Chapter (1)

Introduction

The use of ionizing radiation in different applications has been expanded now a day. Radiation processing applications are one of the techniques being used in food irradiation and preservation, polymers cross-linking, medical devices sterilization etc..

There are tow types of ionizing radiation used in radiation processing, gamma irradiation from gamma sources and electron beam from electron accelerators.

The Gammacell 220 Excel is a cobalt 60 irradiation facility for research purposes has been used in this work with activity 10038Ci.

A C0-60 emits photons with energies of approximately 1.17 and 1.33 MeV in nearly equal proportions. It has a half-life of 5.261 years.

In radiation processing accurate estimation of radiation dose absorbed by the product is extremely important, as absorbed radiation is the only parameter that needs to be controlled in the process. Therefore, dosimetry plays a vital role in setting of process parameter to meet variety of specifications, dose mapping in products, carrying out validation, commissioning procedure as well as in the day-to-day operation of the plant. The dose delivered to the material/product should be high enough to ensure the intended radiation effect but must not exceed the level necessary for reasons of time, efficiency and uniformity in the performance of the processed product. The reliability of dosimetry is now so well recognized, that the use of test procedure are reduced to the minimum, or often, even dispensed with on a routine basis. (1)

The choice of the dosimetry technique generally depends on the specific properties and needs of the particular irradiation facility, such as geometry factors, magnitude of the dose-rate, homogeneity and spectral distribution of the radiation field on one side, and on the properties of the material to be irradiated, on the other side.

The dosimetry system used for calibration and routine monitoring of various radiation-processing facilities is better to be chemical dosimetry.

A chemical dosimetry is a system that measure the energy absorbed from ionizing radiation by virtue of chemical changes produced in it when it is exposed to ionizing radiation. In all chemical dosimeters radiation induced chemical reaction produces

at least one, initially absent species, which is long lives enough to determine its quantity or the change in the properties of the system. Fricke and Alanine dosimeters can be used as chemical dosimeters to measure the absorbed dose in the Gammacell.

The classification of dosimetry put the Fricke dosimeter as a reference dosimetry cause of the high metrological quality that can be used to calibrate other dosimeters while Alanine dosimeter classified as a routine dosimetry calibrated against reference dosimeter and use for dose mapping and for process monitoring for quality control.

In this work the Fricke dosimeter was used as reference dosimetry, to calibrate the Gammacell 220 E irradiator. Fricke can be used for doses up to 400 Gy, while for high doses calibration Alanine dosimeter was used for the first time in this Lab. It gives linear dependence with the dose up to 5000 Gy.

To check the dose distribution inside the irradiation chamber of the Gammacell, both Fricke and Alanine were used for the dose mapping. The results showed the homogenous of the dose inside the chamber.

Chapter (2)

BASIC PRINCIPLES OF DOSIMETRY

2.1 Ionizing radiation

The term radiation is very broad, and includes such things as radio waves, microwaves, visible light, ultra-violet and x-rays. Ionizing radiation has sufficient energy to remove electrons from atoms when it passes through a material, thereby causing the atoms to become electrically charged or ionized .In some circumstances, the presence of such ions in living tissues can disrupt normal biological processes. Excessive exposure to ionizing radiation may result in sickness or death.

The constantly exposed from natural source of ionizing radiation, High-energy ionizing radiations called cosmic rays penetrate the earth's atmosphere from the sun and outer space. Naturally occurring radioactive materials are present in the earth, in the houses where we live, in our own bodies and in the buildings in which we work. The food we eat, the water we drink, and the air we breathe, all contain small amounts of radioactive materials.

Ionizing radiation also may be exposed from man-made sources. The most common man-made sources of exposure are x-rays, radioisotopes used for medical diagnosis and treatment.

The ionizing radiation emitted by the decay of unstable atoms of cobalt 60 is called Beta and Gamma radiation. The gamma radiation emitted by cobalt 60 is very penetrating; therefore the cobalt 60 sources must be surrounded by shielding material to reduce the radiation levels to acceptable levels at any accessible location. (1)

Ionizing radiation refers to particles that can produce ionization in the medium. The ionizing radiations, which are encountered in the study of radiation effects, can be classified in to two types.

- **Charged particles – such as α, β, protons and electron**s.
- **Uncharged particles – such as X – rays, γ-rays, and neutrons**.

Ionization produced by the particles is the process by which one or more electrons are liberated in collisions of the particles with atoms or molecules. This may be distinguished from excitation, which is a transfer of electrons to higher energy levels in atoms or molecules and generally requires less energy. When charged particles have slowed down sufficiently, ionization becomes less

likely or impossible and the particles increasingly dissipate their remaining energy in other processes such as excitation or elastic scattering. Thus near the end of their range, charged particles that were ionizing becomes non-ionizing.
The charged particles produce ionization directly through coulomb interactions with atomic electrons. The uncharged particles produce ionization through knocking of some charged particles in the medium. The neutrons produce nuclear reactions at all energies. Generally, the energy of ionizing radiation is expressed in multiples of electron volt, e.g. keV, MeV. (3)

2.2 Interactions of photons with matter

The electromagnetic radiation (photons) produces ionization through knocking of some charged particles in the medium. An X-ray or gamma ray photon is uncharged and creates no direct ionization or excitation of the material through which it passes. The detection of gamma ray is therefore critically dependent on causing the gamma-ray photon to undergo an interaction that transfers all or part of the photon energy to an electron in the absorbing material. There are three main processes, which describe the interaction of gamma rays with matter. They are (1) **photoelectric effect**, (2) **Compton Effect** and (3) **pair production**. (3)

2.2.1 Photoelectric effect

In photoelectric process, a photon gives up all its energy (E) to a bound electron which is then ejected from the atom with a kinetic energy (E_1) equal to the energy of the photon less the binding energy (E_b) of the electron ($E_1 = E - E_b$) fig(1.1). The probability of ejection of an electron is maximum when the photon energy is just higher than the binding energy of the electron. The photoelectric cross-section varies with energy approximately as $1/E^3$ and mass absorption coefficient varies approximately as Z^3 where Z is the atomic number of the medium. Therefore the photoelectric effect increases with increasing atomic number of the absorber and with decreasing energy of the gamma rays. The photoelectric effect is a true absorption process. (3)

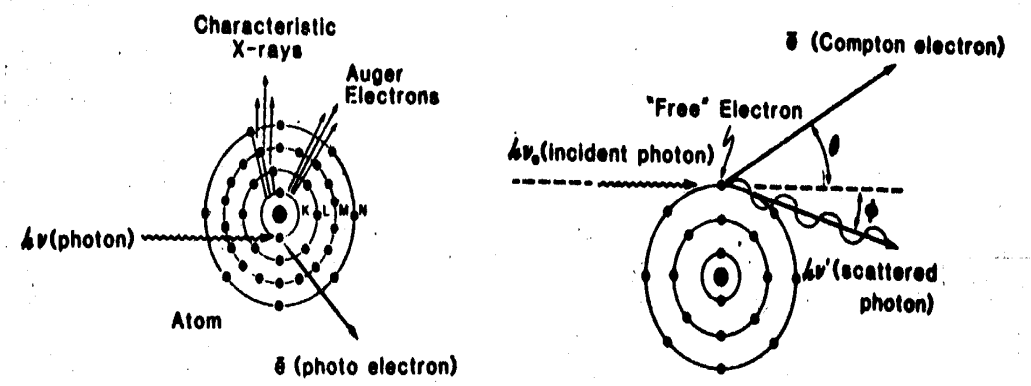


Fig. (2.1) Photoelectric process and Compton Effect

2.2.2 Compton effect

Compton Effect involves collision of a photon with an electron, which is regarded as free. The photon transfers some of its energy to the electron, which recoils, and the remainder of the energy appears as the energy of a scattered photon fig (1.1). Since most of the materials (except for hydrogen) have nearly equal number of electrons per gram, it follows that absorption by this process is nearly equal for all materials. The total Compton cross-section varies very little at low energies of the photons and falls off at high energies. In Compton process, there is merely a decrease in the photon energy, the extent of this decrease being greater the larger both the initial energy and the scattering angle. (3)

2.2.3. Pair production

Pair production is the conversion of a photon into a pair of positive and negative electrons in a nuclear field fig(1.2). Since the creation of the pair requires a minimum energy of 1.02 MeV (which is twice the rest mass energy of the electron). Unless the photon posses at least 1.02 MeV, the process will not take place. Energy in excess of 1.02 MeV is shared equally in the form of kinetic energy of the pair formed. The Positive electron having lost its kinetic energy while traversing through the medium combines with a negative electron giving rise to annihilation radiation normally in the form of to photons each with 0.51 MeV moving in opposite directions. The pair production increases rapidly with atomic number as Z^2 per atom. (3)

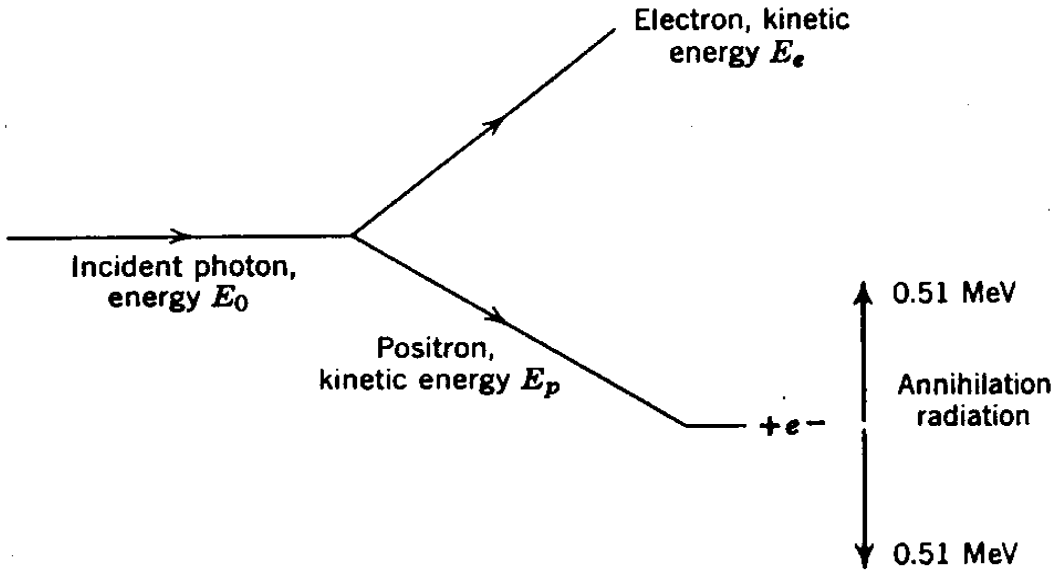


Fig. (2.2) Pair production

All these processes lead to the dissipation of energy by ionizing radiation in the system exposed. While traversing matter, electromagnetic radiation may undergo all the three processes in varying degrees. The dominating process depends on the energy of the radiation and the nature of the medium. A probability of occurrence is attributed to each process, which is referred to as photoelectric, Compton or pair production attenuation coefficients. The total attenuation coefficient is the sum of all the three coefficients. (3)

- At low energies, the total mass attenuation coefficient is made up of photoelectric process.
- From 100 keV to 10 MeV, the major contribution is from Compton process.
- At 1.02 MeV, pair production starts and it increases with energy.

2.3 Interactions of Charged particles with matter

Interaction between the electric fields of a charged particle and the orbital electrons of the absorbing medium leads to electronic excitation and ionization. The energy gained by the atomic electron is proportional to 1/E, where E is the energy of the charged particle. In many ionizing collisions, only one ion pair is produced. In other cases, the ejected electron may have sufficient kinetic energy to produce a small cluster of several ionizations; and in a

small proportion of the collisions the ejected electron may receive a considerable amount of energy, enough to cause it to travel a long distance and to leave a trail of ionizations. Such electron, whose kinetic energy may be of the order of 1 MeV, is called a delta ray. The number of ion pairs produced per unit path length of charged particle is called specific ionization.
Electrons interact with matter by a number of processes of which the most important are the inelastic and elastic collisions and the emission of electromagnetic radiation. The relative importance of these processes varies strongly with the energy of the incident electrons and with the nature of the absorbing material; at high energies energy is lost predominantly by radiation emission and at low energies through inelastic collisions. Elastic scattering (change in the direction of motion without conversion of kinetic energy to any other form of energy) is of greatest importance at low energies. In inelastic collisions, the charged particles lose energy though coulomb interactions with atomic electrons of the stopping material and produces ionization and excitation in the material.

2.4 Bremsstrahlung

When a fast moving charged particle passes close to a nucleus, it undergoes deflection and loses energy in the form of electromagnetic radiation. Radiation thus emitted is known as **bremsstrahlung**. The intensity of bremsstrahlung increases with the atomic number of the medium, and decreases with increase in the mass of the particle. Hence, energy loss by radiation is more important in heavy elements than in light elements and for light particles such as electrons than for heavy particles such as protons and alphas. If the speed of the particle through a medium exceeds the speed of light in the medium, radiation in the visible region known as **Cerenkov** radiation is emitted.
For an electron of energy E MeV, the ratio of the energy loss by radiation to the loss by collision is given (in the MeV range of energies) by

$$\frac{(dE/dl)_{rad}}{(dE/dl)col} \approx \frac{EZ}{1600 m_e c^2} \qquad (2.1)$$

Where, $(\frac{dE}{dl})_{el}$ is the mass collision stopping power due to collisions with electrons, $(\frac{dE}{dl})_{rad}$ is the mass radiative stopping power due to

emission of bremsstrahlung in the electric fields of atomic nuclei or atomic electrons.
For electrons, bremsstrahlung emission is negligible below 100keV but increases rapidly with increasing energy, becoming the predominant mode of energy loss at electron energy between 10 and 100MeV (the exact energy depends on the stopping material). The bremsstrahlung energy spectrum, which extends from zero to the energy of the incident electron, is the continuous x-ray spectrum. (3)

2.5 Radiation Quantities and Units

A unit is necessary for the measurement of any physical quantity. The international Commission on Radiation Units and Measurements (ICRU) reviews and updates, from time to time, the concepts related to quantities and their units in radiation physics, dosimetry and radiological protection. Important quantities related to the Radioactivity, Interaction coefficients (cross section) and dosimetry are briefly explained in the following texts.

2.6 Radioactivity

2.6.1 Activity

The activity of a radioactive material is a measure of its spontaneous disintegration. The activity, A, of an amount of a radio-nuclide in a particular energy state at a given time, is the quotient of dN by dt, where dN is the number of spontaneous nuclear transformation from that energy state in the time interval dt, thus

$$A = \frac{dN}{dt}. \quad \text{Unit: s-1} \qquad (2.2)$$

The special name for the unit of activity is Becquerel (Bq). The old unit of activity is Curie (Ci). 1 Ci = 3.7 x 10^{10} Bq

2.6.2. Decay Constant

The decay constant, λ of a radio-nuclide in a particular energy state is the quotient of dP by dt, where dP is the probability that a given nucleus undergoes a spontaneous nuclear transformation from that energy state in the time interval dt, thus

$$\lambda = \frac{dP}{dt}. \quad \text{Unit: } s^{-1} \qquad (2.3)$$

Radioactivity in a given mass of a radioisotope decreases with time in an exponential manner. This can be represented by the equation

$$A = A_0 e^{-\lambda t} \qquad (2.4)$$

Where, A_0 is the initial activity, A is the activity after a lapse of time t and λ is the decay constant (disintegration per unit time). λ is characteristic of the radioisotope.

In actual practice, the rate of decay of a radioisotope is usually given in terms of its half-life ($t_{1/2}$), the time required for one-half of the atoms originally present to decay. The disintegration constant and the half-life are related by the expression,

$$\lambda = \frac{0.693}{t_{1/2}} \qquad (2.5)$$

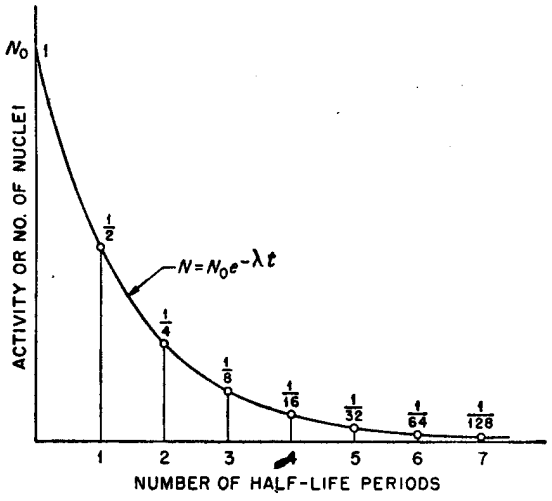


Fig. (2.3) Exponential radioactive decay

Fig (2.3) represents exponential radioactive decay in terms of half-life periods. For different radioisotopes, the half-lives are different varying from less than a second to several thousands of years. However, it should be noted that half-life of radioactive isotope does not depend upon the amount of radioactivity. (3)

2.6.3. Gamma ray sources

The radioisotope most commonly used in the making of high intensity radiation sources is cobalt-60. It is an isotope formed by thermal neutron capture in a nuclear reactor from ^{59}Co.

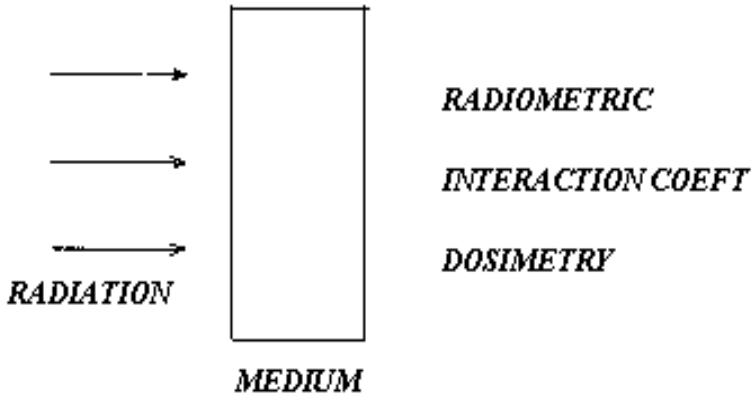
$$^{59}Co + n \rightarrow {}^{60}Co + \gamma$$
$$^{60}Co \rightarrow {}^{60}Ni + 2\gamma \quad (\beta \text{ emission}) \qquad (2.6)$$

The ^{60}Co is a beta emitter and the emission of two gamma rays from ^{60}Ni having energy of 1.33 and 1.17 MeV combined with a useful half-life of 5.27 years makes it a useful radioisotope.
Cesium 137 is another useful radioisotope, which is formed from the fission of ^{235}U in a nuclear reactor. In its decay ^{137}Cs produces one photon in the disintegration

$$^{137}Cs \rightarrow {}^{137}Ba + e- \qquad (2.7)$$

Cesium 137 is also primarily a beta emitter but its daughter product ^{137}Ba (half life 2.6 min) is a source of 0.66 MeV gamma rays. Cesium 137 has a half-life of 30 years. (3)

- Fig (2.4) RADITION AND ITS INTERACTION WITH MEDIUM



2.7 Interaction Coefficients and Related Quantities:

In an interaction, the energy or the direction (or both) of the incident particle is altered or the particle is absorbed. The interaction may be followed by the emission of one or several secondary particles. The likelihood of such interactions is

characterized ***by interaction coefficients***. They refer to a specific interaction process, type and energy of radiation, target or material.

The fundamental interaction coefficient is the ***cross section***. The **cross section**, σ, of a target entity, for a particular interaction produced by incident charged or uncharged particles, is the quotient of P by φ, where P is the probability of that interaction for a single target entity when subjected to the particle flounce, φ, thus (3)

$$\sigma = \frac{P}{\varphi} \qquad \text{Unit: } m^2 \tag{2.8}$$

2.7.1 Mass Attenuation Coefficient

The **mass attenuation coefficient**, μ/ρ, of a material, for uncharged particles, is the quotient of dN/N by ρdl, where dN/N is the fraction of particles that experience interactions in traversing a distance dl in the material of density ρ, thus

$$\frac{\mu}{\rho} = \frac{1}{\rho dl}\frac{dN}{N} \qquad \text{Unit: } m^2\ kg^{-1} \tag{2.9}$$

The linear attenuation coefficient, μ, depends on the density, ρ, of the absorber. This dependence is largely removed by using the mass attenuation coefficient, μ/ρ. The mass attenuation coefficient can be expressed in terms of the total cross section, σ. (3)

2.7.2 Mass Energy Transfer Coefficient

The **mass energy transfer coefficient**, μ_{tr}/ρ, of a material, for uncharged particles, is the quotient of dR_{tr}/R by ρdl, where dR_{tr}/R is the fraction of incident radiant energy that is transferred to kinetic energy of charged particles by interactions, in traversing a distance dl in the material of density ρ, thus (3)

$$\frac{\mu_{tr}}{\rho} = \frac{1}{\rho dl}\frac{dR_{tr}}{R} \qquad \text{Unit: } m^2 kg^{-1} \tag{2.10}$$

2.7.3 Mass Stopping Power

The **mass stopping power**, S/ρ, of a material, for charged particles, is the quotient of dE by ρdl, where dE is the energy lost by a charged particle in traversing a distance dl in the material of density ρ, thus

$$\frac{S}{P} = \frac{1}{\rho}\frac{dE}{dl}. \qquad \text{Unit: } Jm^2kg^{-1} \qquad (2.11)$$

The mass stopping power can be expressed as sum of independent components by

$$\frac{S}{\rho} = \frac{1}{\rho}(\frac{dE}{dl})_{el} + \frac{1}{\rho}(\frac{dE}{dl})_{rad} + \frac{1}{\rho}(\frac{dE}{dl})_{nuc} \qquad (1.12)$$

Where $\frac{1}{\rho}(\frac{dE}{dl})_{el}$ is the mass collision stopping power due to collisions with electrons, $\frac{1}{\rho}(\frac{dE}{dl})_{rad}$ is the mass radiative stopping power due to emission of bremsstrahlung in the electric fields of atomic nuclei or atomic electrons, $\frac{1}{\rho}(\frac{dE}{dl})_{nuc}$ is the mass nuclear stopping power due to elastic coulomb collisions in which recoil energy is imparted to atoms. (3)

2.7.4 Linear Energy Transfer (LET)

The **linear energy transfer** or **restricted linear electronic stopping power**, L_Δ, of a material, for charged particles, is the quotient of dE_Δ by dl, where dE_Δ is the energy lost by a charged particle due to electronic collisions in traversing a distance dl, minus the sum of the kinetic energies of all the electrons released with kinetic energies in excess of Δ, thus, (3)

$$L_\Delta = \frac{dE_\Delta}{dl}. \qquad \text{Unit: } Jm^{-1} \qquad (2.12)$$

2.7.5 Radiation Chemical Yield

The **radiation chemical yield**, G (x), of an entity, x, is the quotient of n (x) by ε, where n (x) is the mean amount of substance of that

entity produced, destroyed, or changed in a system by the energy imparted, ε, to the matter of that system, thus, (3)

$$G(x) = \frac{n(x)}{\varepsilon}. \quad \text{Unit: mol J}^{-1} \tag{2.13}$$

2.7.6 Mean Energy Expended in a Gas per Ion Pair Formed

The mean energy expended in a gas per ion pair formed, W, is the quotient of E by N, where N is the mean number of ion pairs formed when the initial kinetic energy E of a charged particle is completely dissipated in the gas, thus (3)

$$W = \frac{E}{N}. \qquad \text{Unit: J} \tag{2.14}$$

2.8 Dosimetry

The effects of radiation on matter depend on the radiation field, as specified by the radiometric quantities and on the interactions between radiation and matter, as characterized by the interaction quantities. Dosimetric quantities are products of radiometric quantities and interaction coefficients. Radiation interacts with matter in a series of processes in which particle energy is converted and finally deposited in matter.

2.8.1 Kerma

The quantity kerma relates to the kinetic energy of the charged particles liberated by uncharged particles. The kerma, K, is the quotient of dE_{tr} by dm, where dE_{tr} is the sum of the initial kinetic energies of all the charged particles liberated by uncharged particles in a mass dm of material, thus (3)

$$K = \frac{dE_{tr}}{dm}. \qquad \text{Unit: J kg}^{-1} \tag{2.15}$$

The special name for the unit of kerma is **gray (Gy).**

2.8.2 Kerma Rate

The **kerma rate**, K, is the quotient of dK by dt, where dK is the increment of kerma in the time interval dt, thus

$$K = \frac{dK}{dt} \qquad \text{Unit: J kg}^{-1}\text{ s}^{-1} \qquad (2.16)$$

If the special name gray is used, the unit of kerma rate is **gray per second** (Gy s^{-1}). When the reference material is air, the quantity is called air kerma. (3)

2.8.3 Exposure

The exposure, X, is the quotient of dQ by dm, where dQ is the absolute value of the total charge of the ions of one sign produced in air when all the electrons and positrons liberated or created by photons in air of mass dm are completely stopped in air, thus

$$X = \frac{dQ}{dm}. \qquad \text{Unit: C kg}^{-1} \qquad (2.17)$$

The old unit for exposure is Roentgen (R), which is defined as that amount of X, or gamma radiation, which would liberate 1(Electro-statistic unit) ESU of charge of either sign in 1 cc of air at (standard Pressure) STP. 1 R = 2.58 x 10^{-4} C kg^{-1} (air), except at very high energies, the exposure as defined above, is the ionization equivalent of the air kerma. (1 air kerma = 114 R). (3)

2.8.4 Exposure Rate

The **exposure rate**, X, is the quotient of dX by dt, where dX is the increment of exposure in the time interval dt, thus

$$X = \frac{dX}{dt}. \qquad \text{Unit: Ckg}^{-1}\text{s}^{-1} \qquad (2.18)$$

2.8.5 Cema

A quantity called cema is defined which relates to the energy lost by charged particles (e.g., electrons, protons, alpha particles) in collisions with atomic electrons. The **cema**, C, is the quotient of dE_c by dm, where dE_c is the energy lost by charged particles, except secondary electrons in electronic collisions in a mass dm of a material, thus (3)

$$C = \frac{dE_c}{dm}. \text{ Unit: J kg}^{-1} \qquad (2.19)$$

The special name of the unit of cema is gray (Gy).

2.8.6 Cema Rate

The **cema rate**, C, is the quotient of dC by dt, where dC, is the increment of cema in the time interval dt, thus

$$C = \frac{dC}{dt}. \quad \text{Unit: J kg}^{-1}\text{s}^{-1} \qquad (2.20)$$

If the special name gray is used, the unit of cema rate is gray per second (Gy s^{-1}).

2.8.7 Absorbed Dose

The **absorbed dose**, D, is the quotient of dε by dm, where dε is the mean energy imparted to matter of mass dm, thus (3)

$$D = \frac{d\bar{\varepsilon}}{dm}. \quad \text{Unit: J kg}^{-1} \qquad (2.21)$$

The special name for the unit of absorbed dose is gray (Gy).

2.8.8 Absorbed Dose Rate

The **absorbed dose rate**, D, is the quotient of dD by dt, where dD is the increment of absorbed dose in the time interval dt, thus

$$D = \frac{dD}{dt}. \quad \text{Unit: J kg}^{-1}\text{s}^{-1} \qquad (2.22)$$

If the special name gray is used, the unit of absorbed dose rate is gray per second (Gys^{-1}).

2.9 Measurement of Absorbed dose

Measurement of the energy absorbed per unit mass in a medium exposed to ionizing radiation necessitates the introduction of a dosimeter in to the medium. Dosimeter is a device that, when irradiated, exhibits a quantifiable change in some property of the device which can be related to the absorbed dose in a given material using appropriate analytical instrumentation and techniques. Different types of dosimeters like gaseous ionization chambers, thin films, solids and liquids are in use in radiation dosimetry. There is a considerable variation in their size and composition. Also, several wall materials having varying thickness are used to contain the dosimeters.
When a dosimeter is introduced in to a material to be irradiated by gamma radiation, depending on the energy of the incident radiation, it receives the dose from,

1. Photo- and Compton electrons generated in the wall material (build-up material) facing the source.
2. Electrons produced inside the dosimeter.
3. Electrons scattered in the material surrounding the dosimeter (backscatter).
4. Electrons produced by scattered gammas.
   The relative contribution by each of these components will depend upon the thickness and composition of the dosimeter and its surroundings. (3)

2.10 Electron equilibrium

Under certain conditions the maximum absorbed dose is observed not at the surface of the target but at a small depth into it (a few mm in the case of ^{60}Co). This arises because more secondary electrons are lost from the surface of the material than are replaced by the secondary electrons generated in the air between the source and target. As it is these secondary electrons that are responsible for most of the absorbed dose, the dose at the surface will be lower than in a continuous medium. At depths in to the target greater than the range of secondary electrons an equilibrium will exist with electrons escaping from a given volume being replaced by an equal number entering the volume; such a condition is known as charged particle equilibrium or electron equilibrium.

Electron equilibrium conditions can be achieved with good approximation for photons having the energies 0.1 to 10 MeV. In the case of gamma radiation from a ^{60}Co source, approximate electron equilibrium in a small-irradiated volume can be achieved with about 0.5 g cm^{-2} of the material. This corresponds to the approximate range of the highest energy secondary electrons produced by 1.33 MeV photons.
When using a gamma-ray source or X-ray beam for calibration, surround the dosimeter with a sufficient amount of material to achieve approximate electron equilibrium conditions. For measurement of absorbed dose in water, use materials that have radiation-absorption properties essentially equivalent to water. For example, for a ^{60}Co source, 4.5 mm of solid polystyrene (or equivalent polymeric material) should surround the dosimeter in all directions. (3)

2.11 Photon Energy Dependence

Photon energy dependence is generally predicted by the ratio of mass energy coefficient of dosimeter and a reference medium (water) to photons of any energy with respect to that of a reference photon's energy, (usually 1.25 MeV of ^{60}Co gamma rays).

$$\text{Photon Energy Dependence} = \frac{\left(\frac{(\mu_{en}/\rho)_{dosimeter}}{(\mu_{en}/\rho)_{reference}}\right)_{energy}}{\left(\frac{(\mu_{en}/\rho)_{dosimeter}}{(\mu_{en}/\rho)_{reference}}\right)_{1.25MeV}} \qquad (2.23)$$

If the ratio is equal to 1.0 within ±10% over a significant range of the energy spectrum (for both calibration and test irradiations) incident upon both the dosimeter and the material of interest, then the energy-response performance of the dosimeter is acceptable. (3)

2.12 Electron Scattering

Large effects from electron scattering upon the dose distribution at interfaces between materials of different atomic numbers have been observed. When the dosimeter is small in size, even a small difference in Z between the medium and the dosimeter may make a significant difference in equilibrium influence at the interface due to electron scattering. When the dosimeter thickness is more than

the maximum range of the secondary electrons, the electron scatter from materials of different atomic number surrounding the dosimeter does not influence the absorbed dose in the dosimeter.
To overcome the scatter problem, the small volume dosimeters, should be calibrated in the same medium as in the following irradiations, and with the same photon energy spectrum. (3)

2.13 Interface effects

The stopping power of different medium for low energy electrons may differ considerably even when their densities are nearly equal. Since scattered electrons are of very low energy, the differences in stopping powers of two mediums may give rise to interface problems. The absorbed dose variation increases with the difference in atomic number between medium and the dosimeter.
The stopping power of electrons having energy less than 1 keV is about 40% lower in water than in polyethylene, Perspex and polystyrene. Since the scattered electrons are of very low energy, a thin aqueous solution enclosed in the above materials will show a lower response. Less number of these electrons escapes the surrounding materials because of the high absorption in them. These interface problems will be more acute when there is a large difference in composition of the dosimeter and its surroundings. (3)

2.14 Angular dependence

The number of scattered electrons reaching a thin dosimeter from the surrounding material depends upon the inclination of the dosimeter to the radiation beam. The dose received by dosimeters in inclined positions with respect to radiation field is higher than in normal positions. Small size dosimeter will have some angular depending on its size and composition and it will affect the accuracy of the dosimetry. (3)

2.15 Dosimetry principles

The dose in a medium is measured by replacing the medium by a dosimeter. Normally, dosimeter will differ from the medium in both atomic number and density and it therefore constitutes a discontinuity, which will be referred to as a cavity. The energy absorbed in the dosimeter is therefore not the same as that absorbed by the medium.

For irradiations using a photon source, the dosimeter may be considered as a cavity in the material of interest, and the interpretation of absorbed dose in materials as follows. (3)
If the sensitive region of the dosimeter is very thin compared to the range of the highest energy secondary electrons, then most of the energy deposited in the dosimeter and in the material surrounding it, results from secondary electrons produced outside the dosimeter (that is, in the equilibrium layer of material). Thus, the absorbed dose in the material, D_m, is given by,

$$D_m = \frac{(S/\rho)_m}{(S/\rho)_d} D_d \qquad (2.24)$$

Where $(S/\rho)_m$ and $(S/\rho)_d$ are mass collision stopping power for the surrounding material and dosimeter, respectively. D_d is absorbed dose in the dosimeter. (3)
If the sensitive region of the dosimeter has a thickness much greater than the range of the highest energy secondary electrons, then most of the energy deposited in it, results from the secondary electrons produced within the dosimeter itself. Thus, the absorbed dose in the material is given by

$$D_m = \frac{(\mu_{en}/\rho)_m}{(\mu en/\rho)d} D_d \qquad (2.25)$$

Where $(\mu_{en}/\rho)_m$ and $(\mu_{en}/\rho)_d$ are the mass absorption coefficients of the medium, m and the dosimeter material, d, respectively.
If the sensitive region of the dosimeter has a thickness between the two limits, then the above two equations may be combined with appropriate weighting factors to reflect the relative contributions of each term. (3)
The collision stopping powers and the energy absorption coefficients are energy dependent; however, for low atomic number materials and for the energy range 0.1 to 10 MeV, the ratios of the stopping powers and energy absorption coefficients do not vary significantly as a function of energy. (3)
Although this definitions given strictly for absorbed dose at a point in radiation absorbing matter, it is generally averaged over a finite mass of a given material, the absorbed dose being read by a dosimeter calibrated in terms of energy imparted per unit mass of a given material. In radiation processing, the reference material in most calibrations is water. It is important to realise that the

dosimeter is intended to give a measurement of absorbed dose averaged over a small volume. The signal from this dosimeter can therefore be a measure of energy absorbed in that volume, which can be related to the dose in the product.

Chapter (3)
DOSIMETERS IN GENERAL

3.1 Introduction:

Use of ionizing radiation has become increasingly important in diverse fields; industrial processing applications such as polymer degradation, polymer cross linking, polymer grafting, sterilization of medical products, sewage sludge hygienisation, delayed ripening of fruits, sprout inhibition and insect population control. The absorbed doses employed in these applications range from about 10Gy - more than 100kGy.

In the use of ionizing radiation, reproducible and an accurate irradiation of the products to get a desired effects depends on having reliable dosimetry systems. Dosimetry plays a vital role in setting of process parameters to meet variety of specifications, dose mapping in products, carrying out validation, commissioning procedure as well as in the day-to-day operation of the plant. Measuring response of the dosimeter to radiation is generally much easier and quicker to measure than any other parameter of the product that has been irradiated. Documentation of dose is often required in order to ensure operational safety and quality control.

3.2 Dosimetry systems

Dosimeter systems are used to measure absorbed dose. They consist of the dosimeters, measurement instruments and their associated reference standards, and procedures for the system's use. Dosimeters may be divided into four basic classes according to the accuracy of the dosimetry systems and areas of application: primary standard, reference standard, transfer standard and routine dosimeters. (3)

3.2.1 Primary standard dosimeters

Primary standard dosimeters are established and maintained by national standards laboratories for calibration of radiation environments (fields) and other dosimeters. These are either ionization chambers or calorimeters. The overall uncertainly at the stage is ±1% (at 95%). (3)

3.2.2. Reference standard dosimeters

Reference standard dosimeters are used to calibrate radiation environments and routine dosimeters. Reference standard dosimeters may also be used as routine dosimeters. These are chemical dosimetry systems where response not only to radiation but also to other influencing factors such as temperature and dose rate is reproducible and well characterized. (3)
The reference standard dosimeters are classified as Type 'A' and Type 'B'. The type 'A' dosimeters, for e.g., Fricke, Ceric-cerous, dichromate, ethanol-monochlorobenzene solutions, are systems whose radiation chemical response is characteristic of a particular chemical composition and this can be guaranteed, certainly to within a few percent, provided a good chemical practice is followed. Type 'B' dosimeters are systems which exhibits high precision but have to be calibrated against a standard higher in the series. Their radiation response cannot be predicted purely on the basis of composition. The overall uncertainty associated with reference dosimeters is about ± 3%. (3)

3.2.3 Transfer Standard Dosimeters

Transfer standard dosimeters are specially selected dosimeters used for transferring absorbed-dose information from an accredited or national standards laboratory to an irradiation facility in order to establish traceability for that facility. These dosimeters should be used under conditions that are carefully controlled by the issuing laboratory. Transfer standard dosimeters may be selected from either reference standard dosimeters or routine dosimeters and shall have performance characteristics such as long shelf life, easily calibrated, stable, portable, mail-able, broad absorbed dose range, radiation absorption properties similar to those of irradiated product, relatively insensitive to extremes of environmental conditions, correctable systematic errors (e.g., temperature, humidity, etc,), produced in reproducible lots, reproducible response, small dimensions compared to distances over which absorbed dose gradients become significant, etc.(3)

3.2.4 Routine dosimeters

Routine dosimeters may be used for quality control and process monitoring. Routine dosimeters are systems whose performance, particularly with respect to environmental influencing factors, is not

as good as the reference systems, but whose ease of use and low cost makes them ideal for day to day monitoring of radiation doses during processes. Criteria for selection of routine dosimetry system includes suitability of the dosimeter for the absorbed-dose range of interest and for use with a specific product, stability and reproducibility of the system, ease of system calibration, traceability of system calibration to national standards, ability to control or correct system response for system uncertainties, such as those caused by temperature and humidity, ease and simplicity of use, availability of dosimeters in reasonably large quantities, overall initial and operational cost of the system, including dosimeters, readout equipment, and labor, time required for dosimeter response development, and labor required for dosimeter readout and interpretation, etc. (3)

Table (3.1): Various radiation dosimetry standards

Dosimeters		
Primary standards dosimeters (1-2%)	Reference standards dosimeters (3%)	Routine dosimeters (5%)
Ionization chambers Calorimeters	Fricke Ceric-cerous Dichromate Ethanol Monochlorobezene	Plastics Dye plastics Alanine Radio-chromic dye films TL dosimeters

3.2 The procedure for the measurement of absorbed dose or dose rate

The reference source is calibrated by means of one of the primary standards (e.g. Graphite Calorimetry) or standardized reference methods (e.g. Fricke dosimetry), and the dosimeter signal is then converted to absorbed dose or dose rate in water by computations based on cavity theory. The reference source is used to irradiate the routine dosimeter under identical conditions, and the calibration is usually expressed in terms of dose or dose rate in water, rather than in terms of dose or dose rate in the dosimeter medium. The routine dosimeter is then exposed together with the product in the irradiation facility, although the irradiation conditions

(i.e. energy spectrum and geometry) may be markedly different from those of the calibration. (3)

3.4 Measurement traceability

Measurement traceability is the ability to demonstrate by means of an unbroken chain of comparisons that a measurement is in agreement within acceptable limits of uncertainty with comparable nationally or internationally recognized standards. The absorbed dose measurements to be traceable to nationally or internationally recognized standards, it is necessary to calibrate the overall dosimetry system. It is necessary to conduct calibration and performance checks of the individual instruments of the system to ensure that they are functioning in accordance with their own specifications. (3)

3.5 Calibration

3.5.1 Analytical system calibration

Before the overall dosimetry system is calibrated, and at periodic intervals between calibrations, the individual component instruments of the dosimetry system shall be calibrated or shall have their performance verified. (3)

3.5.2 Calibration of Dosimetry system

The calibration of a dosimetry system consists of routine dosimeters to a number of known absorbed doses over the range of use, analysis of the dosimeters using calibrated analytical equipment, and generation of a calibration curve. Calibration verification is performed periodically to confirm the continued validity of the calibration curve.

Routine dosimetry systems can be calibrated by irradiation at a high dose radiation dosimetry calibration laboratory, an in-house calibration facility whose dose rates have been demonstrated to be traceable to appropriate national standards or irradiation of reference or transfer standard dosimeters with routine dosimeters in the production irradiator facility. All possible factors that may affect the response of dosimeters, including environmental conditions and variations of such conditions within a processing facility should be known and taken in to account. (3)

3.5.3 Calibration Curve:

Calculate and document the mean response, $\bar{k}$, and the sample standard deviation (S_{n-1}) for each set of dosimeters at each absorbed-dose value. The sample standard deviation, $S_{n-1,}$ is calculated from the sample data set of n values as follows:

$$S_{n-1} = \sqrt{\frac{\sum(k_i - \bar{k})^2}{n-1}} \qquad (3.1)$$

Where k_i = i^{th} value of k.

Calculate the coefficient of variation, CV, for each absorbed dose value as follows:

$$CV = \frac{S_{n-1}}{\bar{k}} x100(\%) \qquad (3.2)$$

In general, if any CV values are greater than a value prescribed for a specific dosimetry system and application, then a re-determination of the data should be considered, or the stock of dosimeters should be rejected.
Dosimeter response graphically ploted versus absorbed dose, to derive this relationship in mathematical form, an analytical form (for example, linear, polynomial, or exponential) provided an appropriate fit to the measured data. Examine the resulting calibration curve for goodness of fit within specified limits. The goodness of fit can be evaluated by several parameters that are obtained as part of the fitting process. One of such parameters is the coefficient of determination, r^2 (correlation coefficient). The criterion for evaluating the values of the r^2 test is that a good fit is indicated by values close to 1. Calibration curve generated using one analytical instrument shall not be used with another instrument.

3.5.4 Transit Dose effects

Transit dose effects occur when the timing of a calibration irradiation does not take into account the dose received during the movement of the dosimeters into and out of the irradiation position. For example, the timer on a Gamma cell-type irradiator does not start until the sample chamber reaches the fully down (irradiate) position. Some dose is received as the drawer goes down and after the irradiation as the drawer goes up that is not accounted for in the timer setting. This transit dose can be significant for low-

dose irradiations and should be determined experimentally and incorporated into a formula for calculating timer settings. A graph of the transit dose in fig (3.1), the intercept with Y-axis gives the value of the transit dose. (3)

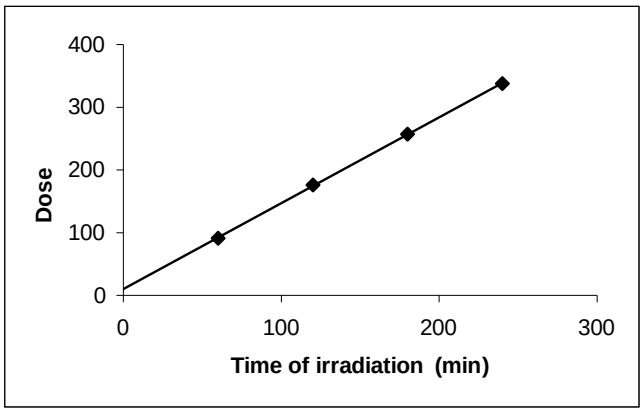


Fig (3.1) Transit dose

3.6 Uncertainty

Uncertainty of measurement comprises many components. Some of these components may be evaluated from the statistical distribution of the results of a series of measurements and can be characterized by experimental standard deviations (Type A). The experimental standard deviation characterizes the variability of the observed values or their dispersion about their mean. The other components, which also can be characterized by standard deviations, are evaluated from assumed probability distributions based on experience or other information (Type B). The standard uncertainty is combined in quadrature with other components of uncertainty to obtain an estimate of combined uncertainty in absorbed dose. (3)
Contributions to the combined uncertainty in the experimental value for the measurement of absorbed dose by a dosimetry system include the following:

1. Uncertainty in the absorbed dose received by the dosimeters during system calibration
2. Analysis of dosimeter response
3. Fit of dosimetry data to a calibration curve and
4. Routine use of dosimeters in a production irradiation facility.

Uncertainty in the absorbed dose received by the dosimeters during system calibration includes response of reference

standards dosimeter, irradiation time, decay correction of the source, non-uniformity in the standard source. Difference in the environment between calibration and use represents a potentially significant source of uncertainty in measurement of absorbed dose with routine dosimeters in the irradiation facility. (3)
Variation in the response of several dosimeters that are irradiated under the same conditions can be used to estimate Type A components of uncertainty. Measurement equipment such readout equipment, etc may introduce Type B component of uncertainty in the determination of dosimeter response. (3)
Sometimes response of the dosimeters has to be evaluated under the conditions of repeatability and reproducibility.

3.7 Repeatability

Repeatability (of results of measurements)- closeness of the agreement between the results of successive measurements of the same measured carried out subject to all of the following conditions: the same measurement procedure, the same observer, the same measuring instrument, used under the same conditions, the same location, and repetition over a short period of time. (3)

3.8 Reproducibility

Reproducibility: closeness of agreement between the results of measurements of the same measured, where the measurements are carried out under changed conditions such as differing: principle or method of measurement, observer, measuring instrument, location, conditions of use, and time. At an irradiation facility, over a period of several days, different sets of dosimeters irradiated to the same nominal dose and measurements are carried out under the condition of reproducibility. Possible sources of random errors could include intrinsic variation in dosimeter response and day-to-day variation in the physical environment, for example, temperature, positioning of dosimeters within the irradiator, and shielding. (3)

Chapter (4)

DOSIMETRY IN RADIATION PROCESSING

4.1 Introduction

In the use of ionizing radiation for processing applications, reproducible and an accurate estimation of radiation dose absorbed by the product is extremely important, as absorbed radiation is the only parameter that needs to be controlled in the process. Therefore, dosimetry plays a vital role in setting of process parameters to meet variety of specifications, dose mapping in products, carrying out validation, commissioning procedure as well as in the day-to-day operation of the plant. The dose delivered to the material/product should be high enough to ensure the intended radiation effect but must not exceed the level necessary for reasons of time, efficiency and uniformity in the performance of the processed product. The reliability of dosimetry is now so well recognized, that the use of test procedures are reduced to the minimum, or often, even dispensed with, on a routine basis. The choice of the dosimetry technique generally depends on the specific properties and needs of the particular irradiation facility, such as geometry factors, magnitude of dose-rate, homogeneity and spectral distribution of the radiation field on one side, and on the properties of the material to be irradiated, on the other side.

Accurate dose measurements, traceable to recognized international/national standards are essential components of industrial radiation processing. It has an additional dimension of regulatory aspects when processes are related to public health such as sterilization of medical products and food irradiation. The radiation processing technology involves a dose span from few tens of grays for sprout inhibition of onions and potatoes to few hundred kGy in degradation of Teflon for making solid lubricant. This clearly indicates that it may not be possible to use a single dosimetry system for all these radiation-processing plants. The dosimetry systems used for calibration and routine plant monitoring of various radiation processing facilities are explained briefly.

4.2 Applications in Radiation Processing

Radiation is being widely utilized in such fields as industry, agriculture and medical treatment, these application are making significant contributions toward improving the standard of living in Sudan. The applications in radiation processing where dosimetry may be used are:

4.2.1 Cross-linking and Degradation of polymers

Polymers are cross linked or degraded in order to change their mechanical and chemical properties e.g. increase resistance to hazardous chemicals, tensile strength etc**.**

4.2.2 Wire and cable irradiation

Irradiation is used to cross link the plastic insulation of electric wire and cable in order to make it more resistant to high temperature and in that way reduce the possibility of a short circuit due to high current in the wire or cable, used in automobile and aircraft cable systems.

4.2.3 Environmental Applications

Radiation is used to reduce biological contaminants (bacteria) in the waste water and sludge and to precipitate sulfur and nitrous oxides pollutants that come out of stack gases in the industrial processes.

4.2.4 Food Irradiation

Irradiation can be used to sterilize, pasteurize or increase the shelf life of food commodities by killing the microorganisms usually present in food products.

4.2.5 Sterilization of medical devices

Radiation has been used for several decades to sterilized disposable medical gloves, needles, catgut and syringes using mainly gamma ray source.

4.2.6 Composite Materials

Radiation can be used for the polymerization of monomers absorbed in wood or concrete to reduce their porosity or for the development of new composite materials consisting of mixtures of resins (epoxy) and fibers (carbon, glass, etc.).

4.3 Gamma Chamber

Gamma chamber is a self-contained dry-storage gamma-ray irradiator that has cobalt-60 source pencils arranged in an annular source cage. Gamma chamber is a compact unit offering an irradiation volume of a few hundred cubic centimeters. Adequate shielding is provided so that the radiation leakage outside the unit is well below the maximum permissible dose level. The main unit consists of a source cage, biological shield for the source and a central shaft incorporating sample chamber. The source cage holds the radiation source in an annular cylinder. The coaxial hole in the center of the cage provides space for the irradiation chamber. The cage is designed to hold pencils containing cobalt-60 in the form of pellets or slugs. The lead shield surrounding the source cage serves the dual purpose of a transport container and the biological shield. The sample chamber is raised or lowered by a wire rope using a system of pulleys and a rotating drum.(2)

It is an ideal irradiation facility establishment to promote the research on applications of radioisotopes in medicine, healthcare, agriculture and industry, for calibration of dosimeters maintained by radiation standards laboratory and radiation processing facilities.

Dosimetry of these units is carried out by Fricke dosimeter using to measure dose rate at the center of the irradiation volume. However, to measure relative dose distribution within this irradiation volume, alanine dosimeter is used. (2)

The Gammacell 220 Excel is a cobalt 60 irradiation facility manufactured by MDS Nordion. Figure (3.1) illustrates the external features, the unit consists of annular sources, permanently enclose within a lead shield, a cylindrical drawer which carries samples to and from the irradiation position, and a drive mechanism to move the drawer up or down along the vertical source center-line.

The chamber, Contained within the drawer, can accommodate sample up to approximately six inches in diameter and eight inches in height. An access tube in the upper portion of the drawer can introduce liquid, gaseous, electrical or mechanical connections into the chamber.

An electrically powered digital timer automatically signals the drawer to raise at the termination of a sample irradiation.

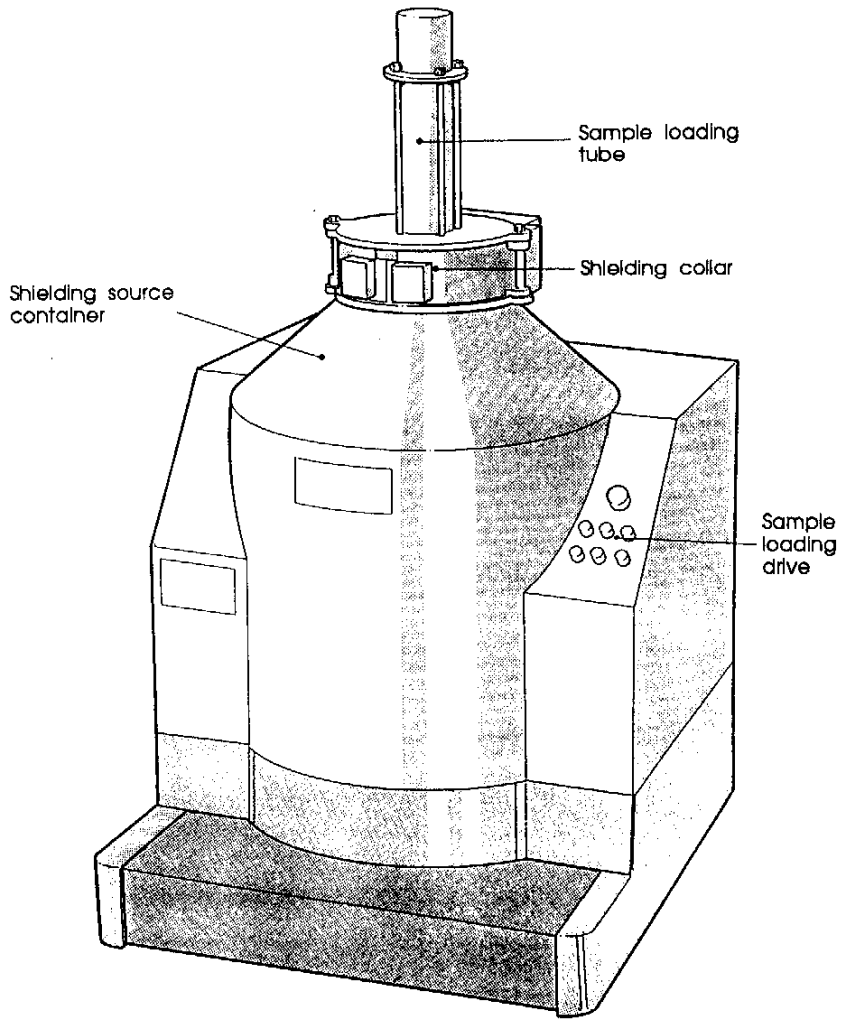


Fig (4.1) The external features of Gammacell 220 Excel

Chapter (5)

CHEMICAL DOSIMETRY (GENERAL PRINCIPLES)

5.1 Introduction

Quantitative studies in radiation chemistry require knowledge of amount of energy absorbed from the ionizing radiation. Determination of this absorbed energy is called Radiation dosimetry. The systems used for this purpose are called dosimeters.
A chemical dosimeter is a system that measure the energy absorbed from ionizing radiation by virtue of chemical changes produced in it when it is exposed to ionizing radiation. In all chemical dosimeters radiation induced chemical reaction produces at least one, initially absent species, which is long lived enough to determine its quantity or the change in the properties of the initial system. The absorbed dose is then derived from this quantity of the change in the properties of the system.

5.2 Salient features of chemical dosimeters

Chemical dosimeters are prominent in present day dosimetry. They are very useful in routine dose measurements due to their greater convenience.
Ample choice of chemical dosimeters enables their use for various purposes. Various systems that have been tried for chemical dosimetry are: (3)

- Water and aqueous solutions
- Pure organic substances and organic solutions
- Monomers and polymers
- Organic and inorganic gases
- Organic and inorganic solids

Calculation of dose using a chemical dosimeter requires the knowledge of G value, for the reaction or the product estimated. This is determined by calibrating a chemical dosimeter against some absolute dosimeter as a reference. Hence chemical dosimeters are secondary standard dosimeters.
When uses a chemical dosimeter the quantity it measures is the average absorbed dose in the material of which it is composed. It can be converted into the absorbed dose in other specimen. Accurate dosimetry in this case is facilitated when the irradiated volume of the dosimeter and the specimen studied have

approximately the same density and atomic composition. From this standpoint, chemical dosimeters offer a broad range of flexibility. (3)

Chemical dosimetry is the obvious method of dose measurements in radiation chemistry since both sample and the dosimeter can be irradiated in the same type of container (irradiation cell) occupying the same volume. This makes dosimetry simple and precise. In biological and therapeutically work, an aqueous chemical dosimeter enclosed in a suitable container, closely approximates the density and atomic composition of biological materials. It is also widely used for calibrating commercial dosimeters such as glasses and plastic films, and in experiments for which maximum accuracy is required. A dose range of 1 to 10^6 Gy can be covered using different chemical dosimeters. (3)

5.3 Requirements a chemical dosimeter should meet

Any quantitative well-characterized reaction can be used for chemical dosimetry. Several such reactions have been extensively studied to exploit their potential as a chemical dosimeter. However a very few systems have come up to the mark. An ideal chemical dosimeter is required to meet the following requirements:

- The chemical change produced in the dosimeter by radiation should be proportional to the absorbed dose in the dose range where it is used.
- The chemical change should be independent of quality and LET of radiation.
- The chemical change should be reproducible with a precision of $\pm$ 1% to $\pm$ 5%.
- The chemical change should be independent of temperature of irradiation and measurement.
- The chemical change should be independent of small changes in the composition of the dosimeter.
- The chemical dosimeter should be easy to prepare and use.

Not a single dosimeter meets all the above-mentioned requirements. However we can always select one, which satisfies maximum requirements and has known dependence on the parameters like LET, dose rate, temperature etc. (3)

5.4 Pre-requisites of chemical dosimetry

Radiolytic reactions are extremely sensitive to trace impurities, hence water; glassware and irradiation cells used for dosimetry must be free from trace amounts of organic and inorganic impurities.

5.4.1 Purification of water

Purity of water required depends on the dosimeter used. The procedure followed to produce triple distilled water free from inorganic and organic impurities is as follows:
To test the purity of distilled water prepares two aerated 1mM ferrous ammonium sulphate solutions in 0.8 N Sulphuric acid, (A) with 1mM sodium chloride and (B) without sodium chloride. Both the solution is irradiated under identical conditions for different doses. On measurement of ferric ion concentration formed, if it is found that: (i) formation of ferric ions is linear versus dose and (ii) the rates of ferric ion formation are identical for both the solution (A) and (B), then it indicates the absence of trace impurities in the distilled water and hence the prepared distilled water is of high purity. (3)

5.4.2 Treatment of glassware and irradiation cells

Glassware and irradiation cell used for dosimetry are generally made from Borosilicate or silica glass. (3)

Cleaning

Glassware or irradiation cells are filled with or dipped in 1:1 mixture of concentrated sulphuric and nitric acids for 24 hours. Then they are washed successively with tap water and distilled water. Then they are filled distilled water and exposed to a dose of approximately 1kGy. (3)

Storage

Irradiation cells purified in this way are kept filled with distilled water or with dosimetric solution, when not in use. (3)

5.4.3 Treatment of plastic irradiation cells

Use of materials like Teflon, Perspex, polyethylene, polystyrene and polypropylene as container for a chemical dosimeter requires a special procedure for cleaning, storage and use. The inner wall of the irradiation tube should be inert i.e. it should not liberate any impurities during irradiation or pre and post irradiation storage of dosimetric solution. The procedure followed is as follows:

Cleaning

Plastic tubes are filled with 10% HNO_3 for 24 hours. Then they are rinsed with tap water and then distilled water. To make the tube free from any further detachable impurity from the inner wall, the tubes are filled with dosimetric solution and exposed to a dose of 40 Gy. (3)

Storage

Plastic tubes are filled with dosimetric solution when not in use. (3)

5.4.4 Use of irradiation cells

When glass or plastic irradiation cells are to be used for dosimetry, the old dosimetric solution in them is discarded. The cells are rinsed at least two times with fresh dosimetric solution. They are then filled with the solution, stoppered and irradiated. (3)

5.5 Definition of G value

G value or Radiation Chemical Yield is the fundamental quantitative characteristic of radiation-induced reaction in a chemical dosimeter. The **radiation chemical yield**, G (x), of an entity, x, is the quotient of n (x) by ε, where n (x) is the mean amount of substance of that entity produced, destroyed, or changed in a system by the energy imparted, ε, to the matter of that system, thus

$$G(x) = \frac{n(x)}{\varepsilon}. \quad \text{Unit: mol J}^{-1} \qquad (5.1)$$

In old unit, it is defined as the number of ions; free radicals, or atoms, molecules formed or disappeared when the system has absorbed 100 eV of energy from the ionizing radiation. (3)
It is designated by the letter G and the convention is to use the letter G followed by the chemical formula of the substance enclosed in the bracket, e.g. G (Fe^{3+}) indicate ferric ions are formed on irradiation. When the formula is preceded by –ve sign it indicates that the substance decomposes or its concentration decreases on irradiation, e.g. G ($-H_2C_2O_4$) shows that oxalic acid decomposes on irradiation. (3)
For any dosimeter in which radiation induces a chemical change, the mean absorbed dose (D_d) over the volume occupied by the dosimeter is given by

D_d = (Yield of radiation induced product)/ (radiation chemical yield)

(mol kg^{-1}/mol J^{-1}) (5.2)

5.6 Measurement of absorbed dose

By employing a chemical reaction for which radiation chemical yield G (X) is known and by measuring the concentration ΔC mol m^{-3} of the product x formed, the energy absorbed in the system "D" Gy can be calculated by the equation

$$\text{Dose (Gy)} = \frac{\Delta C * 9.64x10^{6}}{G(X) * \rho} \qquad (5.3)$$

Where ρ is density of the system in units of kg m^{-3}. The G (X) must be measured for each radiation for which a particular dosimeter is to be applied. (3)

5.7 Spectrophotometry

Spectrophotometry is concerned with the determination of the concentration of a substance by measurement of relative absorption of light with respect to a known concentration of the substance. When monochromatic light passes through a transparent medium, the intensity of the emitted light decreases exponentially, as the thickness of the absorbing medium increases arithmetically, (Lambert's law). Similarly, when monochromatic light passes, the intensity of light decreases exponentially as the

concentration of the absorbing substance increases arithmetically (Beer's law). Combining the above two laws, the following expression is obtained, (3)

$$I_t = I_o.e^{-\varepsilon ct} \qquad (5.4)$$

$$\text{Or} \quad \log I_o/ I_t = \varepsilon ct$$

Where ε the molar extinction coefficient, c is the concentration of the absorbing substance and t is the path-length. This fundamental equation of spectrometry is referred to as Beer-Lambert's law. The ratio $\log I_o/ I_t$ is called as optical density or absorbance of the medium. So optical density is related to ε by (3)

$$O.D. = \varepsilon ct \qquad (5.5)$$

Therefore for absorbance measurements, the equation for estimating dose can be written as (3)

$$\text{Moles product formed per } m^3 = \frac{\Delta A}{\varepsilon.t} \qquad (4.6)$$

Where, ΔA is the difference in absorbance (optical density) between the irradiated and un-irradiated solution.
The absorbed dose is calculated from the following equation, (3)

$$\text{Dose (Gy)} = \frac{\Delta A}{G(X).\rho.\varepsilon.t} \qquad (5.6)$$

Where, ε is in $m^2 mol^{-1}$, t is in m, ρ is in kg m^{-3}, G (x) is in mol J^{-1}.

5.8 CHEMICAL DOSIMETERS

5.8.1 Fricke dosimeter

The Fricke dosimetry system provides a reliable means for measurement of absorbed dose in water, based on a process of oxidation of ferrous ions to ferric ions in acidic aqueous solution by ionizing radiation. The Fricke dosimeter is widely accepted as standard in radiation chemistry and because of its accuracy and reliability it is used for calibrating other dosimeters.

➢ Principle

When an air saturated dilute ferrous ammonium sulfate solution in 0.8 N H_2SO_4 is exposed to ionizing radiation, ferrous ions are oxidized to ferric ions.

$$Fe^{2+} \rightarrow Fe^{3+} \qquad (5.7)$$

Concentration of ferric ions gives the measure of absorbed dose. The concentration is measured spectrophotometrically.

➢ Composition

The standard dosimetry solution consists of air-saturated solution of

$Fe(NH_4)_2(SO_4)_2$	---	1 mol M^{-3}
NaCl	---	1 mol M^{-3}
H_2SO_4	---	400 mol M^{-3}

Prepared in distilled water

➢ Reaction Mechanism

All reactions oxidizing ferrous ions to ferric ions are quantitative, irreversible and well established. (4)

$$H_2O \rightarrow \bar{e}_{aq}, H, OH, H_2, O_2$$

$$H + O_2 \rightarrow HO_2$$

$$Fe^{2+} + HO_2 \rightarrow Fe^{3+} + HO^-_2$$

$$H^+ + HO_2^- \rightarrow H_2O_2$$

$$Fe^{2+} + H_2O_2 \rightarrow Fe^{3+} + OH + OH^-$$

$$Fe^{2+} + OH \rightarrow Fe^{3+} + OH^-$$

$$\mathbf{G(Fe^{3+}) = 3\,GH + 2\,GH_2O_2 + GOH} \qquad (5.8)$$

$$= \mathbf{1.61\ \mu mol\ J^{-1}\ for\ Co\text{-}60\ gamma\ rays.}$$

➢ Measurement

G (Fe^{3+}) for ^{60}Co-gamma rays is 1.61μmol J^{-1}. The range of absorbed dose of the Fricke dosimeter is from 40 to 400 Gy. Fe^{3+} ion concentration is measured spectrophotometrically at 304 nm. The molar absorption coefficient at 304 nm increases with rising temperature at a rate of 0.69% per °C, so that, for high precision, a constant temperature absorption cell holder on the spectrophotometer is required.

➢ Role of NaCl

To desensitize the dosimeter to organic impurities, sodium chloride is added.

➢ Analysis

Chemical titration may be used to determine the ferric ion concentration. Absorption spectroscopy is generally more convenient than titration, for it is rapid, accurate, and expedient for the analysis for low ferric ion concentrations and small quantities of solution.

Ferric ion has absorption peak at 304nm. At 304nm the molar linear absorption coefficient of ferrous ion is less than 0.05% of that of the ferric ion. The molar absorption coefficient at 304nm increase with rising temperature at a rate of 0.69% per °C so that, for high precision, a constant-temperature in cell holder on the spectrophotometer is required and it has to be measured for each spectrophotometer at a standard (25°C) temperature as this value varies from one instrument to other.

Absorbance of the irradiated dosimeters is measured against the un-irradiated solution from the standard flask. Optical density of the un-irradiated reference blanks in the dosimetric tubes is also measured against solution in standard flask. Absorbance of the irradiated solution is corrected for reference blank reading. The absorbed dose is calculated from the following equation.

$$D = \frac{\Delta A}{\rho \varepsilon l G(x)} Gy \qquad (5.9)$$

Where ΔA = the difference in absorbance (optical density) between the irradiated and unirradiated solution.

ε = Molar linear absorption coefficient ($m^2\ mol^{-1}$)

t = Optical path length = 0.01 m

ρ = Density of solution 1024 kg m^{-3}

G (x) = G (Fe^{3+}) =1.61 μmol J^{-1}

Substituting the values,

$$D = \frac{\Delta A}{\varepsilon * 0.01 * 1024 * G(Fe^{3+})} Gy \qquad (5.10)$$

$$D = 0.0977\ \Delta A/G\ (Fe^{3+}) * \varepsilon \text{ at } 25°C \qquad (5.11)$$

The above equation is valid when absorbance of irradiated samples is measured at 25°C. When the temperature of spectrophotometric measurement is different, 0.7% correction has to be applied to molar linear absorption coefficient, ε. The equation will be modified to,

$$\text{Dose (Gy)} = \frac{0.0977 * \Delta A}{G * \varepsilon * [1 + 0.007(25 - t)]} \qquad (5.12)$$

Where t = temperature of spectrophotometer cell and sample solution during absorbance measurement, °C.

The above equation gives absorbed dose in Fricke dosimeter and it can be used to determine the transit dose by substituted the value of intercepted from Y-axis. The equation will be modified as,

$$\text{Transit dose (Gy)} = \frac{0.0977 * \text{intercept of Y-axis}}{G * \varepsilon * [1 + 0.007(25 - t)]} \qquad (5.13)$$

The absorbed dose in water 'D_w' and absorbed dose in Fricke' dosimeter 'D_d' are connected by the equation.

$$D_w = D_d \frac{(\mu en/\rho)w}{(\mu en/\rho)d} \qquad (5.14)$$

Where $(\mu en/\rho)_w$ and $(\mu en/\rho)_d$ are coefficient of mass energy absorption of water and dosimeter respectively. For Co-60 gamma rays $(\mu en/\rho)_w / (\mu en/\rho)_d$ is 1.003

➢ Dose range

The readily accessible dose range of Fricke dosimeter with adequate accuracy is 40 to about 400 Gy with commonly employed 1 mol.m^{-3} ferrous sulphate solution. The upper limit is set by oxygen depletion in the solution and lower limit by the sensitivity of analytical method used for determining ferric ion concentration. With special techniques it is possible to extend the useful range beyond these limits.

5.8.1.1 Influence of various factors

➢ Oxygen concentration

Formation of ferric ions in Fricke dosimeter is directly proportional to energy absorbed as long as oxygen remains in the solution. Slope of dosage curve changes on exhaustion of oxygen. When oxygen is completely exhausted, reaction mechanism changes reducing G (Fe^{3+}) to 0.85 μmol.J^{-1}.

$$Fe_2^+ + H + H \longrightarrow Fe_3^+ + H_2 \qquad (5.15)$$

Here one H atom oxidizes only one ferrous ion to ferric instead of three. Therefore,

$$G(Fe_3^+) = 2GH_2O_2 + GOH \qquad (5.16)$$

➢ Initial ferrous ion concentration

The normal range of ferrous ion concentration is from 0.1 mol.m^{-3} to 10 mol.m^{-3}, both above and below this range of concentration in air saturated solutions G (Fe^{3+}) decreases.

➢ Sulfuric acid concentration

The sulfuric acid concentration can be lowered to 50 mol.m^{-3} with only 2% drop in G (Fe^{3+}) but below this concentration, the G(Fe^{3+}) falls and the system becomes unreliable. Therefore, a concentration of 50 mol.m^{-3} H_2SO_4 is the lowest that can be used for reproducible dosimetry.

➢ LET dependence

The G-value for formation of ferric ions depends on the type of radiation and its energy. Some typical values are given in the table (5.1).

Table (5.1): Types of radiation and its energies with G values

Type of radiation	Energy (MeV)	G (Fe^{3+}) μmol. J^{-1}
210 Po alpha	5.3	0.53
10 B	2.35	0.44
235 U fission products	152	0.31
60 Co gamma rays	1.25	1.61
X-rays	0.10	1.51
X-rays	0.05	1.48
X-rays	0.01	1.43

The G (Fe^{3+}) shows no significant dependence on photon energy within the range from 0.60 MeV. At lower photon energy as the LET of radiation increases with decreasing energy, G (Fe^{3+}) shows decreasing values. For heavier charged particles, which have higher LET the decrease in G (Fe^{3+}) is more pronounced.

➢ Temperature

By increasing temperature during irradiation G (Fe^{3+}) increases slightly. The values reported for this temperature dependence vary considerably, from insignificant to 0.16% per °C. The irradiation temperature is without much practical significance from 10°C to 50°C.

The temperature at which the spectrophotometric measurements are done is important as the molar linear absorption coefficient increases by about 0.69% per °C for the 304 nm peak. For this reason it is important to know the temperature at which the spectrophotometric measurements are done.

➢ Stability

The aerated ferrous sulphate solution is slowly oxidized during storage and oxidation rate depends on the concentration of ferrous ions and oxygen. For prolonged use the dosimetric solution can be

stored in dark preferably in a refrigerator. If dosimetry is done only occasionally, fresh solution should be prepared each time.

➢ Dosimetric container

The irradiation can be carried out in polystyrene and borosilicate (Silica) glass tubes or ampoules. The volume of the sample should be sufficient for the analysis of the ferric ions formed. A set of pre-cleaned tubes is filled with dosimetric solution after rinsing with the same. At least 2 tubes should be kept as reference blank and the remaining tubes by used for dose measurement.

5.8.2 Alanine dosimeter

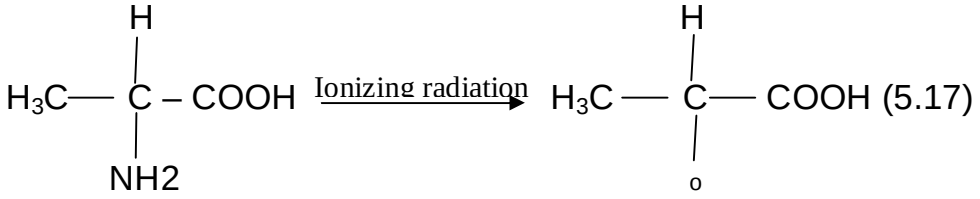
Formation of stable free radicals in solids like Alanine is being used for radiation dosimetry. When amino acid alanine is irradiated in a solid form, free radicals are produced which get trapped in a solid matrix and are stable for a long period. The irradiated alanine is read by ESR technique.

The total number of free radicals formed is proportional to the absorbed dose. The alanine –ESR (electron spin resonance) read out system has been accepted as a reference dosimeter. The dose range covered is from 0.001 to 100 kGy. This method though very accurate is not suitable for routine dosimetry because of high cost of ESR instrument.

An alternative readout technique based on Spectrophotometry uses the indirect oxidation of ferrous ions to ferric ions when irradiated alanine powder is dissolved in an acidic solution containing ferrous ammonium sulphate and xylenol orange. Ferric ions thus formed form a complex with xylenol orange. The complex is measured spectrophotometrically. The change in absorbance is a function of dose.

➢ Principle

Irradiation of the Alanine amino acid, Alanine produced stable free radicals.



(5.17)

➢ Composition

Dosimeter solution (FX) for $C_3H_7NO_2$ **Alanine** consists of,

0.20 mM…$Fe(NH_4)_2(SO_4)_2$ Ferrous ammonium sulphate

0.20 mM …$C_{31}H_{28}N_2Na_4O_{13}S$ Xylenol orange

In 0.05 N … H_2SO_4 Sulphuric acid

➢ Dosimetric container

The irradiation is carried out in stoppered polystyrene tubs of 3 mm id, 20 mm length and 1mm wall thickness.

➢ Measurement

Irradiated Alanine is taken in a dry 25 ml stoppered conical flask and dissolved in 10 ml of the respective FX solution. Concentration of ferric ion formed is measured as ferric-xylenol orange complex in presence of alanine against similar unirradiated solution as blank. In fact the measurements of the irradiated and unirradiated powders are made against FX solution and difference of these gives the correct reading. The absorbance of FX solution containing the unirradiated powder is less than that of the FX solution. Therefore the zero reading of the instrument should be suitably shifted to account for this.

The absorbance of this complex is measured with spectrophotometer at least 5 minutes after the dissolution. Optimum results will be obtained with 50 mg alanine for 10ml of FX solution. Ferric-xylenol orange complex is measured at 525 nm and the calibration curve for Alanine dosimeters are obtained for the xylenol orange concentration of $2x10^{-4}$M.

5.8.2.1 Influence of various factors

➢ Concentration dependence

For a given dose, the amount of ferrous ions oxidized was found to increase almost linearly with the increase in amount of irradiated alanine from 0 to 100 mg.

➢ Autoxidation

Alanine increases the rate of thermal oxidation of ferrous ion in the FX solution. The rate of thermal oxidation is less in irradiated solution as compared to the unirradiated solution. The absorbance thus depends on the time of measurement after the dissolution of alanine in the FX solution. The readings are taken 5 minutes after the dissolution. At this time, reaction in the solution is ever. However, this time limit is not very critical and slight variation does not make a significant error in does estimation. When the absorbance of FX solution containing unirradiated alanine is subtracted from the absorbance of FX solution containing irradiated alanine, the error due to thermal oxidation is negligible provided the measurements are made within an hour after the dissolution of the powder.

➢ Stability

Under normal temperature and humidity conditions recrystallised alanine dosimeter shows about 5 to 10% fading depending on dose after the first day of irradiation. Afterwards the readings remain constant for at least a year.

➢ Radiation dose response

The range of absorbed dose measured with alanine is 0.01 to 5 KGy.

Chapter (6)
Experimental Method

6.1 Fricke dosimeter

To calibrate the gamma chamber of the Gammacell 220 of (SAEC) the fricke dosimeter was used. The principle of fricke dosimeter is when an aerated dilute ferrous sulphate solution on 0.8 N H_2SO_4 is exposed to ionizing radiation, ferrous ions oxidized ferric ions.

$$Fe^{2+} \longrightarrow Fe^{3+}$$

The absorbed dose is derived, by measuring the concentration of the ferric ions in the solution spectrophotometrically after irradiation. The dose range can be used in calibration from 40 to 400 Gray.

Apparatus can be used for fricke are an ultraviolet spectrophotometer, quartz cavette with 10mm path length standard 500 volumetric flask, 10ml graduated pipette, spatula, 5ml breakers two, 5mm irradiation tubes made of polypropylene, 0-50c range thermometer, microbalance and a gamma chamber.

Chemicals used Ferrous Ammonium Sulphate, Sodium Chloride and Sulphuric Acid.

6.1.1 Preparation of Dosimetric solution

500 ml of Fricke Dosimeter prepared by taken 200ml-distilled water in the flask. Pipette out 11.2ml of 0.8N H_2SO_4 in the standard flask. 0.19608g of ferrous ammonium sulfate has been weighed and transfer to the flask.0146g of Sodium chloride weighed and transfer to the flask, the solution was made up with distilled water.

6.1.2 Preparations of dosimeters:

The tubes were rinsed three with the Dosimetric solution before filling them for irradiation. When using a gamma source for irradiation the dosimetric should be surrounded with 5mm thick Perspex build up material to achieve electron equilibrium conditions.

6.1.3 Irradiation and measurement procedure

Dosimeters were irradiated using gamma chamber at central position. For measurement spectrophotometer wavelength has been set at 304nm, clean cells have been filled with un-irradiated Dosimetric solution. Cells have been placed in the sample holder and reference holder. The spectrophotometer is adjusted at zero and 100% transmission. Once the spectrophotometer set to zero absorbance, the un-irradiated and irradiated solutions can be measured. (4)

Utmost has taken to keep the temperature in the cell compartment constant because the optical density (absorbance) changes with temperature. (The change in absorbance is 0.7% per degree centigrade). (4 - 8)

Three containers were used. 20 ampoules rinsed 3 times with the dosimetric solution and then filled. The irradiation times used for the group were, 30, 60, 90, 120, 150 and 180 sec.

* For each irradiated ampoule absorbance was determined.
* Blank absorbance was subtracted from the irradiated one.
* Excel software was used to draw the calibration curve.
* Absorbed dose on the Fricke calculated.
* Absorbed dose on water calculated.

All irradiation were done in Radiation Processing Lab/ Sudan Atomic Energy Commission (SAEC), using Gammacell 220 Excel Irradiator. This irradiator contains a Co-60 source, which has initial activity of 10038Ci; however, because of the natural decay of cobalt the actual activity now is 6421.26Ci (July 2005).

6.2 Alanine dosimeter

To calibrate the gamma chamber of the Gammacell 220 of (SAEC) for high doses using Alanine dosimeter up to 4000 Gy, alanine dosimeter was used.

The principle of the alanine dosimeter is: if irradiated materials are dissolved in ferrous sulphate solution, ferrous ions are oxidized. This suggests that peroxides are formed on dissolution of these irradiated materials. This oxidation of ferrous ions is used to measure the radiation dose. The dose range when using alanine dosimeter is from 10 to 5000 Gy.

Apparatus can be used for Alanine are, an ultraviolet spectrophotometer, quartz cavette with 10mm path length standard 500 volumetric flask, 10ml graduated pipette, spatula, 50ml beakers two, 25 ml stoppered conical flask, 1mm irradiation

tubes made of polystyrene, 0-50c range thermometer, microbalance and a gamma chamber.
Chemicals used Ferrous Ammonium Sulphate, Xylenol orange, Alanine, and Sulphuric Acid.

6.2.1 Preparation of Alanine dosimeter solution

Dosimeter solution (FX) for **Alanine** consists of,
0.20 mM Ferrous ammonium sulphate
0.20 mM Xylenol orange
In 0.05 N Sulphuric acid

Dosimetric solution was prepared using analytical grade reagents and distilled water. Use single distilled water does not affect the response of FX-Alanine system.
500 ml of dosimetric solution was prepared for Alanine; 13.9 ml of concentrated sulphuric acid was diluted to 100 ml. Five ml of this solution has been added to the 500 ml volumetric flask. 39.2 mg of ferrous ammonium sulphate and 76 mg of xylenol orange were added. A solution was made up to 500 ml by adding distilled water.

6.2.2 Irradiation and measurement procedure for Alanine:

Irradiated Alanine powder was taken in a dry 25 ml stoppered conical flask and dissolved in 10 ml of the respective FX solution. The measurements of the irradiated and un-irradiated powders were made against FX solution and difference of these given the correct reading.
The absorbance of this complex was measured with spectrophotometer 10 minutes after the dissolution. Optimum results have been obtained with 50 mg Alanine for 10ml of FX solution. Ferric-xylenol orange complex is measured at 525 nm. Calibration curve for Alanine dosimeters were obtained for the xylenol orange concentration of $2x10^{-4}$M.
Three containers were used. 20 ampoules rinsed 3 times with the Dosimetric solution and then filled. Each group of ampoules was irradiated suing times, 650, 1350, 2000, 2650, 3300 and 3950 sec.

* For each irradiated ampoule absorbance was determined.
* Blank absorbance was subtracted from the irradiated one.
* Excel software was used to draw the calibration curve.

The irradiation was done in Radiation Processing Lab/ Sudan Atomic Energy Commission (SAEC), by using Gammacell 220 Excel Irradiator contains Co-60 source, which has initial activity of 10038Ci; however, because of the natural decay of cobalt the actual activity now is 6421.26Ci (July 2005).

6.3 Absorbed-dose Mapping

Ideally, the radiation process is designed to irradiate the product or specimen uniformly; in realty, a certain variation in absorbed dose through the product will exist. Absorbed –dose mapping is used to determine the magnitude and locations of the Dmax and Dmin for a given set of operating parameters (e.g. timer setting, product loading configuration). For self-contained dry storage irradiators the product or specimen may be relatively close to the radiation source, resulting in pronounced absorbed-dose gradients near the periphery of the irradiation volume. It is important, therefore, to choose a dosimeter, which is small enough to detect these gradients. The routine Dosimetry system may be used for relative or absolute absorbed dose measurements or for mapping the absorbed dose distribution in the irradiation volume. In this work the dose mapping for the Gammacell 220 were determined using Fricke and alanine dosimeters.

Experimental Set-up:

- Sets of Fricke and Alanine dosimeters placed in different positions and irradiated with known dose.
- A correlation between dose in the reference position (center) to the minimum and maximum dose was determined.
- A comparison between Fricke dose mapping and alanine dose mapping to the typical Gammacell 220 dose distribution was carried out.

Chapter (7)
Results and Discussion

7.1 Results and Discussion of Fricke dosimeter:

Measurement of the dose rate using Fricke dosimeters is shown in table (7.1). The uncertainty of the reference dosimeter is 1.34%:

Table (7.1): Results of Fricke dosimeter measurement

Time (Sec)	Absorbance	Diff. Absorbance	Average in Diff. Absorbance	STDEV $\sqrt{\sum \frac{Ni - N^{-}}{n-1}}$	Calculated dose in water Gy/sec	Calculated dose per sec (Gy/sec)
30 30 30	0.258 0.265 0.260	0.211 0.218 0.213	0.214	0.004	1.488	1.484
60 60 60	0.409 0.409 0.401	0.362 0.362 0.354	0.359	0.005		
90 90 90	0.594 0.566 0.560	0.547 0.519 0.513	0.526	0.018		
120 120 120	0.726 0.726 0.723	0.679 0.679 0.676	0.678	0.001		
150 150 150	0.885 0.880 0.882	0.838 0.833 0.835	0.835	0.003		
180 180 180	1.039 1.050 1.028	0.992 1.003 0.981	1.039	0.011		

The value of the blank was found to be 0.047.

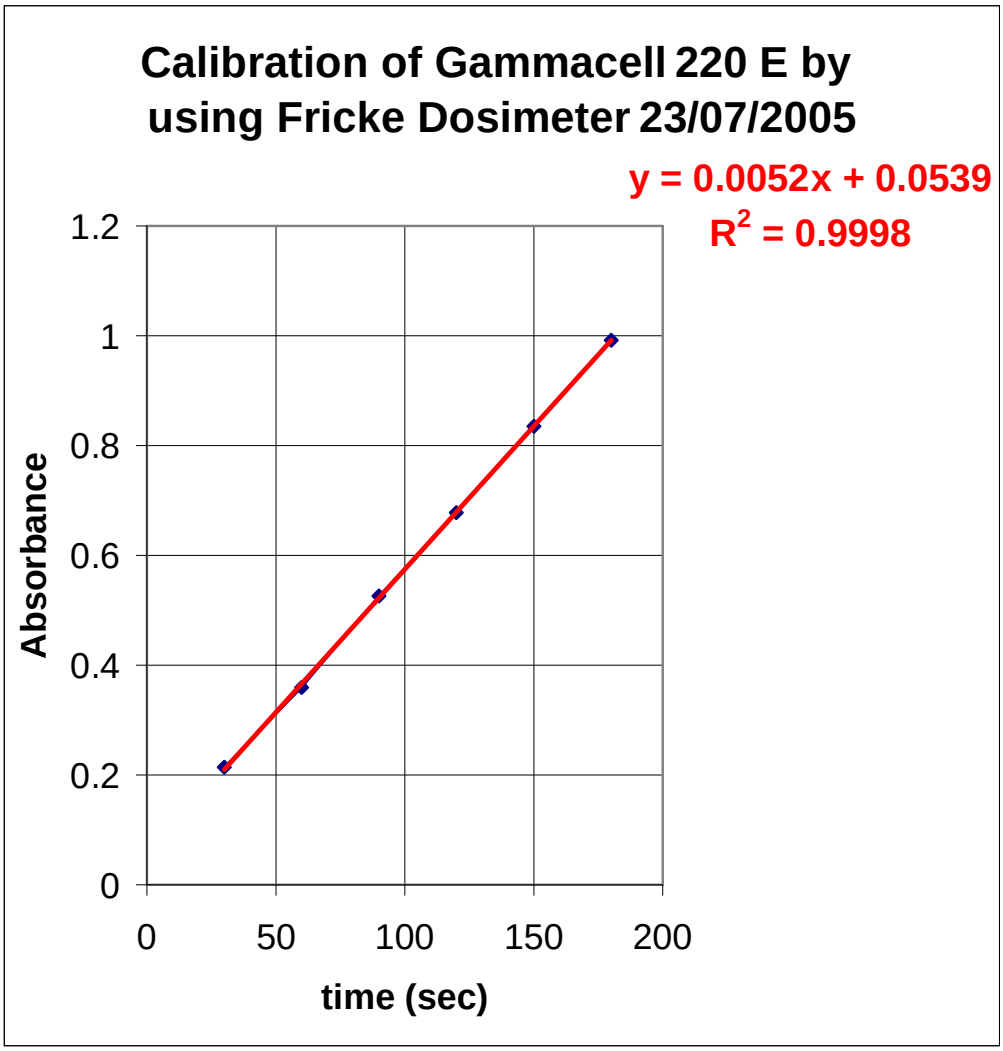
From table (7.1) it was found that the dose rate from the Gammacell 220 E of the SAEC is 1.484Gy/sec. Using equation (5.13), dose in water has been calculated from equation (5.15), which has, the value of 1.488 Gy/sec.

Table (7.2): Results of Fricke dosimeter measurement

Transit Dose (Gy)	Calculated Dose per hour (Gy/h)	Dose per hour by supplier as in July (2005) (Gy/h)	Difference in Percentage %
15.384	**5343.145**	**5276.680**	**1.24%**

From table (7.2) the comparison between the dose measurement by the manufacturer and the dose measured in this work is in good agreement, and the difference between the two doses was found to be 1.24%. This would confirm the suitability of the Fricke dosimeter for reference calibration of the Gamma cell 220 E.
The transit dose in definition is the dose received by the sample when it goes down to the irradiation position and getting up after ending of the irradiation. In case of the sample irradiation the transit dose should be considered. The transit dose received in each irradiation can be calculated using equation (5.14), and it was found to be 15.384Gy, it can be calculated from the intersection with the Y-axis in Fig (7.1), where the absorbance was found to be 0.0539. By plotting the dose received by each group of ampoules against the Difference in absorbance using table (7.1), Fig (7.1) was obtained.

Fig (7.1): Calibration curve of Gammacell 220 E by using Fricke dosimeter.



Fitting the linear relation between the absorbance and the time of irradiation (Dose) for fricke dosimeter equation (7.1) is determined

Absorbance = 0.0052(time) + 0.0539 (7.1)

The fitting shows excellent correlation coefficient (R^2) of 0.999.

7.2. Results and Discussion of Alanine dosimeter:

The Fricke dosimeter can be used as a reference standard dosimeter; it has measurement range form 40 to 400 Gy.
In this work Alanine (chemical dosimeter) is introduce in the irradiator Lab. of SAEC for the first time. Alanine can be used as a routine dosimetry. From the literature it is found that the Alanine dosimeter can be used for high doses range from 10 to 5000 Gy. (Preparation of Alanine dosimeter is mentioned in chapter (5).
The Alanine dosimeter was irradiated for different irradiation times started from 650 sec to 3300 sec, which are equivalent to doses per sec from 967 to 4910 Gy/sec. These values of the doses were deduced from the results obtained from the fricke dosimeter does measurement (1.48 Gy/sec).
Measurement of the dose rate using Alanine dosimeters was shown in table (7.3). The uncertainty of Alanine dosimeter is 5%:

Table (7.3): Results of Alanine dosimeter measurement

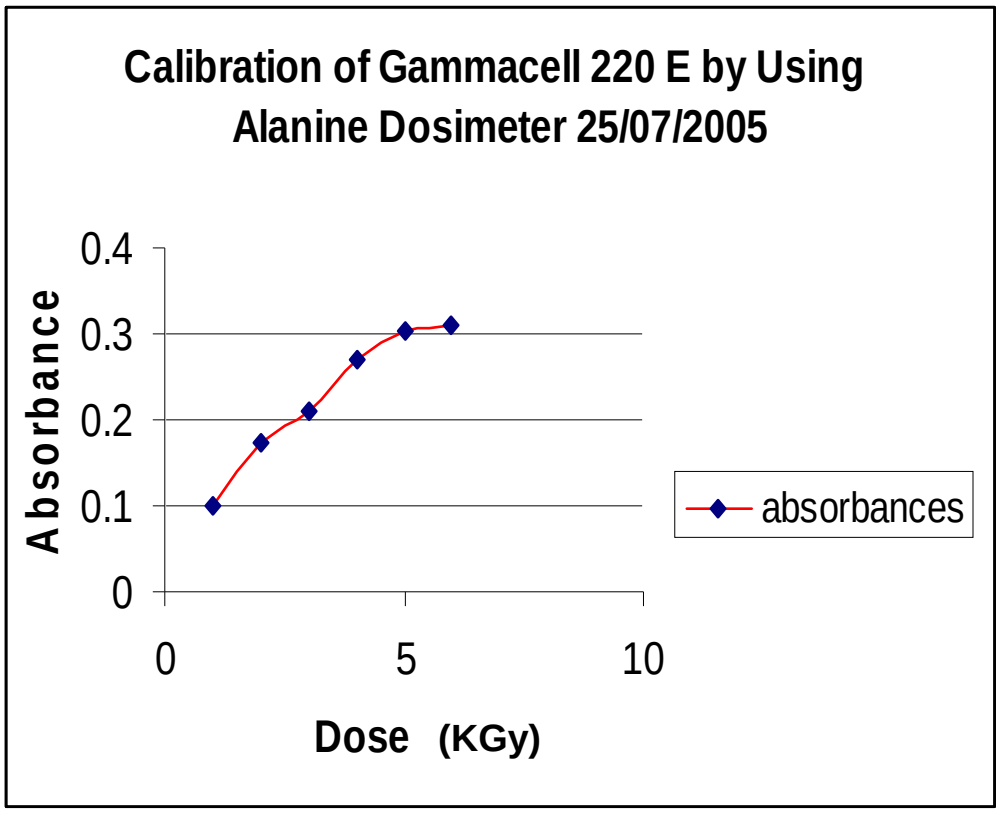
Time (sec)	Absorbance	Diff. absorbance	Average in Diff. Absorbance	STDEV $\sqrt{\sum \frac{Ni - N^{-}}{n-1}}$	Dose (Gy)
650 650 650	0.224 0.225 0.230	0.097 0.098 0.103	0.099	0.003	967.2
1350 1350 1350	0.300 0.301 0.302	0.173 0.174 0.175	0.174	0.001	2232
2000 2000 2000	0.336 0.335 0.335	0.209 0.208 0.208	0.208	0.001	2967
2650 2650 2650	0.400 0.392 0.395	0.273 0.265 0.268	0.269	0.004	3943.2
3300 3300 3300	0.430 0.428 0.429	0.303 0.301 0.302	0.302	0.001	4910.4
3950 3950 3950	0.436 0.435 0.435	0.310 0.309 0.309	0.309	0.001	5877.6

The value of the Blank was found to be 0.126

It can be shown from the table; the standard deviations for the absorbance are small. All readings for the absorbance in table (7.3) were taken after 10 minutes to assure that the irradiated Alanine is completely dissolved in the FX solution.

Plotting the absorbance against the doses from table (7.3), Fig (7.2) was obtained, in this Figure it can be noticed that the absorbance increases with the doses. The relation seams to be linear up to 4000 Gy and then it started to be constant. That means after the dose 4910 Gy the dependence between absorbance and dose is not linear. Therefore for Alanine dosimeter it's useful for doses not exceeding 5000 Gy.

Fig (7.2): Calibration curve of Gammacell 220 E by using Alanine dosimeter.

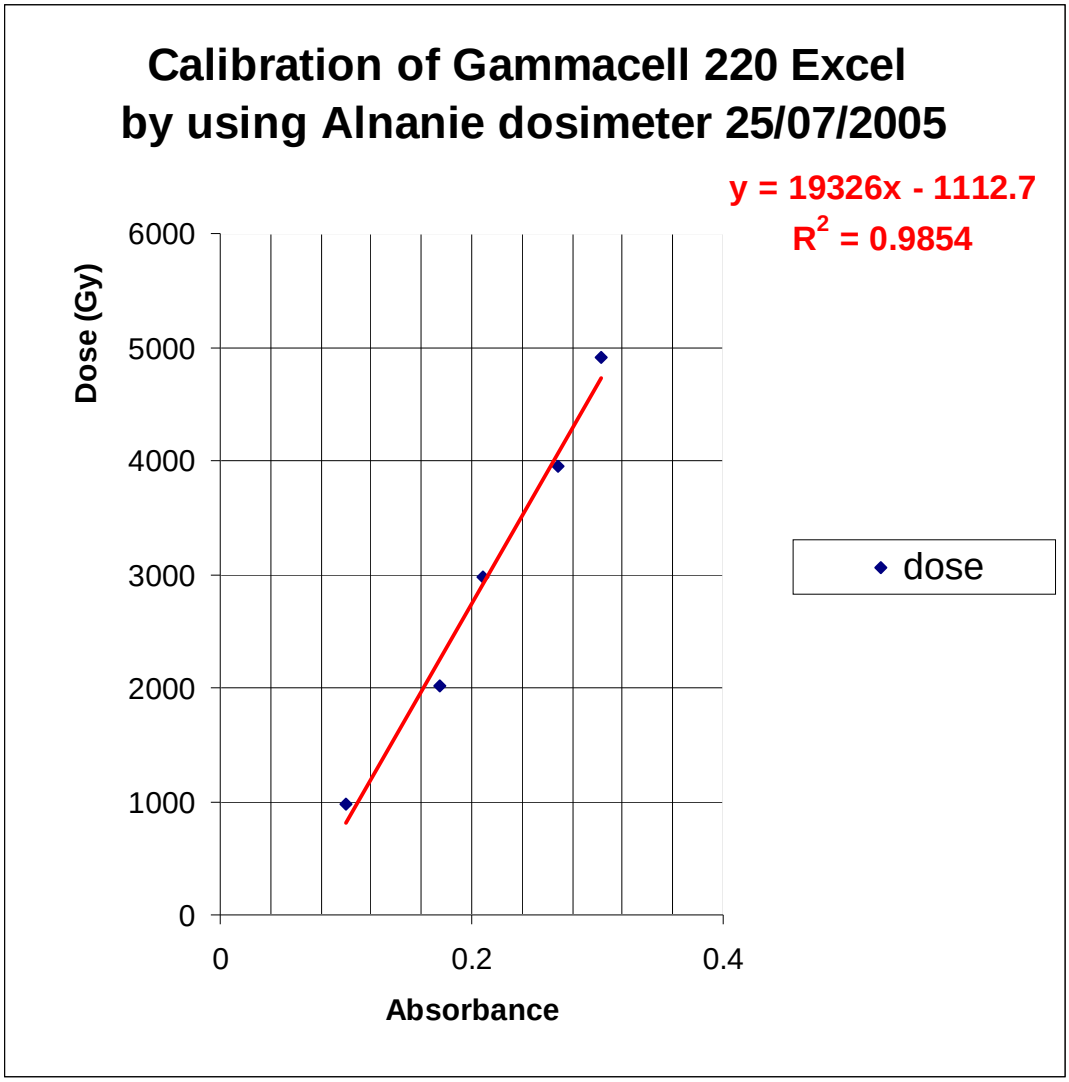


In Fig (7.3) the doses up to 5000 Gy are plotted against the absorbance. The fitting of the data gives linear relation, which fitted to equation (7.2)

Dose (Gy) = 19326 Absorbance - 1112.7 (7.2)

It shows excellent correlation coefficient (R^2) of 0.985.

Fig (7.3): Calibration curve of Gammacell 220 E by using Alanine dosimeter.



(7.3) Comparison of Fricke and Alanine dosimetry:

In comparison of the Fricke and Alanine dosimeters it can be noticed form this work:

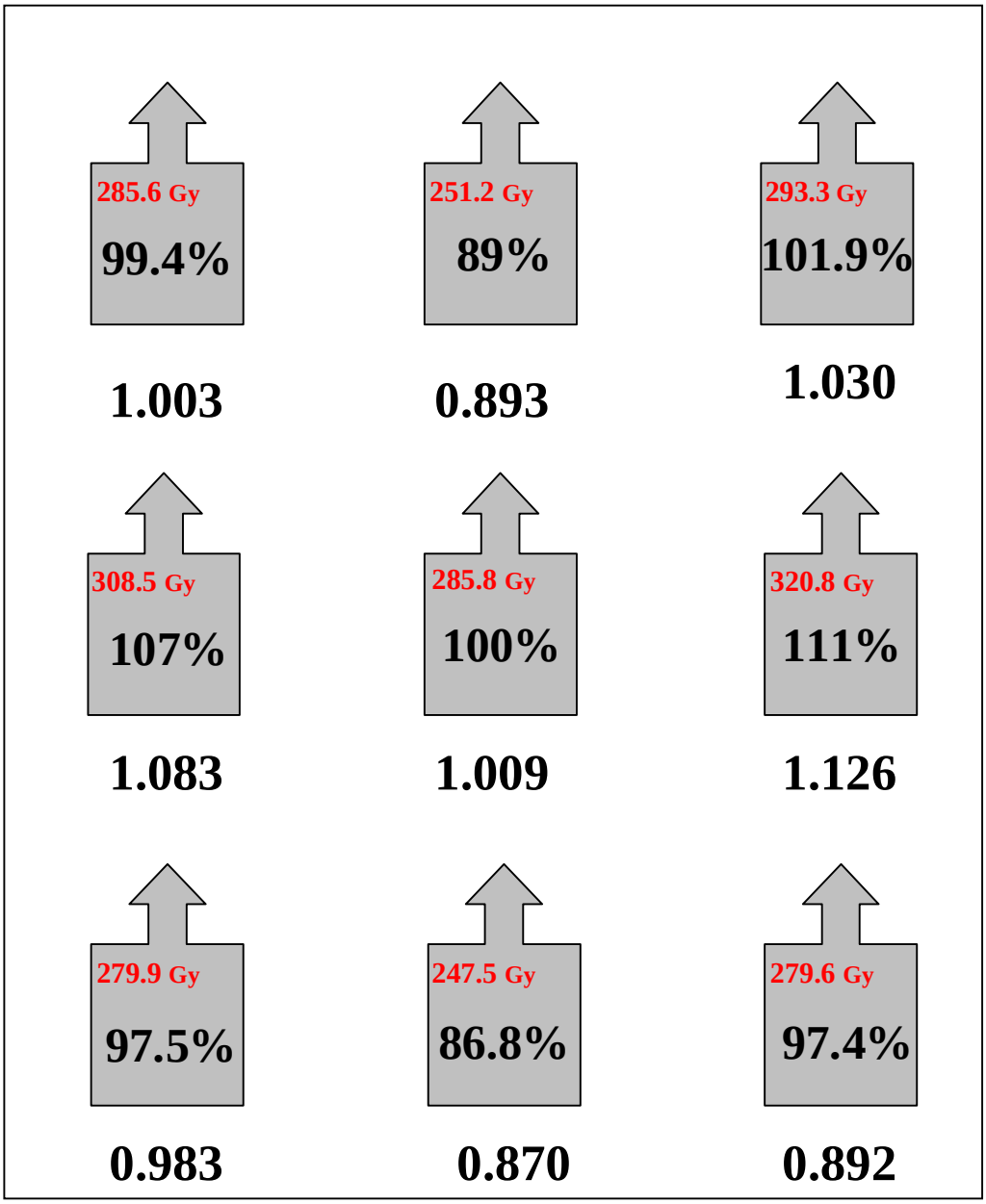
1. The range for Fricke is from 40 to 400 Gy while for Alanine from 10 to 5000 Gy.
2. The chemical preparation of both of them shall be done carefully.
3. The temperature affects both Fricke and Alanine dosimeters.
4. In case of dose determination the Fricke dosimeter is irradiated as solution while the Alanine dosimeter is irradiated as powder and then dissolved in FX solution for measurement.
5. The readings in irradiated Alanine are taken at least 5 minutes after the dissolution in FX solution, while in fricke the reading were taken after irradiation without waiting time.
6. The wavelength of the Fricke dosimeter is 304nm (ultraviolet region); while for Alanine dosimeter it is 525 nm (almost visible region).

7.4 Dose Mapping:

7.4.1 Mapping of dose distribution by using Fricke dosimeter:

In case of sample irradiation using Gammacell 220 it is important to have a good dose distribution inside the chamber of irradiation.
To check for the dose distribution or dose mapping inside the chamber of the cell, fricke dosimeters were placed inside the chamber as seen in Fig (7.4). Dose of 285.8 Gy (irradiation time 191.96 sec) was given to the dosimeters, the dose of each group is written as shown in Fig (7.4). Group at the center give the exact dose, which is 285.8Gy. By normalizing the doses received by each group to the one at the center, the results are shown inside each group.
From these results it can be noticed that for the ones not at the center the doses is sometimes greater than the central one, in other groups doses less than the central one.

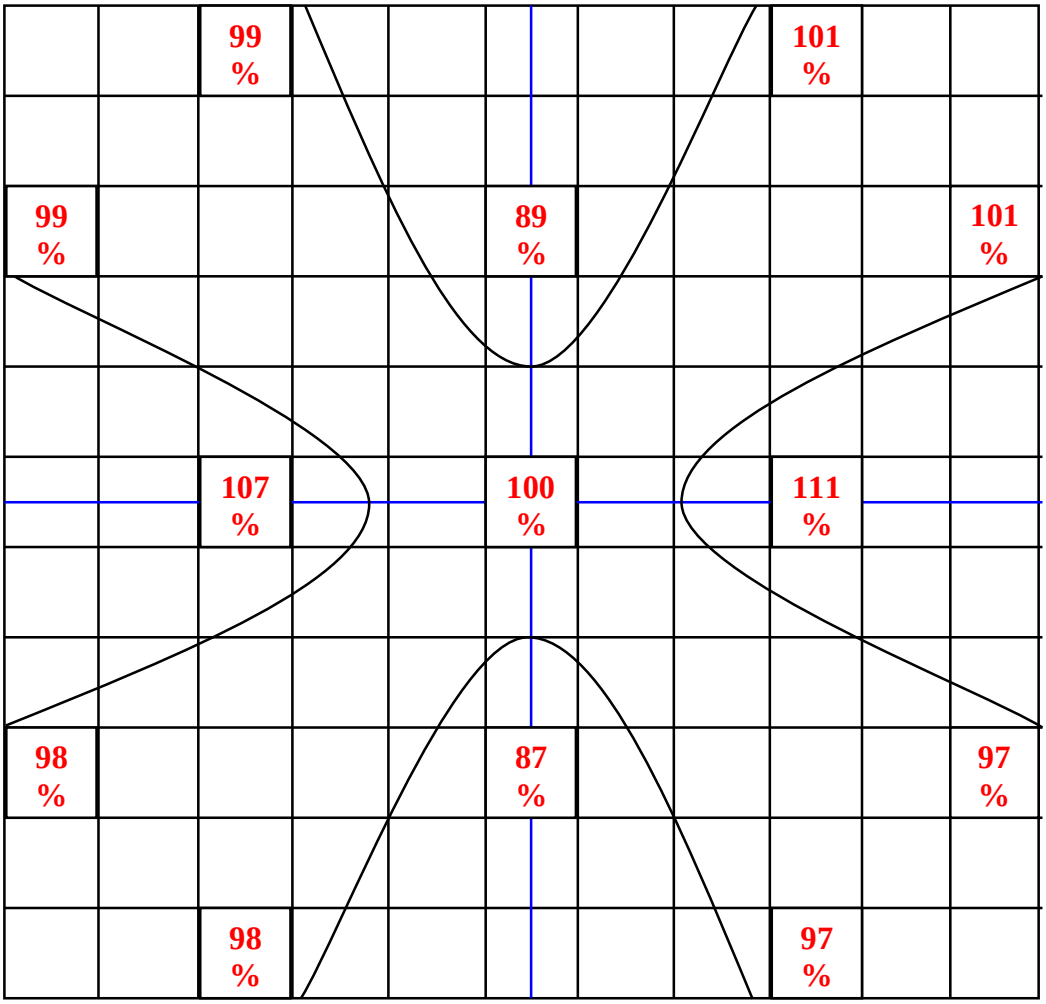
Fig (7.4): Placement of Fricke dosimeters.



23/07/2005
Temperature (T): 29c^0.
The dose is: 285.8Gy.
Time of irradiation: 191.96 sec.
Blank: 0.047.

The results obtained from the doses received by each group of Fricke dosimeters was plotted in Fig (7.5), from the figure it can be noticed that the doses inside the chamber are distributed homogenously.

Fig (7.5) Dose mapping by Fricke dosimeter



The dose distribution determined in this work from Fig (7.5) is compared to the one done for the same type of gammacell by the Nordion Company (the manufacturer) as seen in fig (7.6). Good agreement between the tow doses distribution can be noticed.

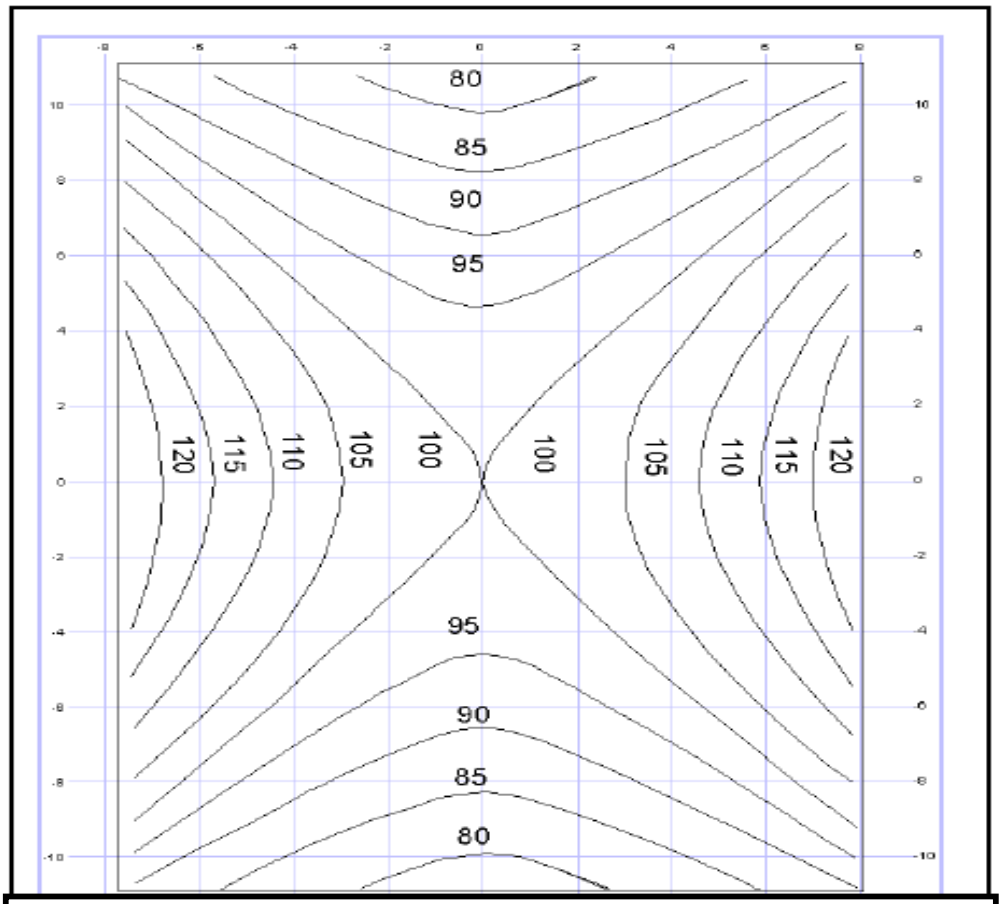
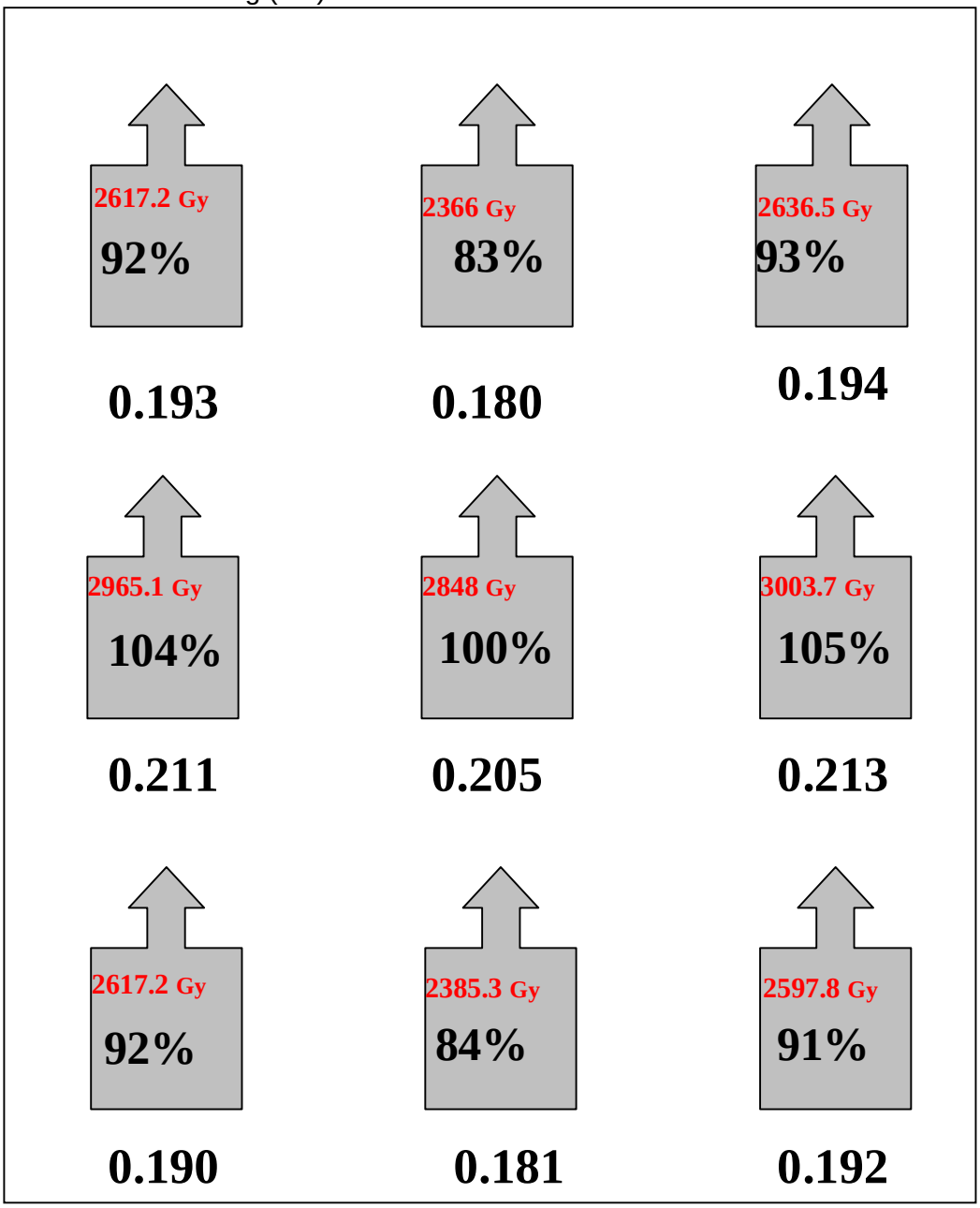


Fig (7.6) atypical dose-rate distribution in the irradiator chamber of the Nordion Gammacell 220 E. the value is normalized to 100 in the center of the Gamma field. Drown from Nordion data. (5)

7.4.2 Mapping of dose distribution by using Alanine dosimeter:

To check the dose distribution for the Gammacell 220 at high doses Alanine dosimeter were used. Different groups of Alanine dosimeters were placed as seen in fig (6.7). A dose 2848 Gy was given to the groups.

Fig (7.7): Placement of Alanine dosimeters.



25/07/2005
Temperature (T): 29c^0. Blank: 0.126.
The dose is: 2848Gy.Time of irradiation: 1914 sec.

By normalizing the dose received by each group in each position to the dose received to the central, it can be noticed that the distribution of doses is homogenous inside the chamber.

The primary purpose of performing dose rate mapping is to verify that the dose variability in the irradiated sample is acceptable for the application on hand and this work should be done before useful irradiation is carried out. If the distribution is winder than acceptable, it points out the need for modifying the irradiation procedure or the container size/shape.

In this work it has been confirmed that the dose distribution in case of high and low doses for Gammacell 220 E of SAEC is acceptable for the applications.

Chapter (8)
Conclusion

(8.1) from the basis results it can conclude that:

- Fricke dosimeter is a good reference dosimetry for Gammacell.
- Alanine dosimeter can be used for dose check and calibration of Gammacell at doses from 150 – 5000 Gy.
- The dose distribution inside the Gammacell chamber is homogenous both at low and high doses.

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