particles and nuclei remain the same and no new particles and nuclei are produced; the process is called scattering. Nuclear reaction is the most important nuclear process to understand the type and properties of nuclei and the production of radioactive isotopes, such as, Pu, Th, etc. There are different types of nuclear reactions.The type of reaction process mainly depends on, the type of particles, target nucleus, the energy of projectile (particles) and cross-section of the given reaction. The latter two factors are the most dominant in analyzing a given reaction.

2.1 Reaction Cross-Section

The cross- section is the measure of the probability for a particular reaction to occur. This quantity has the dimension of area ,1barn = 10^{-24} cm^2. The cross-section of the interaction of different particles /nucleons/ and target nuclei depends on variables,

such as, the type of projectiles, the energy level width and the effective average potential of the target nucleus etc. It can also be calculated from different model of the nucleus by applying the rule of quantum mechanics.

2.2 Energy in Nuclear Reaction

In nuclear reaction; the particles/ ray and target nucleus exchange energy and the energy of the system conserved. Let us consider the reaction of particle a and target nucleus A.A general relation depicted in figure(2.1) with a target nucleus at rest.

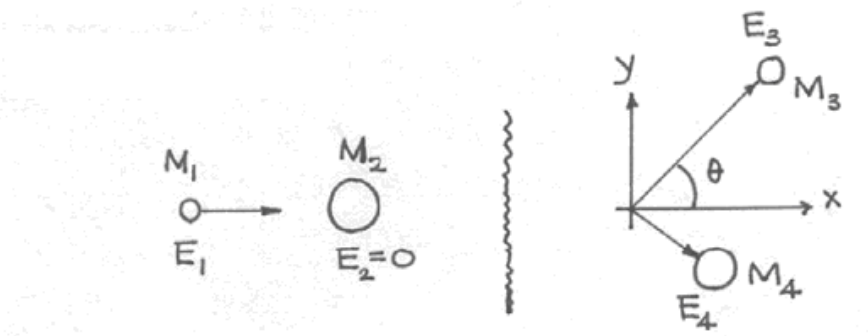


Figure 2.1: Reaction of a projectile with a target nuclei

$$a + A \rightarrow b + B \tag{2.2.1}$$

Where particle a has mass m_1, Particle b has mass m_3 , the target nucleus and product nucleus has mass/ rest mass/m_2 and m_4 . The Q-value for the reaction can be;

$$Q = [(m_1 + m_2) - (m_3 + m_4)]c^2 \tag{2.2.2}$$

Q also can be written in terms of the kinetic energy of the particles and nuclei. (All kinetic energies are in the laboratory coordinate system, LCS, unless specified otherwise.)

Since Q $= T_3 + T_4 - T_1$; the reaction can takes place only if m_3 and m_4 emerge with positive kinetic energies (LCS)[7],[8],[9].

$T_3 + T_4 \geq 0$ or $Q + T_1 \geq 0$[7]

We will see that this condition although quit reasonable, is necessary but not sufficient for the reaction to occur. A fraction of kinetic energy brought in by the incident particle m_1 goes in to the motion of the center of mass and is therefore available for reaction .To see what is the energy available for the reaction we can look in to the kinetic energy of the reacting particles in CMCS (center of mass coordinate system)[9]. First, the kinetic energy of center of mass, in the case where the target nucleus is at rest is;

$$T_0 = \frac{1}{2}(m_1 + m_2)v^2 \tag{2.2.3}$$

Where the center of mass speed is v =[$\frac{m_1}{m_1+m_2}$]v_1, v_1 being the speed of the incident particle. The kinetic energy available for reaction is the kinetic energy of the incident particle T_1, minus the kinetic energy of the center of mass, which we can denote as;T_j

$$T_j = T_1 - T_0 = \frac{m_2}{m_1 + m_2}T_1 \tag{2.2.4}$$

$$= \frac{1}{2}m_1 v_1^2 + \frac{1}{2}m_2 v^2 \tag{2.2.5}$$

The second line in the above shows that T_j, is also the sum of the kinetic energy of particles 1 and 2 in CMCS[8].In addition to the kinetic energy available for reaction there is also the rest mass energy available for reaction, as represent by the Q- value. Thus the total energy available for the reaction is the sum of T_j and Q. A necessary

and sufficient condition for reaction is therefore[7],[8];

$$E = Q + T_j \geq 0 \tag{2.2.6}$$

We can rewrite it as;

$$T_1 \geq -Q\frac{m_1 + m_2}{m_2} \tag{2.2.7}$$

If Q > 0, the condition is always satisfied, which is expected since the reaction is exothermic. For Q <0, the condition shows that the threshold energy; the minimum value of the incident particle kinetic energy for reaction; is the rest mass deficit. The reason for needing more energy than the rest mass deficit, of course, is that energy is needed for the kinetic energy of the center of mass.

At threshold, Q + $T_j = 0$. So m_3 and$m_4$both move in LCS with speed v. (v_3 and v_4 =0). At this condition the total kinetic energies of the reaction products;

$$(T_3 + T_4)_{thres} = \frac{1}{2}[m_3 + m_4]v^2 \tag{2.2.8}$$

2.3 Conservation Law in Nuclear Reaction

A number of conservation conditions apply to any reaction equation: it is important to mention some of them as follows[7].

1,The mass number A and the charge Z must be conserved

2,The total energy must be conserved.

3,The Linear momentum of the reacting particles, before and after the reaction must be conserved.

4,Quantum rules govern the balancing of the angular momentum, Parity and the spin of the nuclear levels.

2.4 Mechanisms of Nuclear Reaction

Among several models of nuclear reaction there are two basic models that explain nuclear reaction depending on, liquid drop model, shell model with spin orbit coupling and complex potential. These are;

1,The compound nucleus model

2,The independent particle model

The compound nucleus model is based on liquid drop model proposed by Bohr (1936), in which the incident particle interacts strongly with the entire target nucleus and the decay of the resulting compound nucleus is independent of the mode of formation. The independent particle model is based on shell model with spin orbit coupling and optical model proposed by Bethe (1949). It describes that, the particle interacts with the nucleus through an effective potential[8]. Each model explains well some aspects of the nuclear reaction. It possible to categorize nuclear reaction into three broad parts depending on the type of collision between the particle and target nucleus[22]. These are;

A, Compound nucleus formation

B, Direct reaction

C, Scattering

2.4.1 Reaction with Compound Nucleus Formation

Nuclear reaction that takes place after incident particles has entered in to the target nucleus. It is a two step process where the incoming particle interacts with the target nucleus to form a compound nucleus which exists for a finite period and then decay

to the final state consisting of an out going particle or rays and the product nucleus. Let us consider the following reaction;

$$a + A \rightarrow c^* \rightarrow b + B \tag{2.4.1}$$

The symbol, (*), indicating that the compound nucleus is in excited state. The first arrow indicates the formation stage and the second arrow indicates the decay state. The mechanism of compound nucleus formation is proceeding by low energy particles at least up to 10Mev[2].

1,Energy and Spin Limitation

The compound nucleus formed in the intermediate stage of the reaction is highly excited and hence unstable. Just like other nuclei, the produced nucleus possesses a system of energy levels. The energy needed to excite the nucleus is only a strictly definite set of energy portions; each of them must coincide with the energy of nuclear level. The excitation energy of nucleus can be provide on the account of kinetic energy of the particles, so, compound nucleus is formed only at same selected kinetic energy otherwise potential scattering takes place[2].

The second limitation lies on the spin of the nucleus. Each excited level has some definite value of mechanical moment, J. The target nucleus at the ground state and the particle has spin I and s. At moderate energy (<0.1Mev), the total angular momentum of the colliding system may be in the range I-s to I+s .If the spin of the exited level of the compound nucleus is not equal to one of the possible values of the total mechanical moment of the colliding particle, the formation of compound nucleus is impossible the fraction of such possible reaction can be determined by statistical factor, g, which depends on J, I and S.

The amount of energy delivered by the particle to the nucleus may determine the life

time (τ) and the decay channel of the compound nucleus. The life time also inversely proportional to the energy width (Γ) and it is less than the time taken by the particle to travel the diameter of the nucleus at a given wavelength. The smaller life time results from the strong nuclear force interaction b/n the nucleons and sometimes due to coulomb and weak electromagnetic interaction in the case of radioactive decay[2].

2,Decay Channels of Compound Nucleus

A compound nucleus formed by interaction with a particle may decay in different ways (channels). The probability of the compound nucleus to decay at a given channel depends up on, the excitation energy of the compound nuclei, the energy level of the target nuclei, the energy level width/ partial level width/ of target nuclei and the nature of the projectile particle. Treating the decay of the compound nucleus like radioactivity decay, the probability of the decay of compound nucleus at excitation energy E and the decay channel where particle b is emitted may be taken as;

$$P_b(E) = \frac{\Gamma_b(E)}{\Gamma(E} \tag{2.4.2}$$

Where $\Gamma(E) = \Gamma_b(E) + \Gamma_e(E)$ + width of any other decay channel allowed with the emission of particles like a, b, e etc.

The cross-section of formation of compound nucleus with the interaction of particle a having kinetic energy T_i and decay of compound nucleus at energy level E by emission of particle b can be found by;

$$\sigma(a,b) = \sigma_c(T_i)P(E) \tag{2.4.3}$$

Some of the decay channels of a compound nucleus formed by interaction of nucleons (neutron, proton) or alpha particles and deuterons are; resonance elastic scattering,

inelastic scattering, radiative captures, fission reactions are the most common ones[8].

2.5 Interaction of Neutron with Matter

Neutrons, however,relatively neutral nucleon, they cannot interact with the atom of the mediums directly through Coulomb force. As a result, even, slow neutrons can be scattered or captured by atomic nuclei strong nuclear force.All processes initiated by neutrons are dependent on the nuclear force.

1,Mechanisms of Neutron Interaction

Free neutrons interact with the nuclei of other materials in various ways, the most common being absorption and scattering. Scattering results in the transfer of some energy and the neutron continues to move through the medium but at a lower energy and hence lower velocity. Neutrons being uncharged do not interact with the electron cloud surrounding the nucleus and, since the nucleus occupies such a tiny space within the atom, the probability of interaction is quite low. This probability is not necessarily related to the size of the nucleus but is measured as a cross section in units of area. The cross sections of different nuclei varies widely and may be greater or smaller than the projected area of the nucleus itself[10].

I, Scattering of Neutrons;scattering of neutrons passing through a give medium may scatter from the nuclei in to two ways these are; elastic scattering and inelastic scattering.

Elastic Scattering: there are two types of elastic scattering of neutrons, potential

scattering and resonance scattering. Potential elastic scattering occurs when a neutron strikes a nucleus and rebounds elastically. Resonance elastic scattering corresponds to neutron decay of compound state formed when the target nucleus capture a neutron. In such a collision kinetic energy is transmitted elastically in accordance with the basic laws of motion. If the nucleus is of the same mass as the neutron then a large amount of kinetic energy is transferred to the nucleus. If the nucleus is of a much greater mass than the neutron then the neutron retains most of the kinetic energy as it rebounds. The amount of kinetic energy transferred also depends upon the angle of impact and hence the direction of motion of the neutron and nucleus after the impact.

Inelastic Scattering: Inelastic scattering occurs when a neutron strikes and enters a nucleus. The nucleus is excited into an unstable condition and a neutron is immediately emitted but with a lower energy than that of the entering neutron. The surplus energy is transferred to the nucleus as kinetic energy and excitation energy. The excited nucleus subsequently returns to the ground state by the emission of a gamma-ray. Such collisions are inelastic since all the initial kinetic energy does not reappear as kinetic energy; some is absorbed by the nucleus and subsequently emitted in a different form (gamma-ray). The emitted neutron may or may not be the one that initially struck the nucleus. In simplistic terms the neutron can be considered simply to be bouncing off an energy-absorbing nucleus[2].

II, Absorption of Neutrons; In the absorption reaction, a neutron combines with a target nucleus of mass number A to produce a compound state in the product nucleus of mass number A +1.The excitation energy of this state is equal to the sum of neutron separation energy and the kinetic energy of neutron. The excited nucleus

may decay by emission of charged particles, gamma rays or two or more neutrons. The type of decay process depends up on the excitation energy of the compound nucleus.

Radiative Capture: Radiative capture can be considered to be similar to the initial process leading to inelastic scattering. A neutron strikes and enters a nucleus. The nucleus is excited but the level of excitation is insufficient to eject a neutron. Instead all the energy is transferred to the nucleus as kinetic energy and excitation energy,the excited nucleus subsequently returns to the ground state by the emission of a gamma-ray. The incoming neutron remains in the nucleus and the nuclide increases its number of neutrons by one. This is a very common type of reaction. It leads to the creation of heavier isotopes of the original element. Many of these may be radioactive and decay over time in different ways. For example we have the following radiative capture reaction[8].

$$^{75}As(n,\gamma)^{76}As \tag{2.5.1}$$

$$^{1}H(n,\gamma)^{2}H \tag{2.5.2}$$

$$^{8}Be(n,\gamma)^{9}Be \tag{2.5.3}$$

Nuclear Transmutation (Charged Particle emission): Nuclear transmutation is similar to radiative capture and inelastic scattering. A neutron strikes and enters a nucleus. The nucleus is excited into an unstable condition but a particle other than a neutron is emitted. The emitted particles are either protons or alpha-particles. This leaves the nucleus still in an excited state and it subsequently returns to the ground state by the emission of a gamma-ray. In this process the atomic number of the nucleus increases by one for proton emission and reduces the total number of protons in the nucleus by two for alpha-particle emission. The original element is thus changed or transmuted

into a different element. The following reactions are the examples of transmutation reaction[10].

$$^{16}O(n,p)^{16}N \tag{2.5.4}$$

$$^{27}Al(n,\alpha)^{24}Na \tag{2.5.5}$$

Neutron Producing Reaction: Neutron producing reactions occur when one or two additional neutrons are produced from a single neutron. As before a neutron strikes and enters a nucleus. The nucleus is excited into an unstable condition as with inelastic scattering but two or three neutrons instead of only one neutron are emitted. The still excited nucleus subsequently returns to its ground state by the emission of a gamma-ray. This is an uncommon reaction occurring in only a few isotopes[12].

Fission Reaction: Although spontaneous fission occasionally occurs, fission is generally induced by neutrons. A neutron strikes and enters a heavy nucleus. The nucleus is excited into an unstable condition as with most of the foregoing interactions. In this unstable condition the nucleus splits into two new mid-range nuclei usually of unequal mass. Since these new nuclei do not need as many neutrons for stability some neutrons are emitted immediately. The surplus binding energy drives the new nuclei (fission fragments) and neutrons away from one another with high velocity. The new nuclei subsequently lose their kinetic energy by ionizing reactions with the surrounding nuclei through which they pass and return to their ground states by emission of gamma-rays. They are invariably still unstable with too many neutrons and subsequently decay usually by beta-particle and gamma-ray emission. The high-energy neutrons lose energy by scattering collisions with nuclei of the surrounding medium and are subsequently generally captured by other nuclei to produce one of

the reactions described in this section.

An isotope is referred to as "fissile" if the excitation energy provided by the incoming neutron is sufficient to cause fission. Thus, the incident neutron can have zero kinetic energy and still cause fission.The major fissile isotopes are: U-233, U-235, Pu-239, and Pu-241 Of these, only U-235 is naturally available (0.71persent uranium)[13]

2.5.1 Neutron Nuclear Cross Sections

I ,Microscopic Cross Sections:; A solid material may be considered as being made up of tiny nuclei suspended in empty space. Each nucleus has an imaginary projected area, which may interfere with the passage of a neutron. A neutron entering the solid will see these projected areas scattered everywhere but they are so small and so far apart (as seen by the neutron) that the chances of hitting one of these nuclei is practically nil. Eventually a neutron may hit a nucleus and will then interact with it in any of a number of possible ways. Some other neutrons will simply pass it without any interaction. It is interesting to note that the imaginary projected area or target area of a nucleus may be larger or smaller than the actual projected area as determined from the physical size of the nucleus. It may be larger because the nucleus has a sphere of influence surrounding it and any neutron passing within this sphere of influence may be attracted to interact with it. It may be smaller because some nuclei may allow neutrons to pass right through themselves without any interaction-taking place. The imaginary projected area may thus be considered as being related to the probability of a reaction occurring-the larger the area, the greater the probability of interaction. It is also interesting to note that for different reactions with the nucleus there are different degrees of probability of interaction and therefore effectively

different imaginary projected areas. For example, some nuclei have a larger imaginary target area for elastic scattering than for radiative capture illustrating that there is a greater probability of elastic scattering occurring. These imaginary projected areas are known as nuclear cross sections and indicate the probability of any interaction occurring. The cross sections of the nuclei of individual atoms are measured in square centimeters, square meters or barns where: 1 barn = 1 x $10^{-24} cm^2$[12]

If the actual projected area of a nucleus is calculated it is found that for mid-range elements with an atomic mass number of about 90[10]. This area is equal to 1 barn. Lighter elements have smaller projected areas and heavier elements larger projected areas. A cross section of 1 barn indicates immediately that the imaginary target area is roughly equal to the actual projected area of the nucleus. This allows cross sections to be visualized. There are different types of cross sections; in fact there is one type of cross section for each type of neutron interaction with the nucleus except for the relatively rare neutron producing and nuclear transmutation reactions. These different cross sections may be added to give a total cross section. The nomenclature for different cross sections is given below with the different types of interactions[12]:

σ_s = Elastic scattering cross section

σ_i = Inelastic scattering cross section

σ_n, γ= Radiative capture cross section

σ_a = Absorption cross section

σ_f = Fission cross section

Values for these are tabulated but are often combined into two main types of interactions:

σ_s = Scattering cross section $\sigma_s$andσ_i.

σ_a = Absorption cross section σ_n, γ and σ_f.

When these are combined they are added together so that the scattering cross-section includes both elastic and inelastic scattering and the absorption cross-section includes both radiative capture and fission cross section. For a particular isotope all the individual microscopic cross sections can be added to give the total microscopic cross section[10].

$$\sigma_{total} = \sigma_s + \sigma_i + \sigma_a + ... \tag{2.5.6}$$

Generally however any particular calculation requires the application of a specific microscopic cross section only.

II Macroscopic Cross Sections; The macroscopic cross section Σ is the cross section density in a material. It is defined as the number of nuclei per unit volume N multiplied by the microscopic cross section σ. The units are the inverse of length ($\mathrm{cm}^{-2} or m^{-1}$)

$$\Sigma = N\sigma \tag{2.5.7}$$

This provides a basis for the comparison of different materials. A dense material with nuclei of small cross section would be seen by neutrons to be effectively the same as a rare material with nuclei of large cross section. For a single isotope the macroscopic cross section can be determined from the above equation. This gives the effective cross section density in a pure material and indicates the probability of a neutron interaction within that material. If there is a homogeneous mixture of different isotopes the cross section density can be calculated separately for each and then added to give the total macroscopic cross section[10].

$$\Sigma = N_a\sigma_a + N_b\sigma_b + N_c\sigma_c... \tag{2.5.8}$$

Note that N is the number of nuclei or atoms of each element per unit volume in the material.

So far, neutron's cross section depends on: nature of target nucleus and energy of interacting neutrons. The elastic scattering cross-section σ_s is constant in the low energy potential region, Fluctuates in the resonance region and falls slowly with increasing energy in the smooth region. The inelastic cross-section σ_i is only apparent above certain threshold energy. The radiative capture cross-section $\sigma_\gamma$is inversely proportional to velocity in the 1/v region, fluctuates in the resonance region and drops to a low value or disappears at high energies. The total cross-section σ_t is a summation of all the individual cross-sections including fission[2].

Neutron Attenuation: When a beam of neutrons impinges upon a solid body the neutrons interact with nuclei within the body. Those not interacting continue through the body. As the beam progresses through the body more and more interactions occur and less and less neutrons continue on through the material. The beam of neutrons diminishes in intensity and is attenuated by the material. The decrease in intensity dI over any section of material is proportional to the neutron beam intensity I, microscopic cross-section of the material σ, number density of nuclei and the thickness of the material dx:

$$dI = -\sigma I N dx \tag{2.5.9}$$

If the macroscopic cross section is used this becomes:

$$dI = -I\Sigma dx \tag{2.5.10}$$

The solution to this differential equation is:

$$I = I_O e^{-\Sigma x} \tag{2.5.11}$$

2.5.2 Neutron Moderation

Neutron Energy Changes When neutrons interact with nuclei by elastic or inelastic scattering their energy is decreases. Generally neutrons produced from any neutron source have energy above 2 MeV. While neutrons after thermalization have an energy of about 0.025 eV[2]. The number of interactions to bring about this degradation depends upon several factors including the initial energy of the neutrons and the type of scattering. Inelastic scattering generally requires that the incoming (captured) neutron have sufficient energy to excite the nucleus to a level that will result in the ejection of a neutron. Hence inelastic scattering occurs only at high neutron energies and the resulting neutrons will have very much lower energies. Elastic scattering, on the other hand, occurs at all neutron energies and may not necessarily decrease the neutron energy very much. Hence, any neutrons produced from neutron sources that suffer inelastic collisions initially will subsequently be subject to a series of elastic collisions. Those that suffer elastic collisions initially will also likely continue to degrade their energy by elastic collisions. Hence most collisions are elastic[9].

Logarithmic Mean Energy Decrement: When neutrons interact with nuclei in elastic scattering collisions they lose energy. The amount of energy lost depends upon the mass of the nucleus and the angle of incidence of the neutron. More energy is lost when a neutron strikes a light nucleus than when it strikes a heavy one. Also more energy is lost in a head-on collision than in a glancing collision[2]. The minimum energy after one collision is:

$$E_{min} = \alpha E_0 \tag{2.5.12}$$

Where:

$$\alpha = [\frac{A-1}{A+1}]^2 \tag{2.5.13}$$

Here A is the atomic mass number of the nucleus and it is evident that the higher this number the closer α will be to unity and the smaller the maximum loss in energy. Considering the results of various angles of incidence of the incoming neutron, it is found that the average energy after one collision is:

$$E_{ave} = \frac{1}{2}(1+\alpha)E_o \tag{2.5.14}$$

The average energy loss after one collision is thus given by:

$$\Delta E = E_o - E_{ave} \tag{2.5.15}$$

$$\Delta E = \frac{1}{2}(1-\alpha)E_o \tag{2.5.16}$$

Since the loss or change in energy depends upon the incoming neutron energy and since this is lower in each subsequent collision in an exponential manner, it is convenient to express the change in energy in logarithmic terms. Furthermore, since the change in energy is different for each collision, the average of the logarithmic values of the initial energy E_o and resultant energy E is used[10]. The logarithmic mean energy decrement ζ is the average of the difference of these logarithmic energy values:

$$\zeta = [lnE_o - lnE]_{ave} \tag{2.5.17}$$

$$\zeta = [-ln(\frac{E}{E_o})]_{ave} \tag{2.5.18}$$

The value of the logarithmic mean energy decrement for any isotope of atomic mass number A is given as follows:

$$\zeta = 1 + [\frac{(A-1)^2}{2A}]ln\frac{A-1}{A+1} \tag{2.5.19}$$

An approximate value for the logarithmic mean energy decrement is given by the following empirical equation:

$$\zeta = \frac{2}{A + \frac{2}{3}} \tag{2.5.20}$$

This equation has negligible error for all but the very lowest atomic mass numbers hence it is widely adopted in place of the theoretical equation. The number of elastic collisions required for the neutron energy to drop from an initial energy to a final energy is then given by

$$N\zeta = ln\frac{E_i}{E_f} \tag{2.5.21}$$

The value for a high-energy neutron from fission (2MeV) to become thermalized at normal conditions (0.025eV) is 18 for Hydrogen, 43 for Helium and 115 for Carbon. Lighter elements are efficient at reducing neutron energy because they are light and absorb a lot of energy when struck by a nucleus[2].

Chapter 3

Neutron Activation Analysis

Neutron Activation Analysis (NAA) is a quantitative and qualitative method of high efficiency for the precise determination of a number of main-components and trace elements in different types of samples. NAA, based on the nuclear reaction between neutrons and target nuclei, is a useful method for the simultaneous determination of about 25-30 major, minor and trace elements of geological, environmental, biological samples without or with chemical separation[14].

In NAA, samples are activated by neutrons. During irradiation the naturally occurring stable isotopes of most elements that constitute the given are transformed into radioactive isotopes by neutron capture. Then the activated nucleus decays according to a characteristic half-life; some nuclide emit beta particles and gamma-quanta, too, with specific energies.As the irradiated samples contain radionuclide of different half-lives different isotopes can be determined at various time intervals[15].

The advantage of the method is its sensitivity and accuracy especially in respect of some trace elements. The method is of a multielement character, i.e. it enables the simultaneous determination of many elements without chemical separation. In the

case of instrumental determination, (INAA), the preparation of samples involves only the preparation of representative samples, and this reduces the danger of contamination to a minimum and accelerates the whole analytical process. If the determination of some special elements or groups of elements can be carried out only through chemical separation, it is possible to carry out after irradiation[16].The development of the method has contributed to the elaboration of some very simple and accurate methods of standardization, which lead to a surpassingly accurate analysis.

3.1 Instrumental vs. Radiochemical NAA

With the use of good sample handling, prompt/Delayed gamma-ray measurement with solid-state detectors, (HPGE), and computerized data processing it is generally possible to simultaneously measure more than thirty elements in most sample types without chemical 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 can be performed infrequently[16].

3.2 Prompt vs Delayed Neutron Activation Analysis

Depending up on the time of measurement, NAA can be divided in two categories.

(a)*In-beam (Prompt gamma ray neutron activation analysis (PGNAA))*: The measurements should be taking place during irradiation. The PGNAA technique is usually 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 times 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 section elements which decay too rapidly to be measured by DGNAA; elements that produce only stable isotopes after the emission of prompt gamma ray; or elements with weak gamma ray intensities[14].

(b) *Off-beam (Delayed gamma ray neutron activation analysis(DGNAA))*; The gamma ray measurement takes place after sample irradiation. This technique is used for the most majority of elements that produce radioactive nuclides. The DGNAA technique is flexible with respect to time such that the sensitivity for a Long-lived radionuclide that suffers from the shorter-lived radionuclide to decay[14].

3.3 Fundamental Equations in NAA

Consider an isotope X; in the nuclear reaction, it absorbs a neutron (n).

$$X + n \rightarrow X^* + \gamma \tag{3.3.1}$$

Where γ is a prompt gamma ray, X^* designates excited radioactive nucleus. X^* nucleus decay via beta minus decay which converts a n (neutron) to a P (proton) thereby creating a stable isotope of Y, a negative electron, an antineutrino ν and gamma rays.

$$X^* \rightarrow Y + \beta^- + \overline{\nu} + \gamma \tag{3.3.2}$$

So in the nuclear reaction we form X* and it decays. Hence,the number of X* radioactive isotopes formed with time is given by an equation for;

accumulation = production in reaction - disappearance by decay.

That is.

$$\frac{dN}{dt} = \phi\sigma No - \lambda N \tag{3.3.3}$$

where ϕ =thermal neutron flux in neutrons/cm^2 sec

σ =capture cross section, i.e., probability of absorbing neutrons in cm^2

λ= decay constant

No = original number of X nuclei in the neutron source. This equation can be integrated to obtain

$$N = \frac{\phi\sigma No(1 - e^{-\lambda t_{irr}})}{\lambda} \tag{3.3.4}$$

where N is the number of nuclei activated at time t_{irr}. The term within parentheses is called the Saturation Factor "S" ,If;

$$(a), t_{irr} \gg t_{1/2} \tag{3.3.5}$$

then the value of S = 1 and the above expression becomes;

$$N = \frac{\Phi\sigma N_O}{\lambda} \tag{3.3.6}$$

If,

$$(b) t_{irr} \ll t_{1/2}, e^{-\lambda t} \sim 1 - \lambda t \tag{3.3.7}$$

so the number of activated target nuclei becomes;

$$N = \phi\sigma N_o t_{irrad} \tag{3.3.8}$$

The activity of the sample just after irradiation time, t_{irr}, can be written as,

$$A = \phi\sigma N_o[1 - e^{-\lambda t_{irr}}] \tag{3.3.9}$$

where A =λ N, number of decays per second at the end of irradiation, (Activity) dps and

$$N_o = \frac{m \times \theta \times 6.023 \times 10^{23}}{w} \tag{3.3.10}$$

m = mass of the element in the irradiated sample

θ =isotopic abundance

w = Atomic weight of the element

After a delay of time t_d(the time needed to take the samples from the neutron source to the detector) ,the decay rate of the sample becomes;

$$A_o = \phi\sigma N_o[1 - e^{-\lambda t_{irr}})]e^{-\lambda t_d} \tag{3.3.11}$$

After counting time of t_c, the counts per second (n) accumulated in the given counter becomes;

$$n = \eta h\phi\sigma N_o[1 - e^{-\lambda t_{irr}]e^{-\lambda t_d}[1-e^{-\lambda t_c}]} \tag{3.3.12}$$

$$n = \eta h A_O[1 - e^{-\lambda t_c}] \tag{3.3.13}$$

where h is the gamma ray abundance and η the photo peak efficiency of a gamma ray[17]. The neutron flux up on the the target nuclei may be calculated from the above equation.i.e

$$\Phi = \frac{ne^{\lambda t_d}}{h\eta\sigma N_o[1 - e^{-\lambda t_{irr}}][1 - e^{-\lambda t_c}]} \tag{3.3.14}$$

where n is the activity of the sample in cps.

3.4 Principles of the Method

In the process of NAA the neutrons interact with the stable isotopes of the target element converting them to radioactive ones. The compound nucleus formed emits gamma rays promptly with extremely short half lives in the order of ms and these can be measured during irradiation through a technique called prompt gamma activation analysis (PGAA). In most cases, the radioactive isotopes decay and emit beta particles accompanied by gamma quanta of characteristic energies and half-life, $T_{1/2}$, and the radiation can be used both to identify and accurately determine the amount of elements of the sample. Subsequent to irradiation, the samples can be measured instrumentally by a high resolution semiconductor detector, or for better sensitivity, chemical separations can also be applied to reduce interferences. The qualitative characteristics are: the energy of the emitted gamma quanta (E_γ) and the half life of the nuclide ($T_{1/2}$)[14]. The quantitative characteristic is: the intensity, which is the number of gamma quanta of energy E_γ measured per unit time. Sample preparation means in most cases only pulverising, homogenising, mass determination, packing, as well as the selection of the best analytical process and the preparation of the standards, if any. For irradiation one can choose from the various types of neutron sources according to need and availability. If there is no chemical treatment, the measurement is performed after a suitable cooling time . Measurement, evaluation and calculation involve taking the gamma spectra and the calculating trace element concentrations of the sample. The most widely used gamma spectrometers consist of germanium based semiconductor detectors connected to a computer used as a multichannel analyser for spectra evaluation and calculation.

3.5 Irradiation Facilities

Neutron Sources;

Isotopic neutron sources; In the case of the most frequently used isotopic neutron sources an alpha emitting radioactive material is mixed with beryllium and an (α,n) reaction generates the neutrons. The major advantage is that the isotopic neutron sources can be made portable and generate a stable neutron flux. But, as the neutron flux is rather low in comparison to a nuclear reactor their use in NAA is limited to the determination of elements of high activation cross section which are present in major concentrations[24].

3.6 Irradiation Conditions

The incident flux of the irradiating particles directly affects the level of radioactivity produced, generally the neutron self absorption is negligible.

In a nuclear reactor the neutron flux cannot be freely changed, but there are some possibilities to choose. The effective cross-section of a nucleus depends on the energy of the bombarding particles so; the use of a thermal neutron filter is an important option of selection. This type of analysis is called Epithermal Neutron Activation Analysis (ENAA), where Cd or B is used as a filter.Some elements can be determined with higher sensitivity by ENAA, e.g. As, Br, Rb, Sr, Mo, Sb, Ba, Ta and U[16]. Measurement of radioactivity Generally, the activation product emits more than one gamma quanta, each of different energy and emission probability. To choose the proper analytical gamma line(s) the gamma abundance, the efficiency and the possible interferences must be considered, in exceptional cases self absorption can be the main problem. Sometimes, it is also possible to use more than one gamma line in order to

improve the accuracy and the reliability of the measurement. The efficiency of the measurement depends on the solid angle presented by the detector to the sample. Therefore, the detection efficiency is highly affected by the shape of the sample and its distance from the detector. The counting rate can also be changed in this way. In the case of shortage of samples i.e. very low concentrations in samples the use of a well-type detector is desirable[24].

3.7 Experimental Parameters

By optimizing the irradiation, decay and measuring times, a lot of elements can be determined with higher sensitivity. A number of interferences can be avoided in this way, too. For elements with short half lives the shortest irradiation and cooling times are determined by the technical limitations. In this case some special methods are required, e.g. the process of cyclic activation. In this method the samples are repeatedly irradiated and counted, and then, the spectra are summed. The transfer, irradiation and counting times, as well as the number of cycles should be optimized. The radioactive isotopes of long half lives produced after irradiating the elements for long time (e.g. eight hours) in a thermal channel of the reactor are measured several times. The cooling times are one week and one month or longer in special cases. By this way, usually 25-30 elements can be determined in different types of samples[15].

3.8 Absolute Method

Since the theory of NAA is well established, an "absolute" standardisation procedure can be applied. The quantitative measurement can be affected by determining the neutron flux and counting the absolute gamma/beta rays. The direct calculation of

concentration is made by applying nuclear constants according to equations earlier in this chapter. According to Girardi's investigation, systematic errors up to tens of percent may be the consequences of the uncertainties of nuclear data taken from literature, especially those on decay schemes and activation cross-section[17].

3.9 Relative Method

The method is based on the simultaneous irradiation of the sample with standards of known quantities of the elements in question in identical positions, followed by measuring the induced intensities of both the standard and the sample in a well known geometrical position. The calculation of the unknown quantity (M_{sample}) is made according to equation given below[24].

$$\frac{A_{sample}}{A_{standard}} = \frac{M_{sample}}{M_{standard}} \tag{3.9.1}$$

The disadvantage of the classic relative method lies in the multielement application. The procedure of the standard preparation and counting is rather laborious, and this is coupled with the occasional loss of information if an unexpected element appears for which no standard has been irradiated. The use of home-made multielement standards can be an answer to these problems, but this raises the question of homogeneity and stability. There are commercial multielements Standards Reference Materials (SRM) available[17].

3.10 Application

Because of its detection sensitivity, the NAA method has found important applications in many fields, like, medicine, biology, geochemistry, industry, art, military, archaeology, environmental science and forensic chemistry[5].

Chapter 4

Experiment

4.1 Introduction

Radioactivity can be regarded as a particular form of nuclear reaction where the emission of the decay product(s) takes a much longer time in comparison with the time taken for a particle of typical energy to move a large distance from the nucleus. In this experiment we will study how to identify a given sample of interest and its capture cross section in the case of radioactivity resulting from bombardment with slow neutrons. "Slow" means with energies comparable with the Boltzmann energy kT, i.e. around 0.03 eV.

4.2 Experimental Setup

4.2.1 Neutron Source

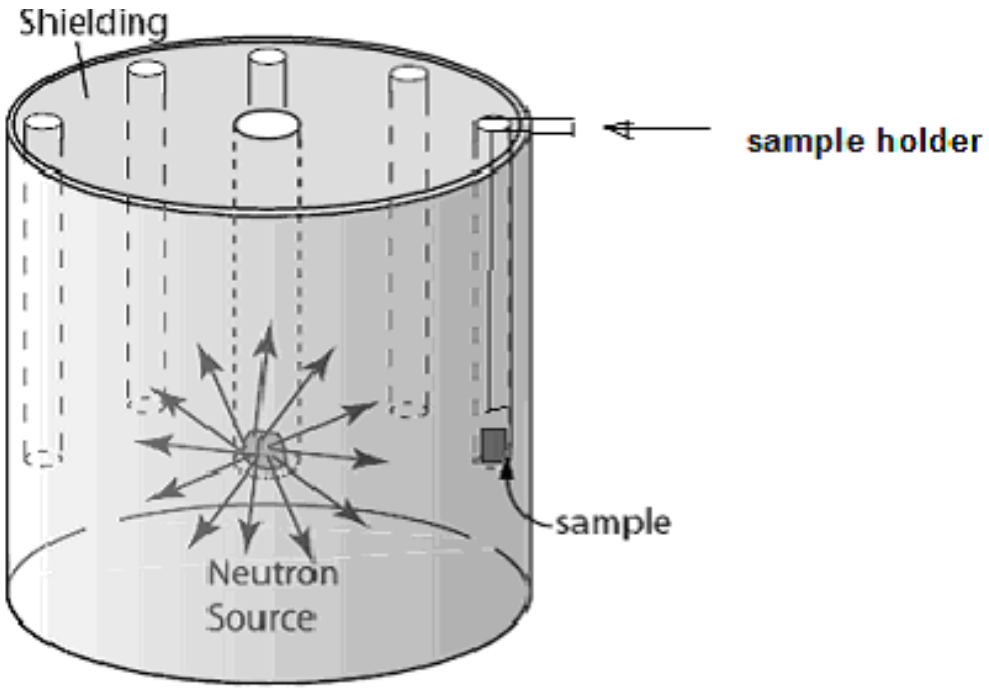


Figure 4.1: schematic diagram of Am-Be neutron source

The neutrons are produced in an americium-beryllium source. In this source 241Am is mixed into beryllium powder. The following reactions occur[11]

$$^{241}Am \rightarrow ^{237} Np + \alpha(2Curie, i.e. 74GBq). \tag{4.2.1}$$

$$\alpha + ^{9} Be \rightarrow ^{12} C + n(5.75MeV) \tag{4.2.2}$$

One in 40000 of the alphas produce two neutron so that the yield of neutrons is about 4 million per second. Neutrons ejected with this energy can cause nuclear reactions but the cross-section is small. They can be slowed down by putting them into a medium containing light nuclei which does not contain nuclei that absorb them. Such a medium is called a moderator; the neutrons elastically scatter from the light

nuclei, losing a portion of their energy at each collision. After a small number of collisions their energy becomes comparable with the thermal energy of the scattering nuclei and they are said to be thermalised. Suitable moderator nuclei are protons, deuterons, beryllium and all the stable nuclei of carbon, nitrogen and oxygen. All have very low cross-sections for neutron capture except the proton. This disadvantage is partly compensated for by the proton's small mass and by the fact that suitable hydrogen containing chemical compounds are freely available: water, plastics, wax, etc[11]. With a source of slow neutrons we can cause nuclear reactions to occur by processes such as

$$^{A}Z + n \rightarrow ^{A+1} Z + \gamma \tag{4.2.3}$$

where $^{A+1}$Z is radioactive with a "reasonable" half life and where the slow neutron capture cross-section is reasonably large. By "reasonable" we mean that the half life is much greater than the time required to take the irradiated sample from the source to the counter[11].

4.2.2 Geiger-Muller Counter

A typical Geiger-Muller (GM) Counter consists of a GM tube having a thin, mica end-window, a voltage supply for the tube, a scaler to record the number of particles detected by the tube, and a timer which will stop the action of the scaler at the end of a preset interval. The sensitivity of the GM tube is such that any particle capable of ionizing a single atom of the filling gas of the tube will initiate an avalanche of ionization in the tube. The collection of the ionization thus produced results in the formation of a pulse of voltage at the output of the tube. The amplitude of this pulse, on the order of a volt or so, is sufficient to operate the scaler circuit with little further

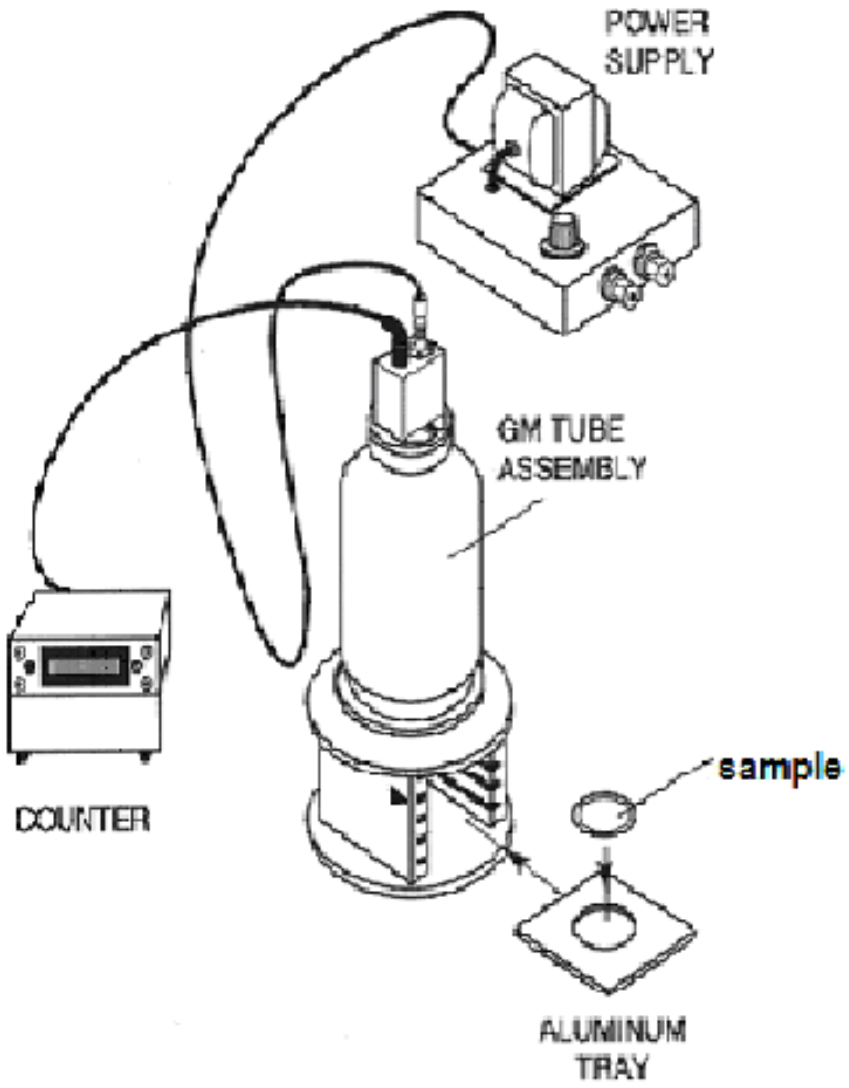


Figure 4.2: setup of GM Counter

amplification. However, the pulse amplitude is largely independent of the properties of the particle detected, and, therefore, can give little information as to the nature of the particle. In spite of this fact, the GM Counter is a versatile device in that it may be used for counting alpha particles, beta particles, and gamma rays with, however, varying degrees of efficiency[18].

4.2.3 Characteristics of the GM Counter

Every GM tube has a characteristic response of counting rate versus voltage applied to the tube. A curve representing the variation of counting rate with voltage is called a plateau curve because of its appearance. The plateau curve of every tube that is to be used for the first time should be drawn in order to determine the optimum operating voltage. To find the plateau curve for our tube; we have to follow the procedure outlined below.

A. Check to see that the high voltage as indicated by the meter on the instrument is at its minimum value.

B,Insert a radioactive source (^{137}Cs) into one of the shelves of the counting chamber.

C, Turn on the count switch and slowly increase the high voltage until counts just begin to be recorded by the scaler. The voltage at which counts just begin is called the "starting voltage" of the tube.

D,Beginning at the nearest 30 volt mark above the starting voltage, takes 20 seconds counts every 30 volts until a voltage is reached where a rapid increase in counts is observed. Tabulate counts versus voltage.

E, Plot the data of (D). A plateau should be observed in the curve. The optimum operating voltage will be about the middle of the plateau. Set the high voltage to this point and record. The slope S of the plateau of a GM tube serves as a figure of

Background Radiation =14.6c/s

no	Volt(v)	Activity(c/20s)	$\pm\sqrt{N}$
1	330	69	8.307
2	360	4952	70.370
3	390	5034	70.951
4	420	5040	70.993
5	450	5047	71.042
6	480	5063	71.154
7	510	5087	71.323
8	540	5122	71.568
9	570	9007	94.906
10	600	9546	97.704
11	630	9645	98.209

Figure 4.3: Volt vs.Activity data of GM-Counter

merit for the tube. The slope is defined to be the percent change in count rate per 100 volts change in applied voltage in the plateau region. A slope of greater than 10 percent indicates that the tube should no longer be used for accurate work. The slope may be computed using;

$$S(precentper100V) = [\frac{2(C_2 - C_1)10^4}{(C_2 + C_1)(V_2 - V_1)}]^0/_0 \tag{4.2.4}$$

where V_2 is the voltage at the high end of plateau, C_2 is the count rate at this voltage, V_1 is the voltage at the low end of the plateau, and C_1 the corresponding rate. The relation between applied volt and the count rate used to determine the effective operating voltage of the counter.

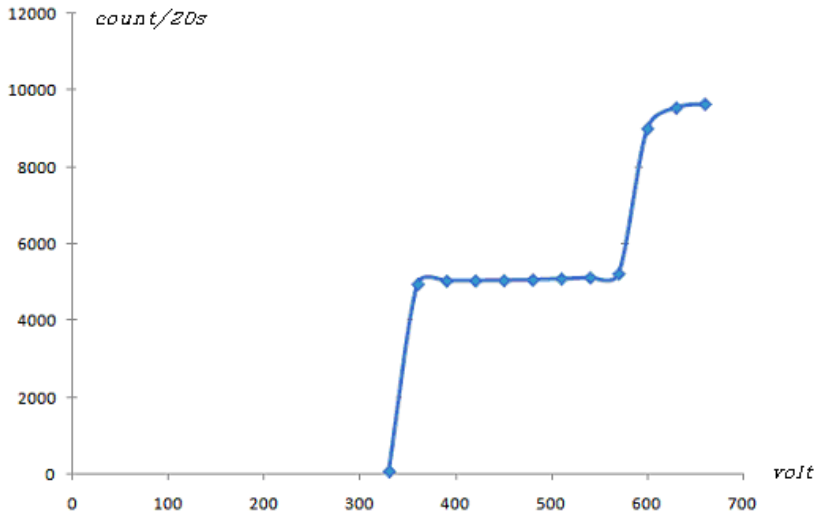


Figure 4.4: GM-Counter characteristics

The slope of the Plateau in fig(4.4) region of our GM-Counter is 1.875 percent.

1,The operating voltage is at 460v.

2,Mean count (n) =5049.286

3,Standard deviation ($\pm\sqrt{n}$) =71.058

4,71 percent of the counts in the Plateau region falls b/n n $\pm\sqrt{n}$.

5,Background radiation is 14.6c/20s

4.3 Resolving time of the GM Counter

There is an interval of time following the production of a pulse in the GM tube during which no other pulse can be recorded. This interval is called the resolving time of the system. If this time is known it can be used to make a correction to the observed count rate to yield the true count rate. A good estimate of the resolving time can be determined by the equation;

$$\tau = \frac{R_1 + R_2 - Rc}{R_2 R_1} \tag{4.3.1}$$

where R_1 and R_2 are the counts of two separate sources and Rc is the combined count of the two sources[18].

The resolving time τmay be used to correct an observed count rate using the expression:

$$R = \frac{r}{(1 - \tau r)} \tag{4.3.2}$$

Where r = Observed count rate

R = True count rate

4.4 Background Measurements

Extraneous radiation called background radiation is always present. Gamma rays emitted by certain radioisotopes in the ground, in air, and from various building materials as well as cosmic radiation from outer space can all provide counts in a detector in addition to those from a sample being measured. This background counting rate should always be subtracted from a sample counting rate in order to obtain the rate from the sample alone.

4.5 Sampling and Irradiation

The chemical compounds or samples prepared for the experiment are;

Potassium iodide(KI)

Pure Arsenic powder(As-75)

These samples were manufactured by Jonson Mathey inc.com ,london make and were specpure which are borrowed from india by prof.A.K Chaubey in a powder form. Due to the volatility of pure iodine atom , the metal iodide salt is preferred for the determination of neutron flux. 99 percent of the salt has I-127 than the rest isotopes of iodine and the natural abundance of As-75 is 100percent.The samples has to be prepared suitably for irradiation in solid form by putting them in the ring and fasting both side by sticky tape,so that, we can have three samples having the following masses.

1,Potassium iodide(sample1) = 0.1985g

2,Potassium iodide(sample2) = 0.2287g

3,Arsenic(As-75) (sample3) = 0.1444g

The above samples were made ready by sandwiching the arsenic sample in between the two KI targets in order to have fixed geometry and by putting them in the sample holder. Irradiation process started when the samples are inserted in the Am-Be neutron source found in our laboratory of AAU Science faculty nuclear physics department.The duration of irradiation was about 20days for the first experiment and 8days for the second experiment.

4.6 Measurement

Before the measurement of the activity of the samples detection system or detector (GM-Counter and the associated electronics) has to be prepared in its best operating region and the background radiation had to be determined , then samples can be taken by noting down the time of irradiation stopped and taken to the detector as

fast as possible in order to minimize the delay time(t_d).The activity of each samples measured consecutively by putting at zero distance from the GM-Counter for 100second,counting time(t_c).The two KI samples counted for 24hour and the long half-life As-75 was counted for 8 days(eight half-life).At each count the corresponding time is recorded.

Two such measurement were done with different samples and cross section for ^{75}As(n,γ)^{76}As reaction was calculated from both measurements.The final result is taken from the mean of both experiments.

Chapter 5

Data and Data Analysis

Using instrumental neutron activation analysis method ,the reaction of thermal neutron with As-75 can be studied by taking two KI samples which are used to determine the thermal neutron flux.

5.1 Activation of KI

Thermal neutron reaction with the KI target results in (n,γ) reaction with that of I-127 Metallic element.

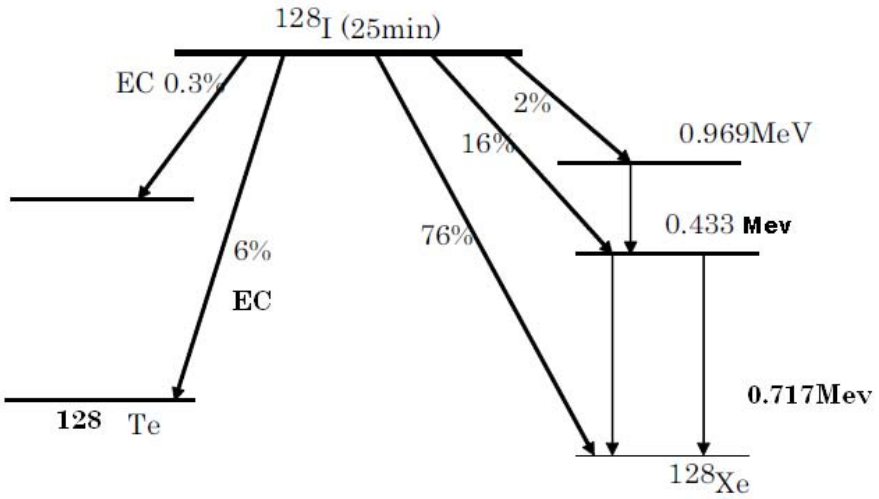
$$n +^{127} I \rightarrow [^{128}I]^{\star} \rightarrow Xe + \beta^{-} + \nu + 2.119Mev(93.1percent) \tag{5.1.1}$$

$$[^{128}I]^{\star} + \rightarrow^{128} Te + \overline{\nu} + 1.252Mev(6.9percent) \tag{5.1.2}$$

The excited iodine[^{128}I] de-excited to the ground level of Xe atom by emitting three types of beta particles having the following end point energy and branching ratio[21].

$$\beta_1 \rightarrow 2.12Mev \rightarrow 76percent \tag{5.1.3}$$

$$\beta_2 \rightarrow 1.665Mev \rightarrow 15.5percent \tag{5.1.4}$$



- Production: $^{127}I(n,\gamma)^{128}I$
- Mode of decay: β- (93.7%), EC (6.3%)
- Main gamma energy: 443KeV(16%), 527KeV(1.57%), 969.4KeV(0.4%)

Figure 5.1: Decay scheme of I-128

$$\beta_3 \rightarrow 1.125Mev \rightarrow 2percent \tag{5.1.5}$$

5.2 Activation of Arsenic

Arsenic is a non metallic element discovered in 1250 AD by a person called Albertus Magnus.Naturally,it exists in compound form (AsO_4,H_3AsO_4,As_2O_3,etc).Pure solid As is obtained by chemical processing[20]. Arsenic has 32 known isotopes (^{66}As-^{92}As)[26].In general, the arsenic isotopes emit β^+, β^-, particles, gamma rays and neutrons when bombarded by neutrons.When As-75 bombarded by slow neutrons, it emits betas and gamma with a half life of 26.3hr and end point energy of 2.97Mev[26],[21].The thermal neutron capture cross- section is 4.230±0.1 barns[21].

Now we have suitable nuclide in elemental arsenic, ^{75}As, of neutron bombardment leads to long lived activities and the decay curve can be drawn when the count rate of a GM counter is plotted against time.The reaction produced by the neutron is:

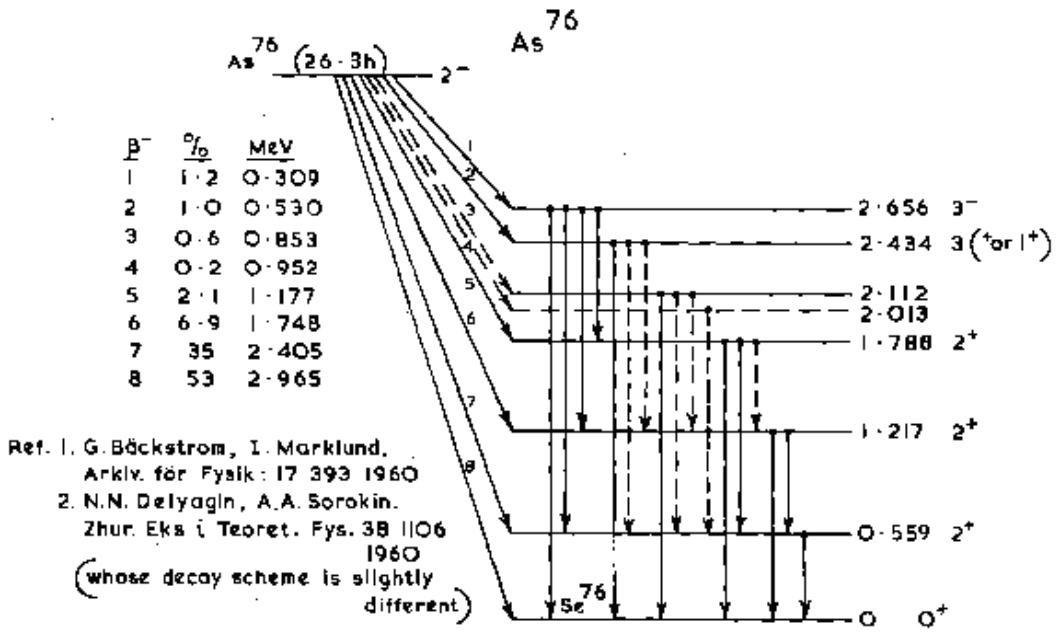


Figure 5.2: decay scheme of ^{76}As

$$n+^{75}As \rightarrow^{76} As+\gamma(prompt) \rightarrow^{76} As+\beta^{-}+\gamma(delayed)(Emax = 2.97MeV, t_{\frac{1}{2}} = 26.3hr) \tag{5.2.1}$$

Backgroud Radiation = 37c/100s

NO	TIME(SEC)	COUNT/100SEC	±√N
1	100	241	15.524
2	523	239	15.459
3	931	207	14.387
4	1328	182	13.490
5	1728	155	12.445
6	2109	132	11.489
7	2490	129	11.357
8	2865	119	10.908
9	3232	89	9.434
10	3610	77	8.775
11	4088	62	7.874
12	4370	35	5.916
13	4748	39	6.245
14	5128	33	5.744
15	5502	32	5.657
16	5872	31	5.567
17	6256	27	5.196
18	6644	32	5.657
19	7032	31	5.567
20	7413	10	3.162
21	7787	22	4.690
22	8155	15	3.873
23	8584	15	3.873
24	8966	07	2.646
25	9340	14	3.741
26	9820	11	3.316
27	10486	14	3.742
28	11152	10	3.162
29	11817	07	2.646
30	12629	18	4.243
31	13310	16	4.000
32	14820	13	3.605
33	16691	03	1.732
34	19144	10	3.162

Figure 5.3: Decay table of front KI target

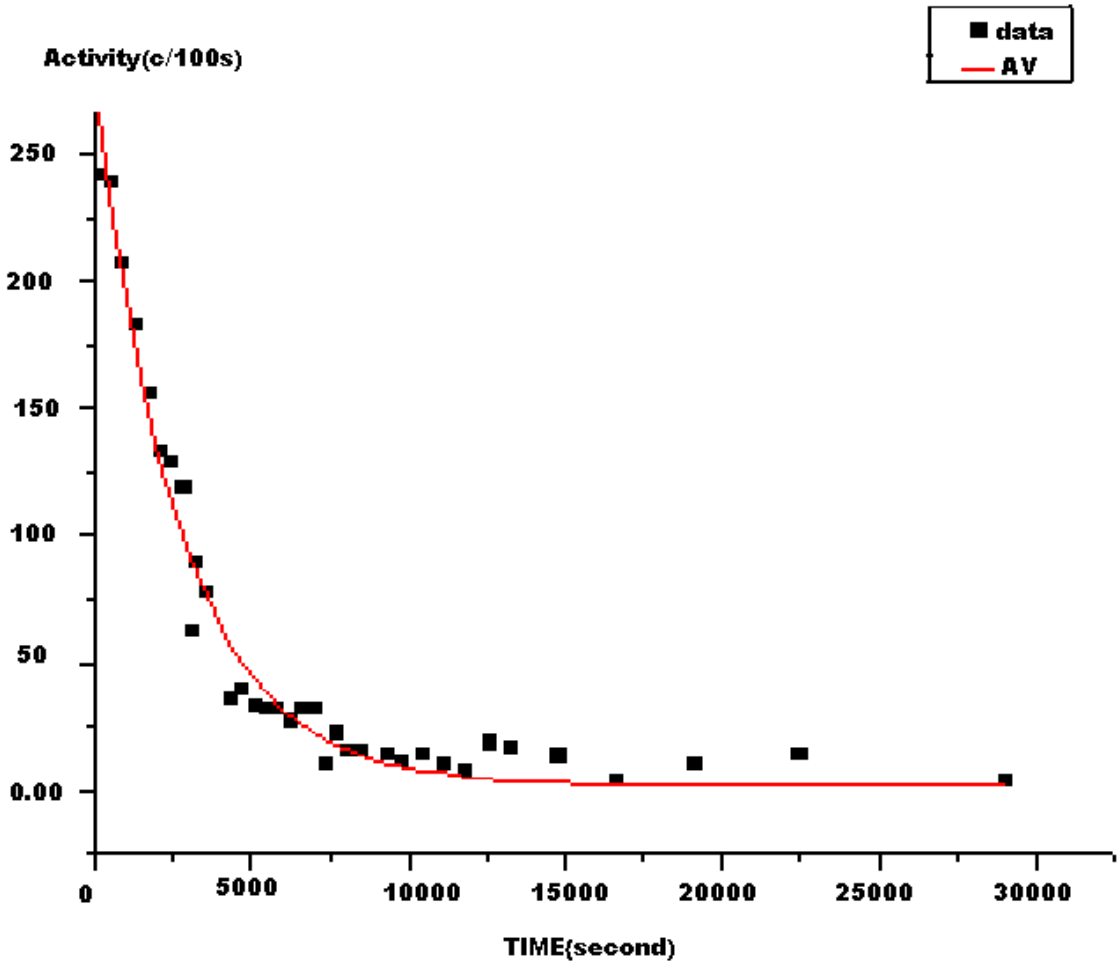


Figure 5.4: Exponential decay curve of front KI target

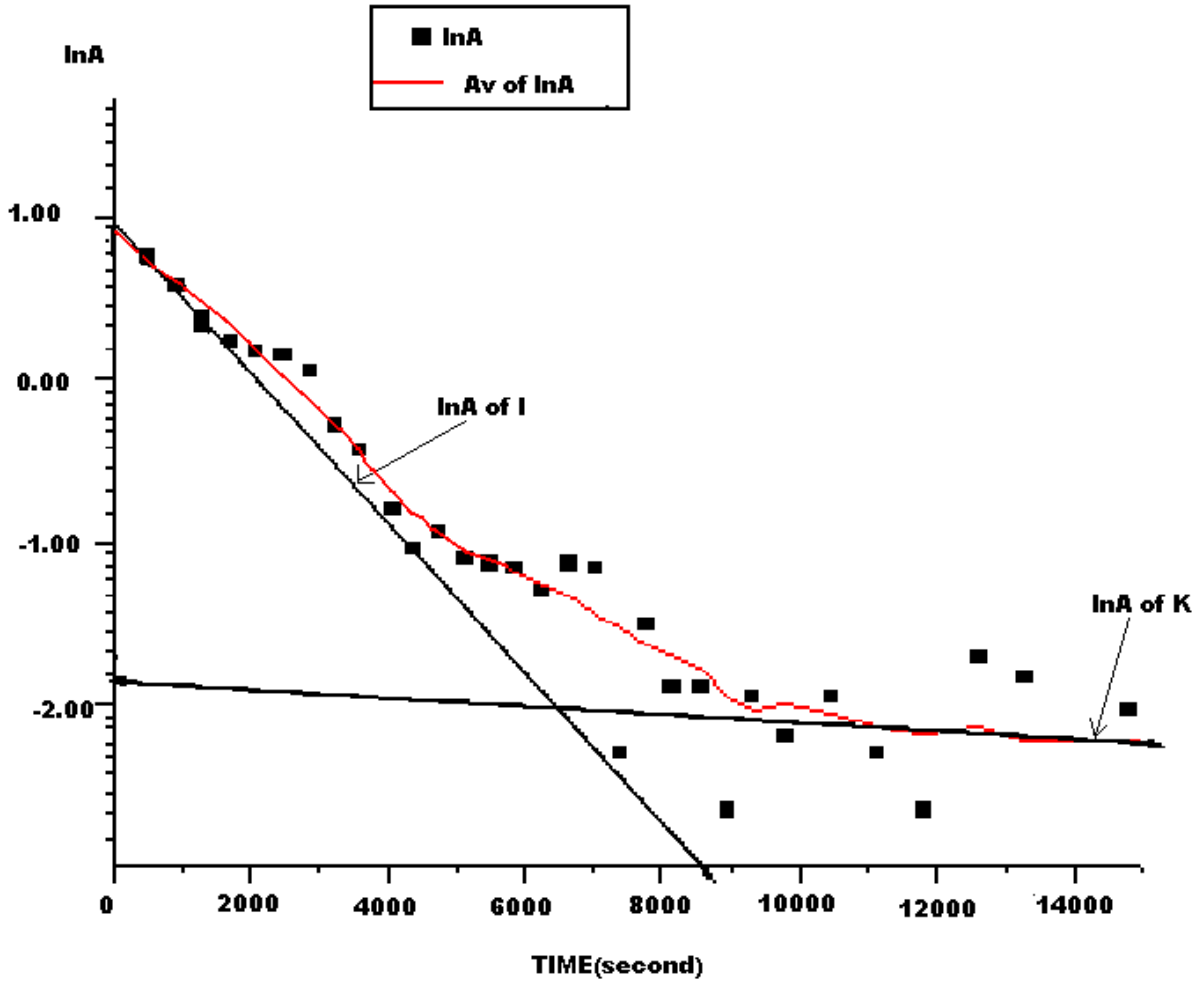


Figure 5.5: Logarithmic decay curve of front KI target

NO	TIME(SEC)	COUNT/100SEC	±√N
1	100	281	16.763
2	505	216	14.696
3	901	216	14.696
4	1289	208	14.422
5	1691	130	11.401
6	2069	109	10.440
7	2451	101	10.049
8	2824	91	9.539
9	3197	71	8.426
10	3573	52	7.211
11	3949	47	6.865
12	4326	44	6.633
13	4708	42	6.481
14	5082	30	5.477
15	5457	23	4.796
16	5828	20	4.472
17	6221	27	5.196
18	6606	30	5.477
19	6986	21	5.482
20	7359	14	3.742
21	7739	12	3.464
23	8119	15	3.873
24	8532	08	2.828
25	8920	15	3.873
26	9298	12	3.464
27	9866	09	3.000
28	10532	10	3.162
29	11198	08	2.828
30	11863	24	4.898
31	12795	09	3.000
32	14081	17	4.123
33	15706	12	3.464
34	17700	15	3.873
35	20655	01	1.000
36	24924	06	2.449
37	32031	11	3.316
38	40840	14	3.742
39	51492	07	2.646
40	66190	01	1.000
41	75065	06	2.449

Figure 5.6: Decay table of back KI target

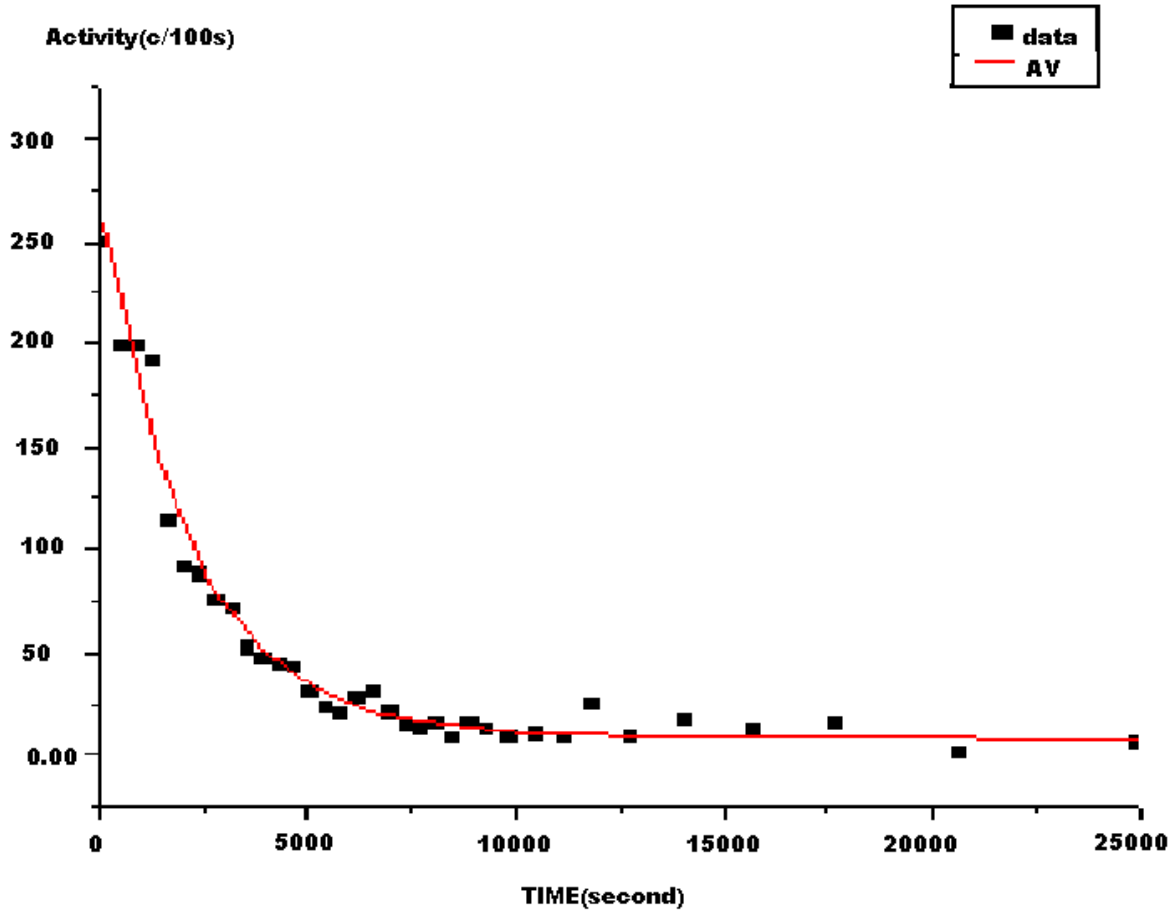


Figure 5.7: Exponential decay curve of back KI target

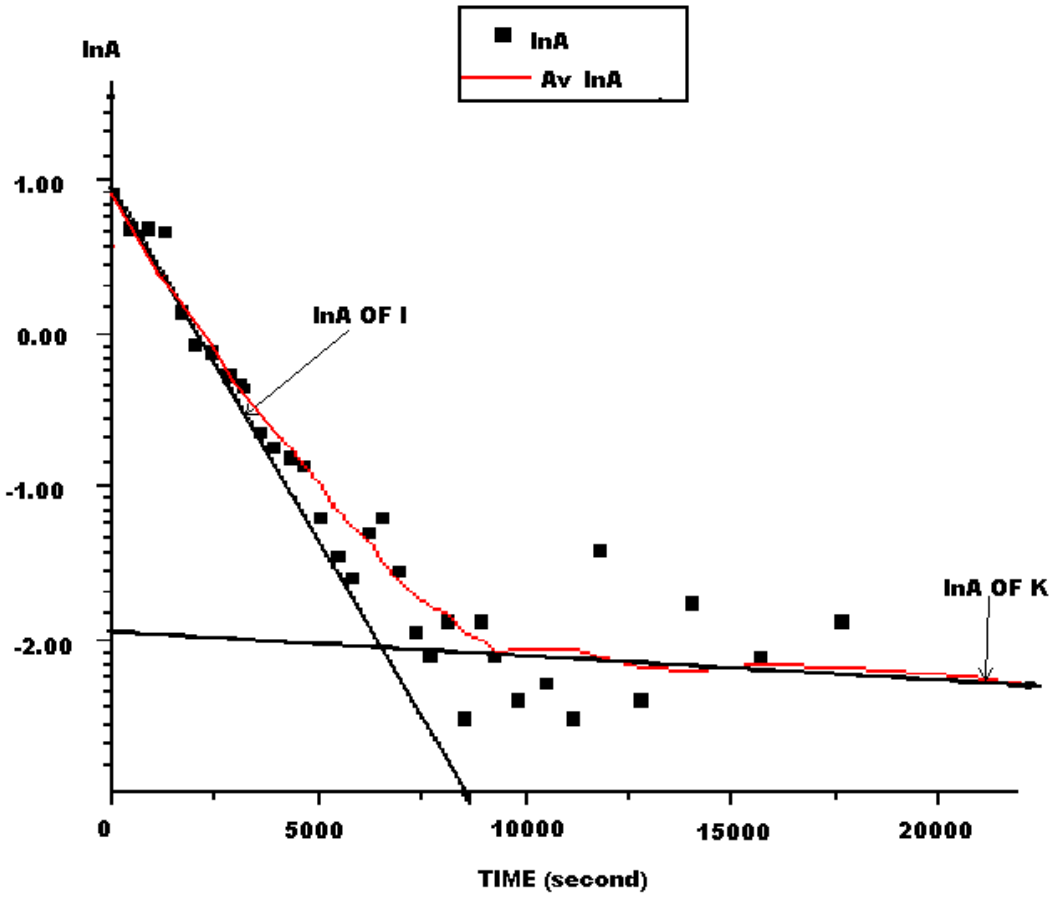


Figure 5.8: Logarithmic decay curve of back KI target

NO	Time(sec)	Count/100sec	± √N	NO	Time(sec)	Count/100sec	± √N
1	100	451	21.23	34	127548	159	12.61
2	901	404	20.09	35	147276	134	11.57
3	1683	408	20.19	36	157106	128	11.31
4	2440	396	19.89	37	167936	114	10.67
5	3190	348	18.65	38	178419	101	10.05
6	3943	369	19.21	39	190698	99	9.95
7	4702	392	19.79	40	20265	98	9.89
8	5448	365	19.10	41	217766	87	9.33
9	6215	361	19.00	42	225869	74	8.60
10	6979	405	20.12	43	237506	64	8.00
11	7732	388	19.69	44	248264	60	7.74
12	8535	379	19.46	45	260555	58	7.61
13	9291	370	19.23	46	275306	50	7.07
14	9955	373	19.31	47	298106	43	6.56
15	10622	363	19.05	48	313452	34	5.83
16	11286	338	18.38	49	322465	33	5.74
17	11950	346	18.44	50	333766	32	5.66
18	13018	365	19.10	51	345746	30	5.48
19	14408	378	19.44	52	356211	25	5.00
20	16154	361	19.00	53	368116	22	4.69
21	17930	304	17.43	54	375916	29	5.38
22	21002	311	17.63	55	386807	26	5.09
23	26114	306	17.49	56	397891	20	4.47
24	11834	331	18.19	57	408937	19	4.36
25	43388	289	17.00	58	428937	19	4.36
26	54706	273	16.52	59	439038	09	3.00
27	66766	236	15.36	60	458662	16	4.00
28	74574	250	15.81	61	467857	13	3.61
29	79764	220	14.83	62	482334	10	3.16
30	87907	200	14.14	63	503589	08	2.83
31	99274	187	13.67	64	540777	05	2.24
32	105838	190	13.78	65	583097	01	1.00
33	116857	162	12.73		—	—	—

AV.BG =37C/100S

Figure 5.9: Decay table of As-76

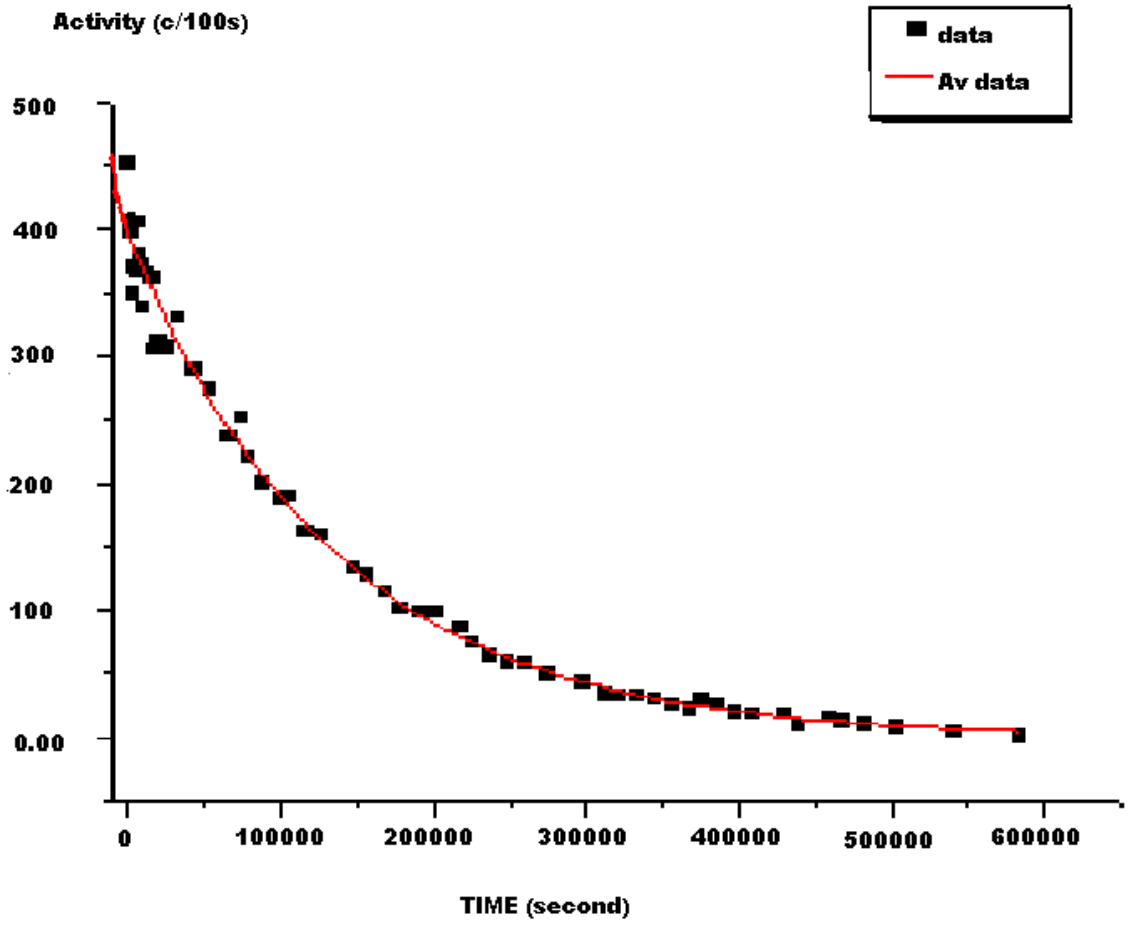


Figure 5.10: Exponential decay curve of As-76

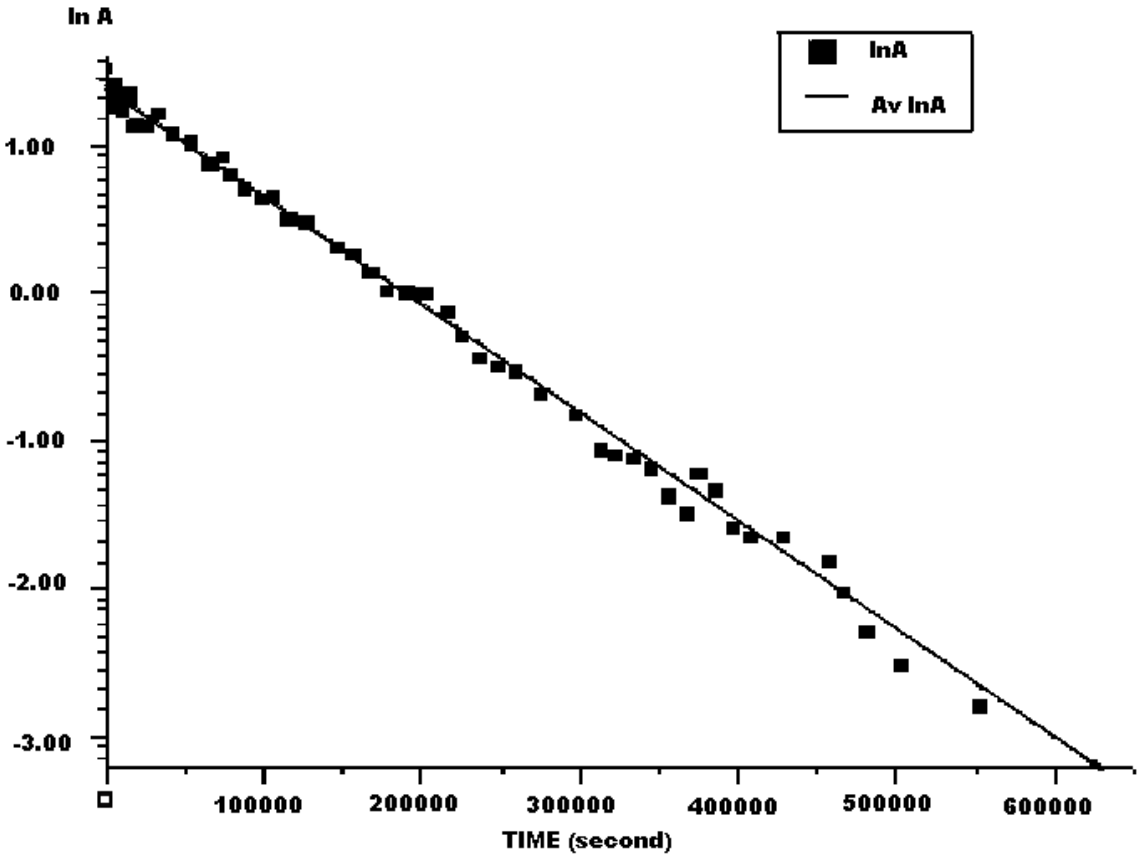


Figure 5.11: Logarithmic decay curve of As-76

5.3 Results and Discussions

5.3.1 Decay constant determination for the front and back I target for first experiment

From the logarithmic curve presented in fig(5.5) for the front KI target, the decay constant of I-127 can be determined by finding the slope of the straight line curve which shows the logarithmic decay curve of iodine.

$$\lambda_1 = \frac{4.21659}{9149.659} = 4.618 \times 10^{-4} s^{-1} \tag{5.3.1}$$

From the logarithmic graph for the back I target,fig(5.8), the decay constant or the half-life of I-127 can be calculated by finding the slop of the straight line,which represent the logarithmic decay of its activity.

$$\lambda_2 = \frac{4.089633044}{8857.14286} = 4.62 \times 10^{-4} s^{-1}. \tag{5.3.2}$$

5.3.2 Neutron flux determination

The incident neutron flux used to activate the target elements may be calculated from the equation[6];

$$\Phi = \frac{n e^{\lambda t_d}}{N \eta \sigma K_D} \tag{5.3.3}$$

where

$$K_D = (1 - e^{-\lambda t_{irr}})(1 - e^{-\lambda t_c}) \tag{5.3.4}$$

For large irradiation time K_D reduced to ;

$$K_D = (1 - e^{-\lambda t_c}) \tag{5.3.5}$$

The efficiency of GM-Counter,i.e, fraction of pulses registered by the counter can be calculated by;

$$\eta = \frac{1}{100}[ae^{-\mu_1 d} + be^{-\mu_2 d} + ce^{-\mu_3 d} + ...] \tag{5.3.6}$$

where d represent the sum of the thickness of beta absorbers in the counter.

$$d = d_1 + d_2 + d_3 \tag{5.3.7}$$

d_1 is the thickness of the tape.

d_2 is the thickness of the GM- Counter window.

d_3 is the half thickness of the sample.

The constants a,b,c,... represent the branching ratio of the respected beta particles and the constant μ,the mass absorbution coefficient of the betas, can be;

$$\mu = 17[E]^{-1.14}[23] \tag{5.3.8}$$

E is the endpoint energy of beta particles in terms of Mev. For the front I target, the above constants become;

$$d_1 = 2.5mg/cm^2 \tag{5.3.9}$$

$$d_2 = 2.0mg/cm^2 \tag{5.3.10}$$

$$d_3 = 0.126g/cm^2 \tag{5.3.11}$$

$$d = 0.1305g/Cm^2 \tag{5.3.12}$$

$$\mu_1 = 17[2.12]^{-1.14} = 7.218cm^2/gm \tag{5.3.13}$$

$$\mu_2 = 17[1.665]^{-1.14} = 9.509 cm^2/gm \tag{5.3.14}$$

$$\mu_3 = 17[1.125]^{-1.14} = 14.683 cm^2/gm \tag{5.3.15}$$

Then putting these values in the eq(5.3.6),we can have;

$$\eta = \frac{1}{100}[29.544 + 6.83 + 0.289] \tag{5.3.16}$$

$$\eta = 0.367 \tag{5.3.17}$$

The other parameters in the flux equation has the following values.

$$N_1 = [\frac{127 g/mole}{166 g/mole}] \times [\frac{0.1985 g}{166 g/mole}][\times 6.02 \times 10^{23} atoms/mole] = 5.51 \times 10^{20} atoms \tag{5.3.18}$$

$$n_1 = 2.46 count/second \tag{5.3.19}$$

from fig (5.3)(the number of count at t_d=0)

$$\sigma = 6.2 \times 10^{-28} m^2 \tag{5.3.20}$$

$$K_D = 4.5 \times 10^{-2} \tag{5.3.21}$$

The value of neutron flux incident is found to be;

$$\Phi_1 = \frac{2.46 c/s}{0.367 \times 6.2 \times 10^{-28} m^2 \times 5.51 \times 10^{20} atoms} \tag{5.3.22}$$

Where the value of K_D is one at t_d=0[24]

$$\Phi_1 = 1.9629513 \times 10^7 n/m^2 s \tag{5.3.23}$$

The flux in the back I target can be found using the same procedure.All the rest parameters are the same except; d, n,η and N.the value of this parameters can be calculated as,

$$d = d_1 + d_2 + d_3 \tag{5.3.24}$$

$$d_1 = 2.5mg/cm^2 \tag{5.3.25}$$

$$d_2 = 2.0mg/cm^2 \tag{5.3.26}$$

$$d_3 = 0.1456g/cm^2 \tag{5.3.27}$$

$$d = 2.5mg/cm^2 + 2.0mg/cm^2 + 0.1456g/cm^2 \tag{5.3.28}$$

$$d = 0.15og/cm^2 \tag{5.3.29}$$

Putting the value of d of the second sample and μ's in the eq(5.3.6), we can get the value of η for the second sample.

$$\eta = \frac{1}{100}[25.739 + 5.744 + 0.215] \tag{5.3.30}$$

$$\eta = 0.317 \tag{5.3.31}$$

The rest parameters can be;

$$n_2 = 2.52count/second \tag{5.3.32}$$

from fig (5.6)

$$N_2 = [\frac{127g/mole \times 0.2287g}{166g/mole}] \times [\frac{6.023 \times 10^{23} atom/mole}{166g/mole}] = 6.348 \times 10^{20} atoms \tag{5.3.33}$$

$$K_D = 4.5 \times 10^{-2} \tag{5.3.34}$$

$$\sigma = 6.2 \times 10^{-28} m^2 \tag{5.3.35}$$

Using the given values in the above, the neutron flux in the second I target becomes;

$$\Phi_2 = \frac{2.52c/s}{0.317 \times 6.2 \times 10^{-28} m^2 \times 6.348 \times 10^{20} atoms} \tag{5.3.36}$$

where the value of K_D becomes one at t_D=0[24]

$$\Phi_2 = 2.0198200 \times 10^7 n/m^2 s \tag{5.3.37}$$

The average neutron flux captured by the two iodine samples calculated as;

$$\Phi_{ave} = \frac{\Phi_1 + \Phi_2}{2} \tag{5.3.38}$$

then,

$$\Phi_{ave} = 1.993856 \times 10^7 n/m^2.s \tag{5.3.39}$$

5.3.3 Neutron Capture Cross-Section of ^{75}As

To evaluate the capture cross-section of As-75 by thermal neutron flux already determined, we have to use the equation of flux,eq(5.3.3).

$$\sigma = \frac{ne^{\lambda t_d}}{N.\eta.\Phi_{ave}.K_D} \tag{5.3.40}$$

The half-life of ^{76}As can be determined by finding the slope of logarithmic decay curve of As-76.i.e,

$$\lambda_3 = \frac{4.614528753}{631428.571} s^{-1} \tag{5.3.41}$$

$$\lambda_3 = 7.308 \times 10^{-6} s^{-1} \tag{5.3.42}$$

$$T_{\frac{1}{2}} = \frac{0.693}{\lambda_3} \tag{5.3.43}$$

$$T_{\frac{1}{2}} = 26.34 hour. \tag{5.3.44}$$

The percentage error as compared to the earlier value is 0.076 percent

$$N = \frac{m}{M} N_A \tag{5.3.45}$$

$$N_3 = \frac{0.1444g}{75g/mole} \times 6.023 \times 10^{23} atoms/mole. \tag{5.3.46}$$

$$N_3 == 11.596 \times 10^{20} atoms. \tag{5.3.47}$$

$$\Phi_{ave} = 1.993856 \times 10^7 n/m^2.s \tag{5.3.48}$$

$$n_3 = 5.01 count/second \tag{5.3.49}$$

from fig (5.9)

$$K_D = 7.305 \times 10^{-4} \tag{5.3.50}$$

The value of η has to be determined for the As-76 sample.As-76 decay from its ground level to the ground level of Se-76 by emitting 17 types of beta particles with different branching ratio when it is activated by thermal neutrons[26],but the following higher end point energy beta particles has an effect in the absorbtion and counting[21].

$$\beta_1 \rightarrow 1.177 Mev \rightarrow 5.1 pecent \tag{5.3.51}$$

$$\beta_2 \rightarrow 1.748 Mev \rightarrow 6.9 pecent \tag{5.3.52}$$

$$\beta_3 \rightarrow 2.405 Mev \rightarrow 35 pecent \tag{5.3.53}$$

$$\beta_4 \rightarrow 2.965 Mev \rightarrow 53 pecent \tag{5.3.54}$$

The possible mass absorbtion coefficient of these beta particles can be;

$$\mu_1 = 17(1.177)^{-1.14} = 14.117 cm^2/gm \tag{5.3.55}$$

$$\mu_2 = 17(1.748)^{-1.14} = 8.994 cm^2/gm \tag{5.3.56}$$

$$\mu_3 = 17(2.408)^{-1.14} = 6.251 cm^2/gm \tag{5.3.57}$$

$$\mu_4 = 17(2.965)^{-1.14} = 4.924 cm^2/gm \tag{5.3.58}$$

The total thickness of the beta radiation absorbers may be calculated as;

$$d = d_1 + d_2 + d_3 \tag{5.3.59}$$

where $d_1 = 2.5 mg/cm^2$ tape thickness

$d_2 = 0.0919 g/cm^2$ sample thickness

$d_3 = 0.20 mg/cm^2$ GM-window thickness

then $d = 0.0965 g/cm^2$

The counting efficiency of the GM-Counter of these beta particles becomes;

$$\eta = \frac{1}{100}[ae^{-\mu_1 d} + be^{-\mu_2 d} + ce^{-\mu_3 d} + fe^{-\mu_4 d}] \tag{5.3.60}$$

where a,b,c and f are the branching ratio of the beta particles respectively.

$$\eta = \frac{1}{100}[1.306 + 2.897 + 19.1466 + 32.954] \tag{5.3.61}$$

$$\eta = 0.563 \tag{5.3.62}$$

Having the above calculated parameters we can evaluate the capture cross-section of As-75.

$$\sigma = \frac{5.01c/s}{0.563 \times 11.596 \times 10^{20} atoms \times 1.9913856 \times 10^{7} n/m^2.s} \tag{5.3.63}$$

$$\sigma = 3.854 \times 10^{-28} m^2 \tag{5.3.64}$$

$$\sigma = 3.854 barn. \tag{5.3.65}$$

The value that we get for thermal neutron capture cross section of 75As less than the value presented in the literature which is 4.23±0.1[22] .

5.3.4 Determination of Neutron Flux and Capture Cross Section of As-75 for the Second Experiment

In this experiment we will try to present the available data taken and the results of calculations only, Because, we followed the same procedure to find the cross section.

For the Back KI

1,Mass of back KI target = 0.144g

2,Background radiation = 15.33dps

3,The number of target nuclei of back I target,N_1 = 3.997×10^{20}atoms

4,Capture cross section of I-127 = 6.2barn

5,Decay constant of back I target, $\lambda_1 = 4.62\times10^{-4}$ s^{-1}

6,Thickness of the back KI sample, d = 0.0962g/cm^2

7,Irradiation time ,t_{irr} = 873432second

8,Counting time, t_c = 100second

9,The detection efficiency of the GM-Tube for the back I target,$\eta_1 = 0.448$

10,The number of count for the back I target when $t_d = 0$,n_1 = 1.5count/second

11,Delay time is 180 second

Using the above measured and calculated data, we can calculate the flux of the incident neutron flux in the back I target.i.e,

$$\Phi_1 = \frac{n_1 e^{\lambda_1 t_d}}{\eta_1.\sigma.N_1[1 - e^{-\lambda_1 t_{irr}}].[1 - e^{-\lambda_1 t_c}]} \tag{5.3.66}$$

at $t_d = 0$

$$\Phi_1 = \frac{1.5c/s}{0.448 \times 6.2 \times 10^{-28} m^2 \times 3.997 \times 10^{20} atoms} \tag{5.3.67}$$

Applying the above gives in the flux equation we can get;

$$\Phi_1 = 1.3511 \times 10^7 n/m^2.s \tag{5.3.68}$$

For the Front KI

1,Mass of the second KI target = 0.1286g

2,Background radiation = 15.33dps

3,The number of target nuclei of front I target,N_2 = 3.5697×10^{20}atoms

4,Capture cross section of I-127 = 6.2barns

5,Thickness of the second KI sample,d = 0.0864g/cm^2

6,Irradiation time ,t_{irr} = 873432second

7,counting time, t_c = 100second

8,The detection efficiency of the GM-Tube for the front I target,$\eta_2 = 0.483$

9,The number of count for the front I target when $t_d = 0$,$n_2 = 1.8$count per second.

10,Decay constant of front I target, $\lambda_2 = 4.62\times10^{-4}s^{-1}$

11,Delay time is 453second

Using the data for the front I target, we can find the neutron flux.

$$\Phi_2 = \frac{n_2 e^{\lambda_2 t_d}}{\eta_2.\sigma.N_2[1 - e^{-\lambda_2 t_{irr}}].[1 - e^{-\lambda_2 t_c}]} \tag{5.3.69}$$

$$\Phi_2 = \frac{1.8c/s}{0.483 \times 6.2 \times 10^{-28}m^2 \times 3.5697 \times 10^{20}atoms} \tag{5.3.70}$$

$$\Phi_2 = 1.6838444 \times 10^7 n/m^2.s \tag{5.3.71}$$

The average flux can be;

$$\Phi_{ave} = \frac{\Phi_2 + \Phi_1}{2} \tag{5.3.72}$$

$$\Phi_{ave} = 1.5174722 \times 10^7 n/m^2.s \tag{5.3.73}$$

For As-75 Target

1,Mass of the As-75 target = 0.0665gm

2,Background radiation = 15.33dps

3,The number of target nuclei,N_3 =5.34×10^{20}atoms

4,The average flux in the sample,Φ_{ave} =1.5174722×10^7n/m^2.s

5,Thickness of the sample,d = 0.0468g/cm^2

6,Irradiation time ,t_{irr} = 873432second

7,Counting time, t_c = 100second

8,The absorbtion efficiency of the GM-Tube,$\eta_3 = 0.754$

9,The number of count for the back I target when t_d =0,$n_3 = 2.33$count/sec

10,Decay constant of the target , $\lambda_3 = 7.308\times10^{-6}$ s^{-1}

The capture cross section of the As-75 target will be ;

$$\sigma = \frac{n_3 e^{\lambda_3 t_d}}{\eta_3.\Phi_{ave}.N_3[1-e^{-\lambda_3 t_{irr}}].[1-e^{-\lambda_3 t_c}]} \tag{5.3.74}$$

$$\sigma = \frac{2.33c/s}{0.754 \times 1.5174722 \times 10^7 n/m^2.s \times 5.34 \times 10^{20} atoms} \tag{5.3.75}$$

$$\sigma = 3.813 barn \tag{5.3.76}$$

The previously determined capture cross section of As-75, in the literature is found to be 4.23±0.1barns[22].the value of capture cross section in this experiment is 3.854barns and 3.813barns respectively.The average cross section can be;

$$\sigma_{ave} = (\frac{3.854+3.813}{2}) barn \tag{5.3.77}$$

$$\sigma_{ave} = 3.834 \pm 0.021 barn \tag{5.3.78}$$

The total error in the experiment can be;

$$Error(^0/_0) = \frac{previousvalue - experimentalvalue}{previousvalue} \times 100(^0/_0) \tag{5.3.79}$$

$$Error(^0/_0) = \frac{4.23-3.834}{4.23} \tag{5.3.80}$$

$$Error(^0/_0) = 9.36^0/_0 \tag{5.3.81}$$

5.4 Sources of Error Estimation

In this experiment the measurement consists of counting the number of β^- emitted in the radioactive decay of KI and As-76 samples during a definite time interval(t_c).There are two types of errors expected in the experiment.These are;

5.4.1 Errors in the Measurement

The possible source of systematic and random source of errors in the measurement process contains;

1,In the measuring time interval.

2,In the efficiency of Geiger tube and counter.

3,In the measurement of masses of samples.

4,The background radiation (the counter is not shielded).

5,The orientation of activated samples in the counter window.

6,In the measurement of the thickness of samples and sticky tape

7, personal errors in the rounding of the calculated results.

8,The capture cross section of As-75 in this experiment is slightly less than the value found in the literature the reason may be that in our neutron source there are about 4percent epithermal neutrons[25].

5.4.2 Errors due to Random Nature of Decay Process

This is the randomness of the decay process or in the decay data distribution that has noting to do with the measurement process.There exist for any radioactive substance a certain probability that any particular nucleus will emit radiation within a given time interval .This is the same for all nuclei of the same type .Here,we can not predict the time at which an individual nucleus will decay. However,when there is a large

number of disintegration takes place, There is a definite average decay rate which is the characteristic for a particular nuclei type, but the actual number of decaying in any specific interval may vary significantly from the average value.Thus, there may not be correct answer to which we can compare our experimental findings.We can emphasis that this errors can not be removed by improving the precession of the experiment but it can be improved only at the expense of significantly increasing the time duration of the counting .Such kind of error may contribute at about 10percent[15].

Generally we note that, the random error in the measurement and randomness in the decay phenomenon being measured added together and gives total experimental error.

5.5 Summary and Conclusion

Instrumental Neutron activation Analysis using Am-Be neutron source has profound effect in identifying the type of given element of sample.Here in this work, the half life and capture cross section of the sample As-76 is determined.The relative error in these measurement were taken as 9.36percent.

The result of this work suggests that, using Instrumental Neutron Activation Analysis technique it is possible to perform environmental radio-analysis with an improved counter shielding and good precision of measurement. This may very important to elemental analysis of a given sample of interest in the fields of like,medicine, forensic, mining, industry and any other applications.

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Declaration

This thesis is my original work, has not been presented for a degree in any other University and that all the sources of material used for the thesis have been dully acknowledged.

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