A THEORETICAL AND EXPERIMENTAL DOSE RATE STUDY AT A MULTIPURPOSE GAMMA IRRADIATION FACILITY IN GHANA

This thesis is submitted to the

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by

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IN

RADIATION PROTECTION

JULY, 2015

DECLARATION

I, Tracy Adjeley Sackey, hereby declare that except for reference to other people's work which have been duly cited, this research work being presented for the partial fulfillment of a Master of Philosophy Degree in Radiation Protection is entirely my own under the supervision of Dr. G.K. Banini and Prof. G. Emi-Reynolds, and has neither in part nor in whole been presented anywhere else for any other purpose.

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DEDICATION

I dedicate this work to all teachers and lecturers who influenced my education both formally and informally.

ACKNOWLEDGEMENTS

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LIST OF SYMBOLS & CONSTANTS

λ	Wavelength
h	Plank's constant = 6.626×10^{-34}Js
c	Speed of light = 3.0×10^{8}m/s
B	Build up factor
μ	Linear Attenuation Coefficient
x	Thickness of shielding material
β	Beta
γ	Gamma
$t_{1/2}$	Half life
C_a	Surface Concentration
Γ	Gamma-ray constant
C_i	Surface Activity
α	Differential Dose Albedo

LIST OF ABBREVIATIONS

RTC	Radiation Technology Centre
GAEC	Ghana Atomic Energy Commission
ICRP	International Commission on Radiological Protection and Units
ALARA	As Low As Reasonably Achievable
TLD	Thermo-Luminescent Dosimeter
ECB	Ethanol Chloro-Benzene
RPI	Radiation Protection Institute
NRA	Nuclear Regulatory Authority
IAEA	International Atomic Energy Agency
ISO	International Standard Organization
BNARI	Biotechnology and Nuclear Agriculture Research Institute
GMT	Geiger Muller Tube
SSDL	Secondary Standards Dosimetry Laboratory

ABSTRACT

Radiation dose rate monitoring out at the Radiation Technology Centre (RTC) of the Ghana Atomic Energy Commission (GAEC) to establish the safety or otherwise of staff at the occupied areas is presented. The facility operates a rectangular source of Co-60 gamma with an having activity of 27.4kCi as at March 2015 and has 14 workers. The aim of the research was determine by means of practical and theoretical evaluations shielding effectiveness of the irradiation chamber. This was to ensure that occupationally exposed workers are not over exposed or their exposures do not exceed the regulatory limits of 7.5µSv/h or 50mSv per annum. The study included dose rate measurements at controlled areas, evaluation of personnel dose history, comparison of experimental and theoretical values and determination of whether the shielding can support a. 18.5PBq (500kCi) Co-60 source.

Practical dose rate measurements when the source was in the irradiation position was carried out using a Thermo Scientific Rad-Eye Gamma Survey Meter in the controlled areas of the facility which included the control room, electric room, deionizer room, on top of the roof of irradiation chamber (specifically above the roof plugs) and the two entrances to the irradiation chamber; the personnel door and the goods door. Background reading was found to be 0.08±0.01µSv/h whilst the average dose rates at the two entrances to the irradiation chamber (ie.,- the personnel door and the goods door) were measured to be 0.090µSv/h and 0.109µSv/h respectively. Practical measurements at the roof plugs produced average values of 0.135µSv/h. A particular point on the roof marked as plug-3 produced a relatively higher dose rate of 8.151µSv/h due probably to leakage along the cable to the drive motor. Measurements in the control room, electrical room and deionizer room had average readings of 0.116µSv/h, 0.089µSv/h and 0.614µSv/h respectively. All these average values were below the regulatory limits of 7.5µSv/h and 50mSv/yr. The practical dose rate measurements were consistent with previous measurements.

Theoretical calculations of dose rates in the irradiation chamber were computed using the F-line software provided by the Hungarian suppliers of the facility. It computes dose rates based on the dimensions and parameters of the source and uses the line source approximation method. Respective values obtained for the personnel and goods door, rooftop, deionizer room and outside the chamber were 0.082μSv/h, 0.076μSv/h, 0.080μSv/h, 0.193μSv/h and 0.07 μSv/h which indicates that the theoretical estimations of dose rates were generally lower than the measured values.

Personnel dose (Thermolumiscent dosimeters) history for a period of 12 months (January to December 2013) was analyzed to estimate yearly doses received by radiation workers of the facility. Collective doses of Hp(10); 2.22mSv and Hp(0.07); 2.29mSv were obtained from the analysis. They were well below limits approved by the regulatory authority.

CHAPTER ONE

INTRODUCTION

1.1 BACKGROUND

Ionizing radiation has been proven over the years to have a great positive impact on man via different aspects of science due to the high energy it possesses. The medical sector is a major beneficiary of ionizing radiation. It permits non-invasive vision of various body parts prior to or during surgical procedures or may prevent them altogether. It is also used in the treatment of cancer via radiotherapy, brachytherapy or nuclear medicine (Medical Uses, 2007). Industrially, radiation may be used to check for defects and cracks in huge pipes and columns, measure density and moisture in soil and even for mining (Radioactive Source Reduction and Management, 2014). Agriculturally, irradiators are used to delay crops and foodstuffs from sprouting and ripening (Adu-Gyamfi, 2011), sterilize various harmful insects to prevent them from producing offspring (Joint FAO/IAEA Programme) and treat sewage and sludge before recycling (Al-Ani and Al-Khalidy, 2006) or discharging them into the environment. Nuclear power production is another use of nuclear energy and which is currently being used by some countries to boost their industries (WNA, 2015). Archaeology and space exploration are other areas that make use of radiation (Hong Kong Observatory, 2003).

Diverse sectors that utilize radiation require specific kinds of radionuclides or radiation generators. The choice is dependent on the type of radiation required and activity of source sufficient to achieve the needed results. Likewise, each field requires distinct protective measures in place to ensure that the radiation workers and the environment are properly protected and prevented from unnecessary exposure in the cases of both normal operation and during emergencies.

Radiation dose and exposure have a direct relationship and are both dependent on the energy of the photons that make up the radiation beam (Cember & Johnson, 2009) and the duration of exposure. Doses received over a period of time produce its time derivative called dose rate (NRC, 2015). At a particular location, it may be judged by international dose limits or as prescribed by the Radiation Protection Board to establish whether there is or may have been over exposure.

Gamma irradiation facilities use extremely high doses of radiation to achieve the required results (Agarwal, 1998). Construction and shielding of such a facility should ensure that during normal operation and maintenance, workers are not overly exposed to radiation and there is insignificant public exposure (IAEA-1454, 2010).

At the Radiation Technology Centre (RTC) of the Ghana Atomic Energy Commission (GAEC), high doses of gamma radiation are used to process crops and foodstuffs mainly for export and to sterilize medical equipment such as syringes, intra-venous infusion sets, and gausses without destruction of their packages (Adu-Gyamfi, 2011). Primary shielding of the source when not in use is deionized water in a pool of 5.7m deep. Concrete walls and lead doors make up the barriers of the irradiation chamber. The dose and dose rates at different positions in the facility are dependent on the locations distance from the source and the shielding design and thickness of the irradiation chamber. Adequate shielding attenuates direct exposure of workers and the public and is essential to ensure optimization of protection (IAEA SSG-8, 2010)

Occupational monitoring of the radiation workplace and individuals is a tool used by radiation protection experts to determine whether or not dose limits are or may be exceeded. It is therefore of paramount importance to have information on dose rates at different locations in the facility which can also enhance decisions on the demarcation of boundaries and restrictions for controlled and supervised areas (IAEA SSG-8, 2010). Area monitoring entails taking and interpreting measurements in the assessment and control of external exposures.

International Commission on Radiological Protection and Units (ICRP 60) gives the conditions for conducting either workplace or individual monitoring. According to them;- any person who normally or occasionally works in a controlled area in a radiation practice and may receive significant exposure shall be provided individual monitoring where appropriate, adequate and feasible. Where individual monitoring is inappropriate, inadequate or not feasible, occupational or area monitoring shall be based on workplace monitoring and information about locations and durations of worker exposure. Workplace monitoring takes into consideration external dose rate and contamination whiles individual monitoring measures cumulative individual doses.

The second principle of radiation protection;- optimization of protection is meant to ensure that in practices, the number of people exposed, the magnitude of individual doses and the likelihood of incurring exposures shall be kept As Low As Reasonably Achievable (ALARA) economic and social factors being taken into consideration (General Principles for The Radiation Protection of Workers, 1997). Maximizing distance from a source, limiting time spent with a source and the provision of adequate shielding are the three (3) basic rules which helps to fulfill the above stated principle and positive results of radiation protection.

In this research workplace and individual/personnel monitoring were carried out to determine dose rates at different areas in the vicinity to ascertain the safety of personnel. The F-line code was used to project dose rates at the two maze entrances with the current source activity and a source activity of 18.5PBq (500kCi).

Previous monitoring conducted at the facility occurred when the facility was loaded with 1.85PBq (50kCi) Co-60 source in a cylindrical configuration/geometry. Measurements were taken within the maze entrances and outside the lead doors, in the control, electrical and deionizer room using a Chinese made dose rate survey meter model SG-102 (Emi-Reynolds et al. 1997). Theoretical calculations of dose rates before the change of source have been carried out by Emi-Reynolds & Akaho (1994) using the ANISN code and by Fletcher et al (2000). Measurements have also been carried out by Thembinkosi (2012) using Thermoluniniscent dosimeters (TLD), Ethanol-Chlorobenzene (ECB) and Fricke dosimeters.

The current work is a more detailed study of the facility and measuremtnts have been compared with previous theoretically and practically obtained values.

1.2 STATEMENT OF PROBLEM

The initial cylindrical source used at the Radiation Technology Centre (RTC) of the Ghana Atomic Energy Commission (GAEC) was changed to a rectangular rack or plaque source with an activity of approximately 1.85PBq (50kCi) in August 2010. The present estimated activity as of 2015 is 1.014PBq (27.4kCi). This change was meant to upgrade the source strength whose activity had decreased and was no longer economical enough for radiation processing. Secondly it was aimed at improving the uniformity ratio of irradiated materials. Theoretical calculations were made prior to the installation of the plaque source to predict dose rates at various sections of the facility. Shape, orientation and position of a source have an impact on dose rates. After installation of the new source, no comparative study focusing on both theoretical calculations based on the exact specifications of the source pencils as well as experimental measurements have been carried out. One of the objectives of the present study is to investigate whether the dose rates at occupied areas both for the controlled and supervised areas are acceptable and do not exceed the dose limit of 50mSv per year for occupational workers.

The shielding provisions at the RTC were designed to allow a maximum loading capacity of a source with an activity of 18.5PBq (500kCi). Activity and dose rate have a positive linear relationship therefore any plans for an increase in activity implies that computations must be made to certify that shielding of the irradiation chamber is sufficient to cater for the upgrade.

This study therefore seeks to carry out theoretical calculations using the F-Line software based on the specifications of the source and irradiation chamber, and practical measurements to verify that doses received by personnel of the facility using the 1.85PBq (50kCi) rectangular rack source are within limits and acceptable.

1.3 OBJECTIVES

The main aim of this research is to find out using a combination of practical and theoretical methods, whether the shielding infrastructure in place at the RTC are sufficient to ensure that workers of the facility are not over-exposed and do not exceed the limits of 7.5μSv/h or 50mSv per annum.

Specific objectives of the research include;

1) Dose rates measurements at controlled areas in the facility.
2) Evaluating dose history of personnel over a period of one year.
3) Performing theoretical calculations of dose rates based on the line-source approximation method and using the current activity of the source (27.4kCi), at the maze entrances.
4) Comparison of experimental and theoretical values with previous calculations.
5) Predicting whether sufficiency of present shielding in place is capable of supporting a source activity of 10 times that initially loaded , i.e. 18.5PBq (500kCi).

1.4 RELEVANCE AND JUSTIFICATIONS

Radiation effects categorized under stochastic and deterministic effects differ in the absence and presence of a threshold limit. Stochastic effects which include cancer may occur for any small amount of radiation, especially when received for a long period of time, irrespective of how minute it is, however, the probability of contracting a stochastic effect increases linearly with an increase in dose. Deterministic effects can only occur above a threshold and include hair loss, skin burns, et cetera and happen most likely in the instance of accidental exposure. The severity of the effect increases with an increase in dose.

The occurrence and severity of both effects increase with an increase in exposure to ionizing radiation, hence can be managed and reduced when adequate measures of shielding and radiation protection are in place. Normal and Accidental exposure can also both be minimized with adequate shielding

The research may confirm the adequacy of shielding of the irradiation chamber. It will hence provide information to the Ghana Radiation Protection Institute (RPI) and the Nuclear Regulatory Authority (NRA) on the safety of personnel and the facility. It will also provide the occupational exposed workers of the facility with verified data on radiation levels at various points making them assured that they are adequately protected. Furthermore, results obtained may be used to further demarcate boundaries of controlled and supervised areas.

1.5 SCOPE

This research focuses on taking confirmative measurements of dose rates around a category IV wet storage irradiation facility of the RTC using a Rad-Eye G-10 Gamma Survey Meter.

Dose history of individual occupationally exposed workers was also taken into consideration for assessment and comparison of results.

Theoretical dose rates prediction was made using a computer code, F-line supplied by the manufacturers of the source.

1.5.1 STRUCTURE OF THE THESIS

Chapter one deals with the introduction to the thesis, objectives, relevance and justification of the work and the scope of the research. It also lays out the organizations of the chapters. In Chapter two, literature review of the subject matter is presented including the different type of gamma irradiators used in radiation processing. The methodology used in this work is presented in Chapter three. Chapter four presents the discussion of results from the study. Chapter five gives a conclusion of the research and appropriate recommendations.

CHAPTER TWO

LITERATURE REVIEW

In this chapter, the subject matter in relation to gamma irradiation facilities and protection of their radiation workers is outlined briefly.

2.1 GAMMA RAYS

Gamma rays are produced from the nucleus of certain isotopes as they de-excite to achieve stability. They usually follow the emission of a beta particle (US EPA Radiation Protection, 2013). They are indirectly ionizing and are similar in behaviour to x-rays but differ in their mode of origin. They form part of the electromagnetic spectrum and have a very short wavelength of $<1\times10^{-12}$ nm (Terr, 2015).

Planck's energy equation, i.e.; equation 2.1, relates the wavelength of a photon to its energy. A very short wavelength implies a very high energy and vice versa.

$$E = (hc)/\lambda \qquad (2.1)$$

Where E is the Energy, h is Planck's constant (6.626×10^{-34} Js), c is the speed of light ($\sim3.0\times10^{8}ms^{-1}$) and λ is wavelength (in meters).

Gamma rays hence have a high frequency and penetrating power and do not induce radioactivity into the irradiated material.

2.1.1 GAMMA IRRADIATION FACILITIES

An irradiator may be defined as a radiation device in which a prescribed or known radiation dose is applied to a specific item to achieve a particular purpose in such a way that occupational and public exposure is kept to a minimum (Agarwal, 1998).

There are over 180 gamma irradiation facilities worldwide (Agarwal, 1998). They provide services such as food irradiation, sterilization of medical equipment, sterilization of insects, and treatment of sewage and sludge.

On an industrial scale, Cobalt-60 and Cesium-137 are the most common gamma emitting radionuclides used and their source activities range from 27Ci to over 3MCi depending on the type or category of irradiation facility in use (IAEA-1454, 2010). However ^{60}Co is more popular due to its higher energies. The ^{60}Co isotope is a beta-gamma emitter with a half-life of 5.27years and energies of 1.332MeV and 1.173MeV and an average of 1.25MeV, as compared to ^{137}Cs with energy of 0.66MeV.

2.1.2 CATEGORIZATION OF GAMMA IRRADIATION FACILITIES

Irradiator source activities, design and specific uses all differ because of the various achievements they are meant to accomplish. Irradiators are categorized into four main groups depending on the activity of source, design of the facility, accessibility and shielding of the radioactive source.

Categories I and II are both dry storage irradiators however category I is self-contained whiles category II is panoramic. Their sources are stored in dry shielding containers. The design setup of sources in category I do not permit access to the radiation source and the item undergoing irradiation when in use. Category II (Figure 2.1) is shielded when not in use and is exposed in an enclosed area of space when required for irradiation using a control system.

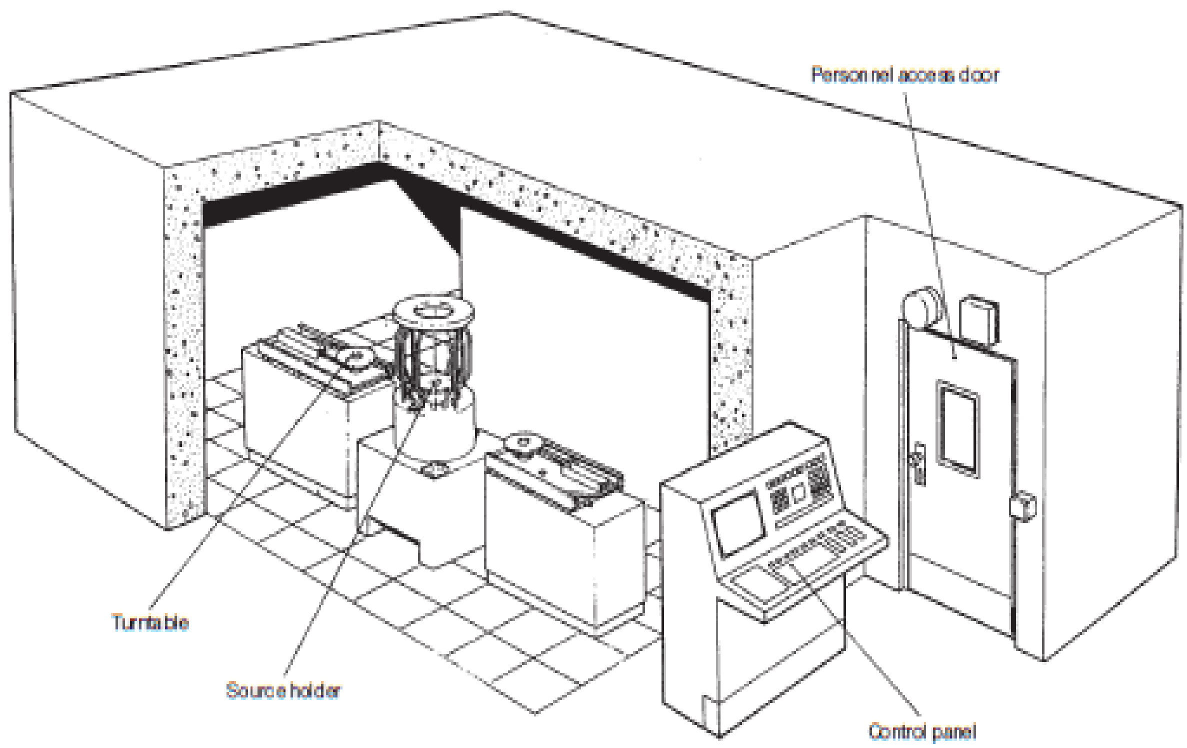


Fig 2.1 Category II Gamma Irradiation Facility: Panoramic Dry Source Storage Irradiator (IAEA – 1454, 2010)

Categories III and IV are wet storage irradiators with category III (Figure 2.2) being self-contained and category IV (Figure 2.3) being panoramic. In both cases the sealed source is shielded in water. Likewise, the design configuration and appropriate mode of use of category III facilities restrict access to the source and size of goods undergoing irradiation. An entry control system determines access to the source and irradiation chamber in category IV facilities (IAEA-1454, 2010).

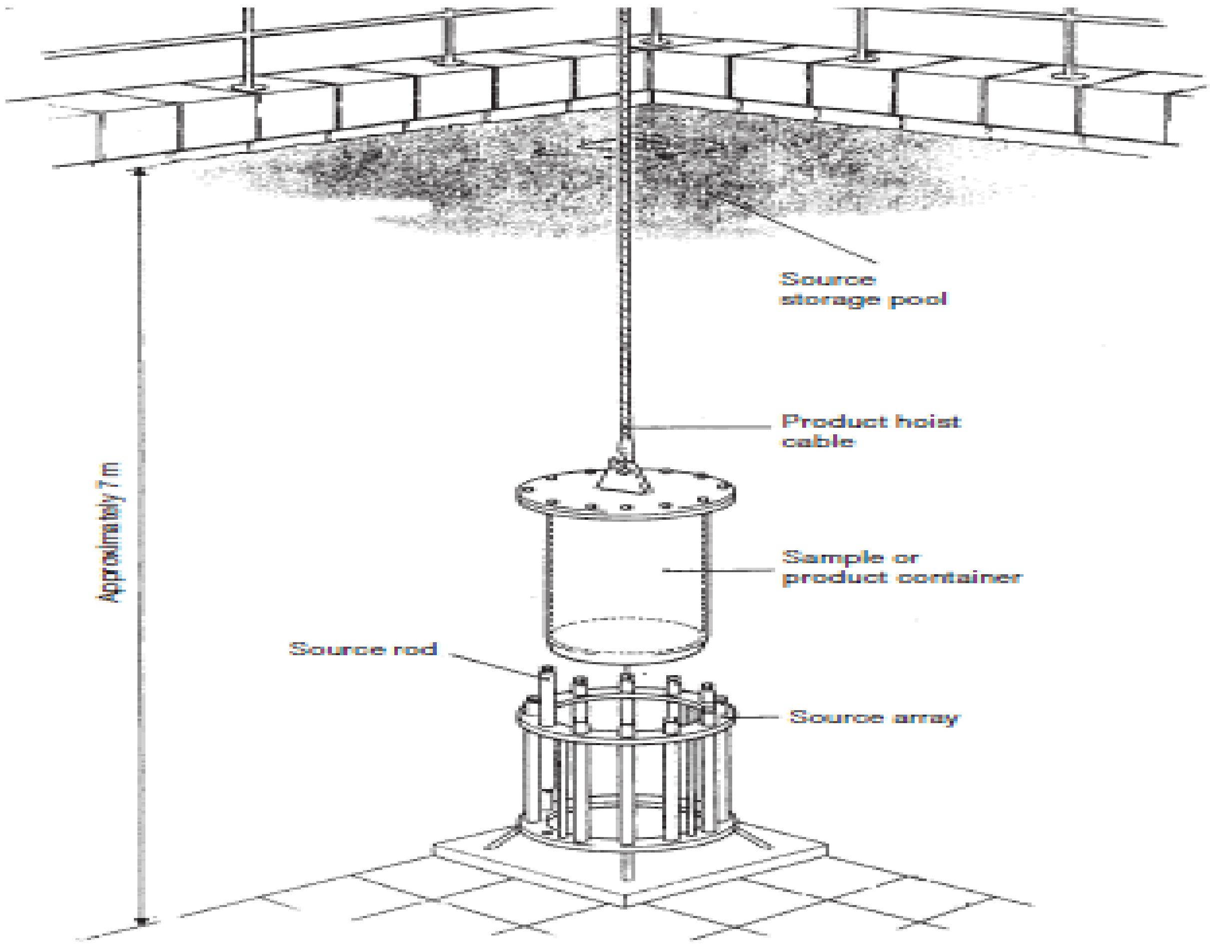


Fig 2.2 Category III Gamma Irradiation Facility: Underwater Irradiator (IAEA – 1454, 2010)

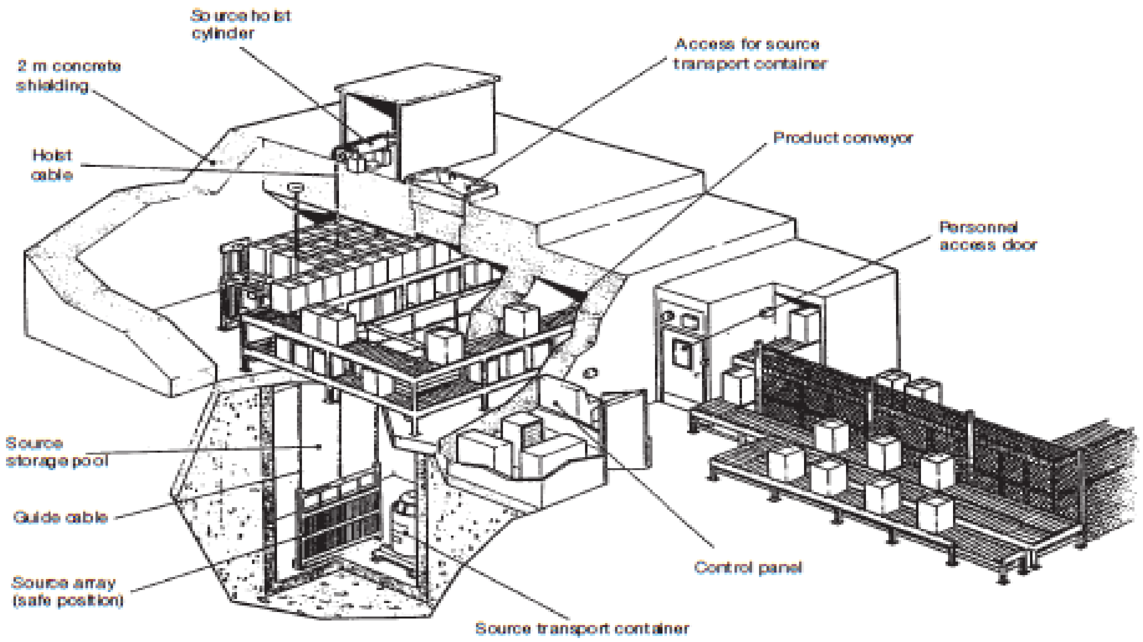


Fig 2.3 Category IV Gamma Irradiation Facility: Panoramic Wet Source Storage Irradiator (IAEA- 1454, 2010)

2.2 CONTAINMENT AND SHIELDING OF GAMMA RADIATION SOURCES

2.2.1 RADIATION ATTENUATION

Gamma rays loose energy and are attenuated via three main methods namely, i) Compton scattering, ii) pair production and iii) photo electric effect.

i) Compton scattering occurs when an incident photon collides with an electron and loses some of its energy to it. Both the photon and electron are deflected in different angles dependent on the energy of the incident photon. Compton scattering, $\sigma\sigma_c$ is governed by the equation;

$$\sigma_c = z\sigma_{ce} \quad (2.2)$$

Where σ_{ce} is the electronic Compton scattering cross-section and Z is the atomic number of the shielding material.

ii) Pair production takes place when photons with energy in excess of 1.02MeV break down and reappear as a positron-electron pair as they pass near a nucleus. Each particle ends up having energy of 0.51MeV. Its cross section is denoted by σ_p.

iii) Photoelectric effect simply refers to the ejection of an electron from its orbit by a photon whose energy supercedes the binding energy of the electron. This interaction results in the disappearance of the photon and all its energy transferred to the ejected electron, hence a vacancy is created in the shell of the atom. The photoelectric effect cross section, σ_f is based on equation 2.3;

$$\sigma_f = 10^{-37} z^5 / (hv)^{7/2} \quad (2.3)$$

Where z is the atomic number and hv represents the photon energy.

When gamma rays come into contact with a material, they undergo either one of these interactions. Hence the total cross-section, σ_T of interaction of gamma rays with the atoms of a material is dependent on equation 2.4 (Poskus, 2012);

$$\sigma_T = \sigma_c + \sigma_f + \sigma_p \quad (2.4)$$

Radiation attenuation refers to the extent to which energy of an incident radiation is reduced upon interaction with a material (McAlister, 2015). It consists of the absorbed and/or scattered radiation. The primary radiation which directly penetrates the shielding material and the secondary radiation resulting from any of the above interactions both contribute to the total radiation at a point. The secondary radiation is catered for using a correction factor called the buildup factor. Several buildup factor formulas exist such as the Capo, Berger, Linear and Taylors build up factors (Jaegar R.D et al, 1968).

The buildup factor, B is a ratio of the total (primary and secondary) photons at a point to the primary photons at that same point (Cember & Johnson, 2009). It is dependent on the mean free path (mfp) of a material which is calculated using its linear attenuation coefficient, μ and thickness of the shielding material, x as in equation 2.5.

$$mfp = \mu x \qquad (2.5)$$

Each shielding material has an extent to which it may attenuate radiation dependent on its buildup factor.

2.2.2 CONTAINMENT OF GAMMA RADIATION SOURCES

Adequate shielding of sources begins with their containment. ISO Standard 2919 specifies the requirements for the containment of sealed sources. The containment takes into consideration the possibility of a fire outbreak, corrosion and explosion. Factors taken into account by manufacturers include;

1) Consequences of failure of source integrity influenced by the quantity, radiotoxicity, leachability, solubility, chemical and physical form of the radioactive material.
2) The environment in which the source is stored, moved and used.

There are specific requirements for sources in categories III and IV which use wet storage. Containment for such sources should be extra-resistant to corrosion when exposed to the water storage conditions of the pool such as temperature, and the source itself should not dissolve easily in water.

2.2.3 SHIELDING OF GAMMA RADIATION SOURCES

Radiation shielding refers to physical barriers and technology designed to provide protection from the effects of ionizing (and non-ionizing) radiation. It is meant to achieve the protection of humans and in some instances materials and equipment.

In situations where the immediate containment and shielding of the source fails, the onus lies on the shielding of the irradiation chamber to prevent over-exposure and contamination of radiation workers and surroundings.

Adequate and appropriate shielding of irradiation facilities is an uncompromising necessity that requires upmost attention because it limits the amount of exposure workers receive. Shielding specifications of an irradiation facility are subject to dose rate requirements as established by the competent authority. The shield openings allow for entry of personnel and goods and for airing and ducting, these may give rise to higher radiation levels at those points (Nuclear Safety Technical Assessment Guide, 2013). To account for these openings, the application of a maze system and shield plugs helps reduce the radiation measurements at the locations to required levels. Shielding of a gamma irradiation facility hence requires calculations undertaken by specialists (IAEA-1454, 2010).

2.2.4 RADIATION SHIELDING MATERIALS

Shielding materials are chosen based on a number of chemical and physical properties they possess such as their resistance to damage, heat and cost efficiency. The main factors however are its atomic number, the density of the material and how well it is able to attenuate photons which fall in the category of indirectly ionizing radiation.

Shielding materials include water, concrete, and dense metals such as lead and tungsten (Materials Used in Radiation Shielding, 2015). In most cases a combination of these materials which provide adequate shielding are used.

2.3 RADIATION TECHNOLOGY CENTRE IN GHANA

The RTC is an extension under the Biotechnology and Nuclear Agriculture Research Institute (BNARI) of GAEC. The facility caters for sterilizing medical equipment and irradiating foodstuffs to delay ripening. They also perform insect disinfestation, microbial decontamination and sprout inhibition. It is graded as a category IV wet storage irradiator having three main sections, the transport section, control room and dosimetry section as in Figure 2.4 (Safety Manual for Radiation Technology Centre, 2010)

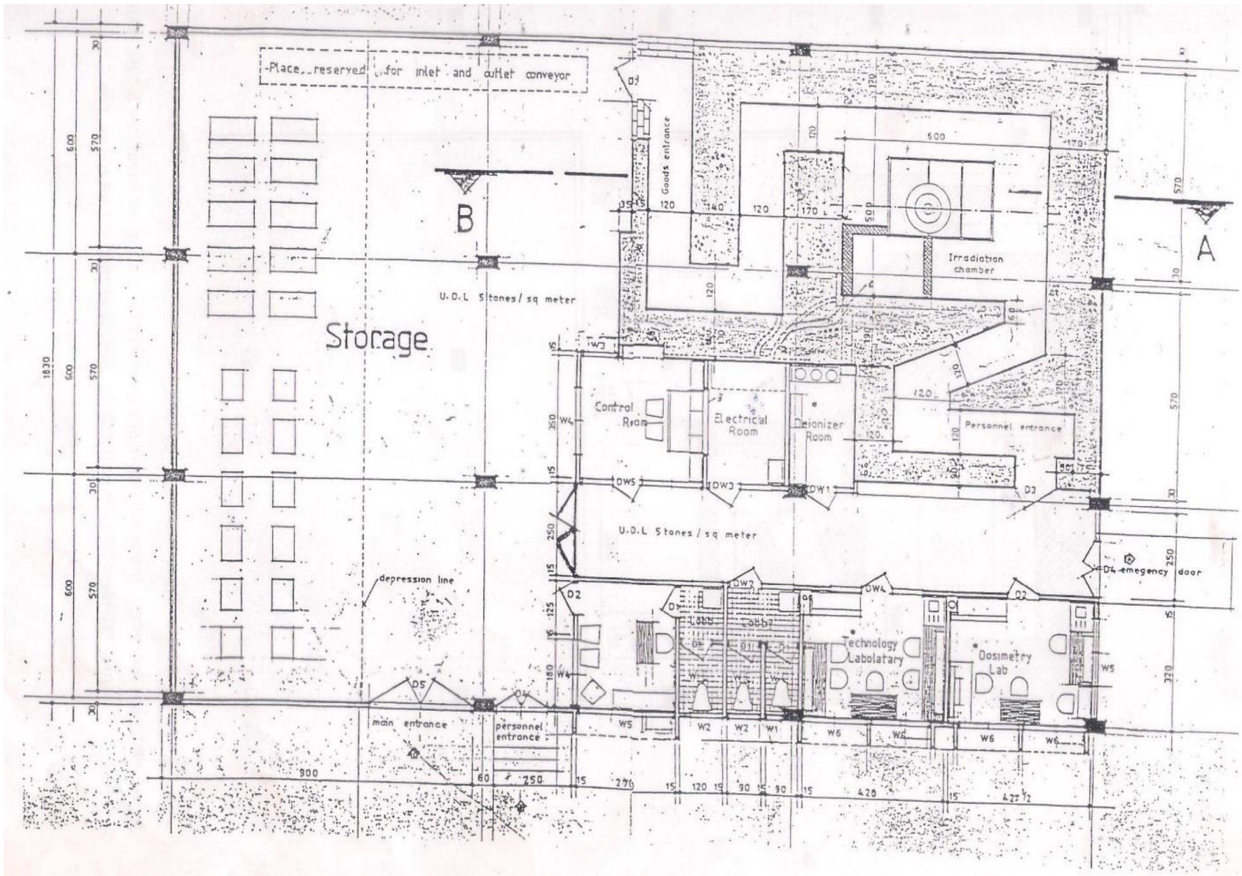


Fig 2.4 Ground Plan of Radiation Technology Centre at GAEC (Safety Manual for Radiation Technology Centre, 2010)

2.3.1 CONTROL SYSTEM

When the source is required for use, a pneumatic hoist mechanism raises it from the rest position to the irradiation position within the irradiation chamber. A transport mechanism conveys the product from the storage facility to the irradiation chamber by means of a tote box. The tote boxes are moved by a tote box car when a command is issued. In all there are 13 tote boxes managed by the control system.

The control system is made up of a Programmable Logic Controller (PC-PLC) based electronic system with operating software that controls the detectors, the ventilation and pressurized air system and mechanical units.

Sequentially interlocked controls are provided for personnel access, radiation chamber lockup sequence and source exposure operations, thereby enhancing safety to operators and security of the source.

2.3.2 DOSIMETRY

A dosimeter may be defined as a device that directly or indirectly measures or evaluates quantities of ionizing radiation (absorbed dose, equivalent dose, exposure and Kerma) and their time derivatives (rates), or related quantities of ionizing radiation (Izewska & Rajan). A dosimeter along with its reader is referred to as a dosimetry system. Measurement of a dosimetric quantity is the experimental process of finding the value of the quantity using dosimetry systems (Izewska & Rajan).

The dosimetry section at RTC accounts for all doses materials sent for irradiation receive at the facility. Dosimeters in use include the ECB (Ethanol Chlorobenzene) dosimeter, which has a range of 10Gy to 2MGy (ISO/ASTM-51538, 2009) and the Perspex (Red) dosimeter which has a range of 5kGy to 50kGy (Fernandez et al, 2015).

2.3.3 COBALT-60 SOURCE AT RTC

Cobalt-60 is a β-γ emitter which decays to stable Nickel-60. Its rays are able to penetrate even metal and also permits it to treat packaged goods or fluids moving through a column. It can be secured inside a sealed container in its chemically neutral form without requiring much maintenance hence is appropriate for industrial use.

The centre uses a ^{60}Co plaque (rectangular tablet) cage type source having an installation activity of approximately 1.85PBq (50kCi) as at August 2010. It was manufactured and installed by the Department of the Institute of Isotopes of the Hungarian Academy of Sciences, Budapest, Hungary. ^{60}Co has a half-life ($t_{1/2}$) of 5.27years and obeys the decay relation, Equation 2.6.

$$A = A_0 e^{-\lambda t} \tag{2.6}$$

Where A is the current activity of the source, A_0 is the activity at the time of loading and λ is the decay constant given by

$$\lambda = ln2/t_{1/2} \tag{2.7}$$

Using the above two equations the current activity of the source is approximately 1.014PBq (27.4kCi).

A pool of water (dimensions (2.0×3.0×5.7) m), 4.0m thick surrounds the source when in the rest position and serves as biological shielding mechanism (Emi-Reynolds & Akaho, 1994). The ions in the pool water are monitored and removed using a de-ionizing unit to prevent corrosion and promote visibility.

The source consists of pencils type Co43HH stored on a rectangular stainless steel source rack with surface area (1×1) m^2 divided into 4 modules as described in Figure 2.5 (Safety Manual for Radiation Technology Centre, 2010). Each module has 44 slots to hold source pencils. 17 active pencils of varying activities are arranged in each module with the 27 other slots holding dummies with no activity.

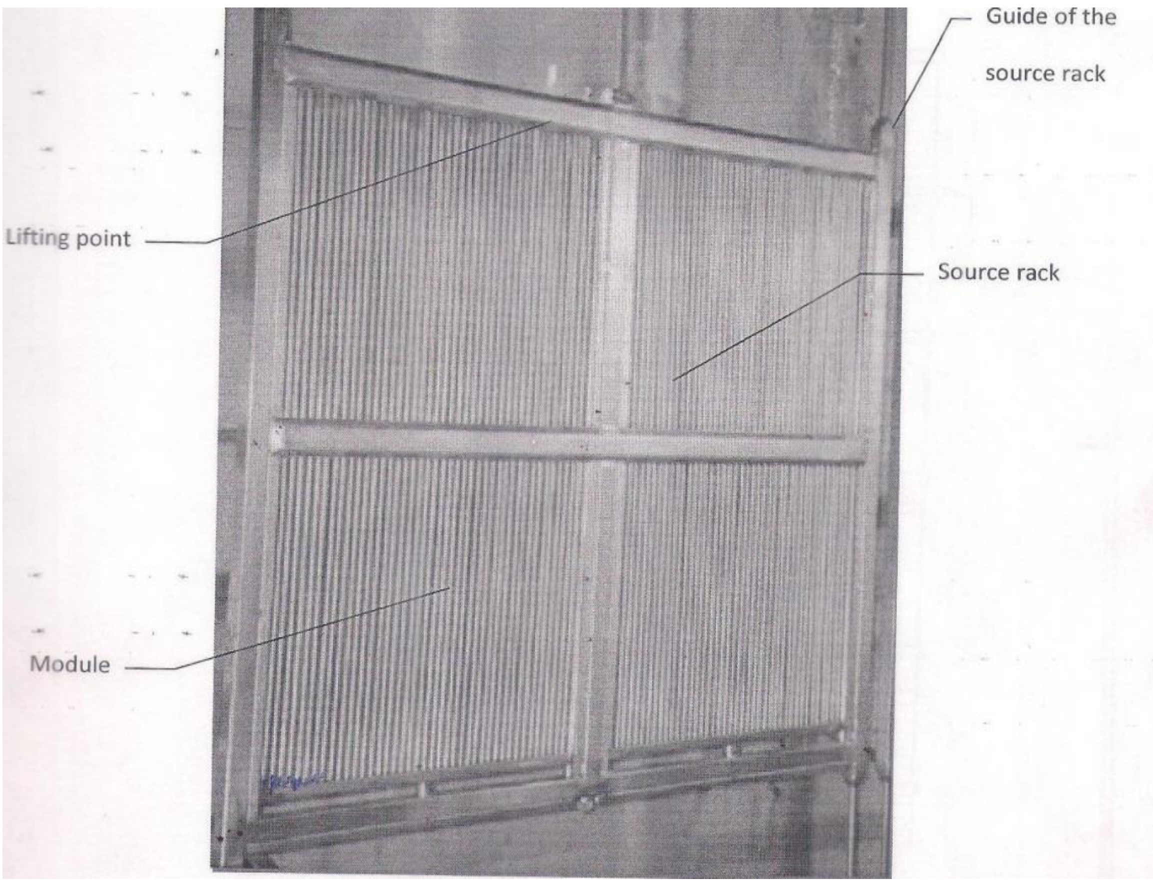


Plate 2.1 Co-60 Stainless Steel Source Rack at Radiation Technology Centre (Adapted from Safety Manual for Radiation Technology Centre, 2010)

2.4 PERSONNEL SAFETY & RADIATION PROTECTION

Radiation Protection as defined by the IAEA is the protection of people and the environment from the harmful effects of ionizing radiation. In every justified practice, measures should be put in place to ensure that the radiation risk, hence occupational exposure is limited.

These measures are dependent on factors including;

- ✓ Type of radiation
- ✓ Activity of the source
- ✓ Doses being applied

The primary aim of radiological protection is stated as providing the right standards of protection for man without unreasonably limiting the beneficial practices giving rise to radiation exposure (ICRP 60). Protection may be optimized by taking steps and using resources to ensure that doses are kept ALARA with social and economic factors being taken into consideration.

Optimization of protection is a course mainly source-related and preferably should initially be implemented at the design stages of building or setting up a radiation practice.

The International Commission on Radiological Protection recommends that the operations of practices should be designed such that any unnecessary or unneeded exposure should be prevented and furthermore, all justified exposures should be kept ALARA (Cember & Johnson, 2009).

Several measures are in place at the RTC to ensure the protection of workers as well as the safety and security of the source.

They include;

- ✓ Automatic Warning lights
- ✓ Motion sensors
- ✓ Interlock Security and Alarm Systems
- ✓ Lead door shielding at entrance of irradiation chamber
- ✓ Test source at entrance of irradiation chamber

The presence and efficiency of biological shields in the design and construction of an irradiation facility is a major factor that has an effect on the safety of personnel and equipment in working areas. The shield must effectively (Emi-Reynolds & Akaho, 1994);

1. Protect personnel working in the irradiation chamber when source is in the storage position.
2. Take account of the operators and other workers who may stay close to the maze entrances or adjacent walls.

2.4.1 RADIATION MONITORING

Radiation monitoring is defined as the measurement of radiation levels, concentration or quantities of radioactive material and the use of the produced results to evaluate potential exposures and doses (Radiation Safety Manual, 2015). It is necessary to carry out radiation monitoring anytime radiation sources are being processed, handled, used, held, stored, moved, transported or disposed (Workplace Monitoring for Radiation and Contamination, 2004). Radiation exposure can either be internal or external. In the case where sealed sources are used, the most likely type of exposure is external exposure. Radiation monitoring aims to achieve three main goals (Izewska & Rajan, 2005):

1) To assess workplace conditions and individual exposures.
2) To ensure that the radiological conditions in the workplace are safe and satisfactory.
3) To keep records of monitoring which may be used for regulation or as good practice.

Radiation monitoring at the workplace may be subdivided dependent on where the measurement is taking place. Workplace monitoring is carried out in the workplace and individual monitoring is carried out using measurements taken for radiation/ radionuclides in or on the individual occupational exposed worker.

It may be conducted for external radiation, air and surface contamination (ICRP 75, 1997). The latter two are of interest when unsealed sources are being used. In the case of a Category IV gamma irradiator which makes use of a sealed source, monitoring of external radiation is of interest. It is carried out using dose rate meters which measure instant external exposure and dosimeters which give a measurement of cumulative external exposure (Workplace Monitoring for Radiation and Contamination, 2004).

Likewise, individual monitoring may be carried out for external and internal exposure, and for skin contamination.

2.5 PREVIOUS EVALUATION OF SHIELDING EFFICIENCY

Theoretical calculations based on the shielding and maze designs of the irradiation chamber at the facility when a cylindrical source was in use have been conducted by Emi-Reynolds & Akaho in 1994. The source, supplied by the Department of the Institute of Isotopes of the Hungarian Academy of Sciences, Budapest had an activity of 1.85PBq (50kCi) at the time. A 1-Dimension transport code, ANISNPC was used to give an indication of the strength of the water and concrete shielding. Results showed shielding was adequate in protecting radiation workers and equipment even if the activity of the source was increased by a factor of 2. Dose rates above the water pool was calculated as 0.02µSv/h, and the back of the irradiation chamber gave a recording of 2.77µSv/h and 4.31µSv/h at the top of the irradition chamber.

The integrity of the shielding of re-enforced high density concrete of the irradiation chamber containing a 1.85PBq (50kCi) cylindrical source was again investigated by Emi-Reynolds et al. (1997). A dose rate-survey meter model SG-102 was used to take measurements of dose rates. 6.273±0.745µSv/h and 0.755±0.944µSv/h were recorded within the personnel and goods entrances respectively. Outside the lead doors, 0.392µSv/h and 0.388µSv/h were the dose rates recorded. Similar recordings were made in the control room; 0.011µSv/h, electrical room- <0.010µSv/h, deionizer room; <0.003µSv/h and concrete rooftop; 0.303µSv/h. It was concluded that there was no radiation risk due to inadequate shielding at the facility.

In 2000, calculations were made prior to the installation of the rectangular plaque source by Fletcher et al. (2000). Calculations of dose rates 1m from the source using the line source approximation method gave a value of 649Sv/h whiles the elliptical method produced a value of 625Sv/h. This implied that a distance 5m from the source which might be occupied by staff would require a concrete shielding of 1.83m. The concrete thickness of the irradiation chamber wall is 1.9m, it was therefore concluded that the concrete shielding was adequate for a change to a rectangular source of 1.85PBq (50kCi).

In 2012, Thembinkosi (2012), took practical measurements by using TLD's, ECB and Fricke dosimetry to measure accumulative dose rates at controlled areas of the irradiation chamber housing the newly installed rectangular Co-60 source. The results showed that the workers at the facility were adequately protected by the shielding mechanisms in place. Recommendations made included the evaluation of shielding using mathematical methods and comparing simulated results with experimental findings. This is what this research seeks to carry out.

CHAPTER THREE

MATERIALS & METHODOLOGY

An outline of the equipment, materials, methods and calculations used in obtaining and processing data are explained in this chapter.

3.1 DESCRIPTION FOR EQUIPMENT USED

The equipment used in taking dose rate measurements at the facility was a Thermo Scientific Rad-Eye G-10 Gamma Survey Meter (Plate 3.1) with dimensions 3.1cm (H) ×6.1cm (B) ×9.6cm (L). It is a digital, highly sensitive, very portable and robust machine that produces quick and reliable dose rate measurements from 0.5μSv/h up to 100mSv/h. It contains a Geiger Muller Tube (GMT) which detects gamma and x- rays in the energy range of 50keV to 3MeV. It has vibrating, visual and audible alarms and is powered using 2 batteries of size AAA.

Plate 3.1 Thermo Scientific Rad Eye G-10 Gamma Survey Meter

3.2 MATERIALS, METHODS & CALCULATIONS

3.2.1 SOURCE PENCIL DESCRIPTIONS

The stainless steel source rack has 4 modules or compartments. Each module has 17 active source pencils of varying activities arranged on it amounting to 68 source pencils with a total manufacturing activity of 1.88PBq (50.89kCi), an installation activity as at August 2010 of 1.84PBq (49.72kCi) and a current activity as at March 2015 of 1.01PBq (27.40kCi).

The individual activities of the source pencils, 31 of which were manufactured in May 2010 and the other 37 in July 2010 as indicated in Appendix A, were used to calculate the activity during installation and current activity using equation 3.1

$$A = A_0 e^{-\gamma t} \tag{3.1}$$

Where A is the activity of the source after time t, A_0 is the initial activity and γ is the decay constant given by

$$\gamma = ln2/t_{1/2} \tag{3.2}$$

Time t, was calculated as 0.250yrs (at time of installation) and 4.83yrs (as at March 2015) for the sources manufactured in May, and was used to calculate the activity at the time of installation, August 2010 as 1.06PBq (28.72kCi) and the current activity as at March 2015 as 0.59PBq (15.83kCi).

Likewise, time t, was calculated as 0.080yrs (at time of installation) and 4.667yrs (as at March 2015) for the sources manufactured in July, and was used to calculate the activity at the time of installation, August 2010 as 0.78PBq (20.10kCi). The current activity as at March 2015 is 0.43PBq (11.57kCi).

The individual source pencils with their serial numbers and corresponding activities are also listed in Appendix A.

3.2.2 ENTRANCE MEASUREMENTS

Measurements were taken at the two entrances of the Irradiation chamber, the personnel door and the goods entrance door.

The personnel door was divided into columns labelled A to E and rows numbered 1 to 7 as in Figure 3.1. Measurements at each point (eg; A1, B1, C1, D1, E1, A2) were taken to record the levels of dose rates. Each measurement was repeated 5 times and the averaged value was recorded in µSv/h as the dose rate reading at that point.

Similarly, the goods entrance door was divided into columns labelled A to C and rows numbered 1 to 5 as in Fig 3.2. Measurements at each point (eg; A1, B1, C1, B2) were taken to record the levels of dose rates at that entrance to personnel at the centre during irradiation. Each measurement was made 5 times and the averaged value was recorded in µSv/h as the dose rate reading at that point. Values of dose rates measured at each position are listed in Appendix B.

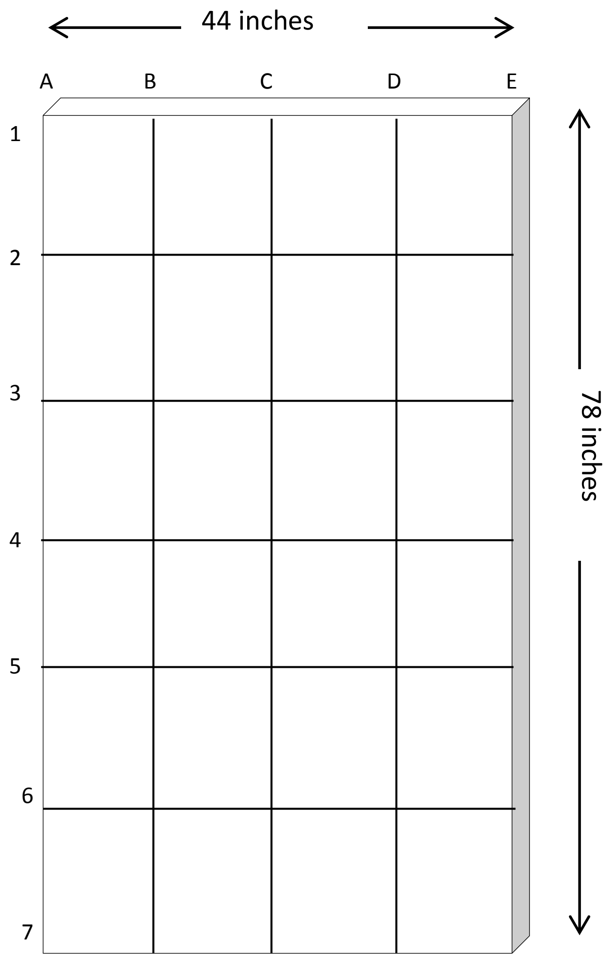


Fig 3.1 Lead Personnel Door
(Thickness = 9mm)

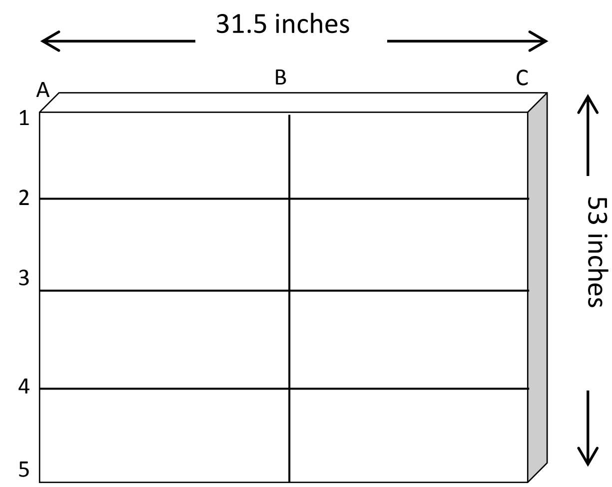


Fig. 3.2 Lead Goods Door
(Thickness = 9mm)

To predict dose rate just behind the personnel door and the goods door, equation 3.4 was used.

$$D = D_0 B e^{-\mu x} \tag{3.3}$$

Therefore

$$D_0 = D / B e^{-\mu x} \tag{3.4}$$

Where D is the dose rate just behind the door (inside the irradiation chamber), D_0 is dose rate measured outside the door, B is the buildup factor, x = 0.9cm is the thickness of the door, and μ is the linear attenuation coefficient (of lead in this case).

The linear attenuation coefficient, μ was calculated from equation 3.5.

$$\mu = ln2/HVL \tag{3.5}$$

Where HVL is the Half Value Layer of a shielding material - For lead, HVL= 1.2 for Co-60 photons with energies of 1.17MeV and 1.33MeV and average energy of 1.25MeV. Therefore $\mu = ln2/1.2 = 0.57762265$ cm^{-1}

The value for the buildup factor, B was obtained from the standard chart of buildup factors for lead corresponding to its value for relaxation length, μx.

$$\mu x = 0.5776 \times 0.9 \ = 0.51986$$

This corresponded to a buildup factor, B of 1.44

3.2.3 ROOF PLUG MEASUREMENTS

Measurements were taken at the five roof plugs of the irradiation chamber. They are slots that hold in place the hoist mechanism of the source. They were labelled P_1 to P_5 as shown in Plate 3.2. Measurements at each plug were each taken five times and recorded in Table 6.4. They were averaged to obtain the dose rate at each plug.

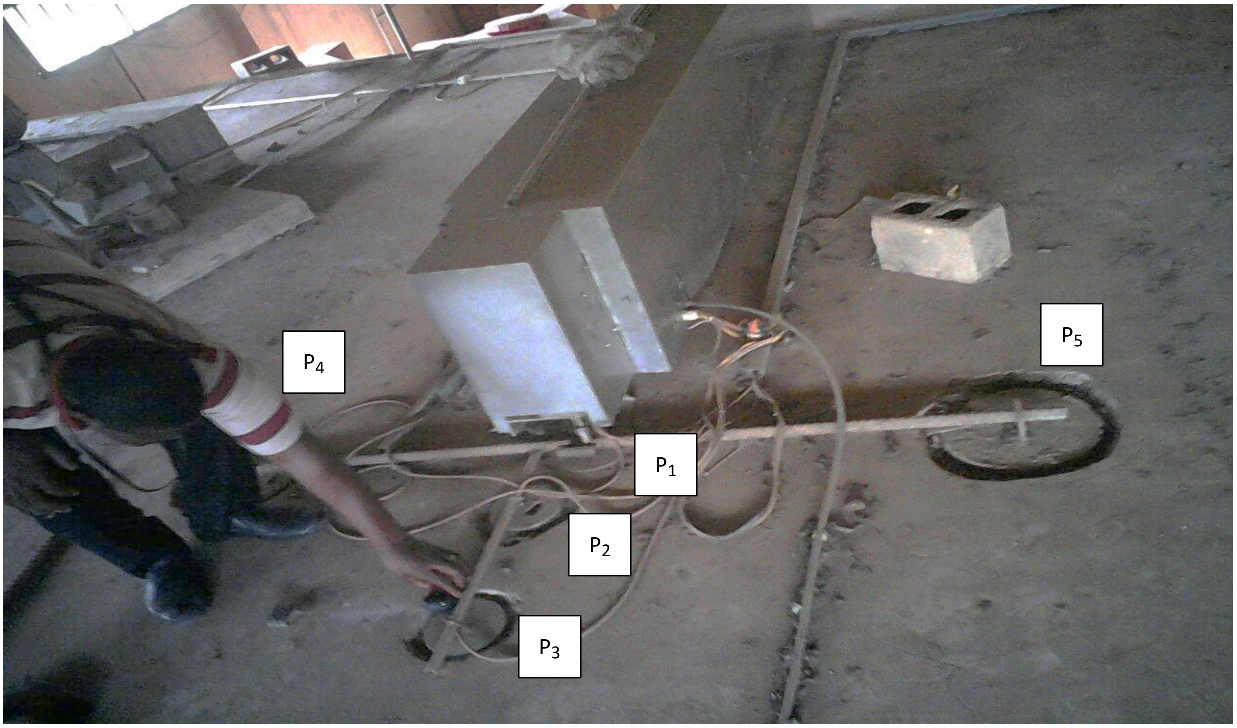


Plate 3.2 Roof top of irradiation chamber showing roof plugs

3.2.4 CONTROLLED AREA MEASUREMENTS

The control room, electrical room (Plate 3.3) and deionizer room (Plate 3.4) make up the controlled rooms and are the rooms closest to the irradiation chamber that are usually occupied or used by personnel. Measurements were taken likewise five times at strategic points in each room which share a wall with the irradiation chamber. Such points had openings to allow penetration of pipes and wiring and were prone to a relatively higher radiation dose rate measurement. Values recorded are listed in Appendix B.

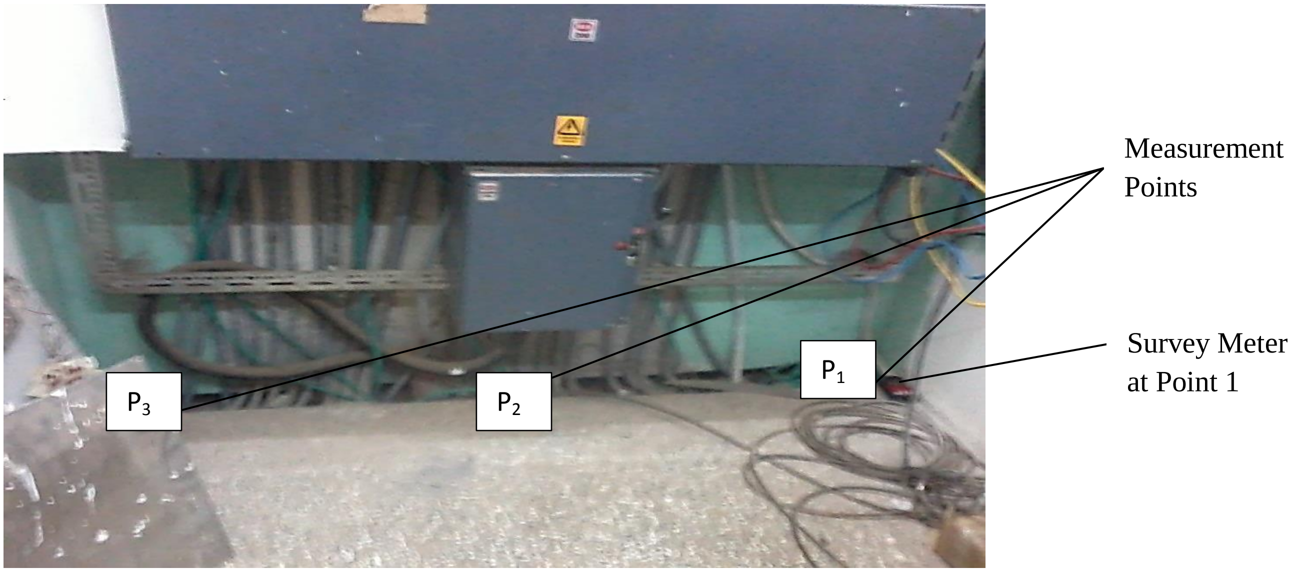


Plate 3.3 Electrical room showing measurement points

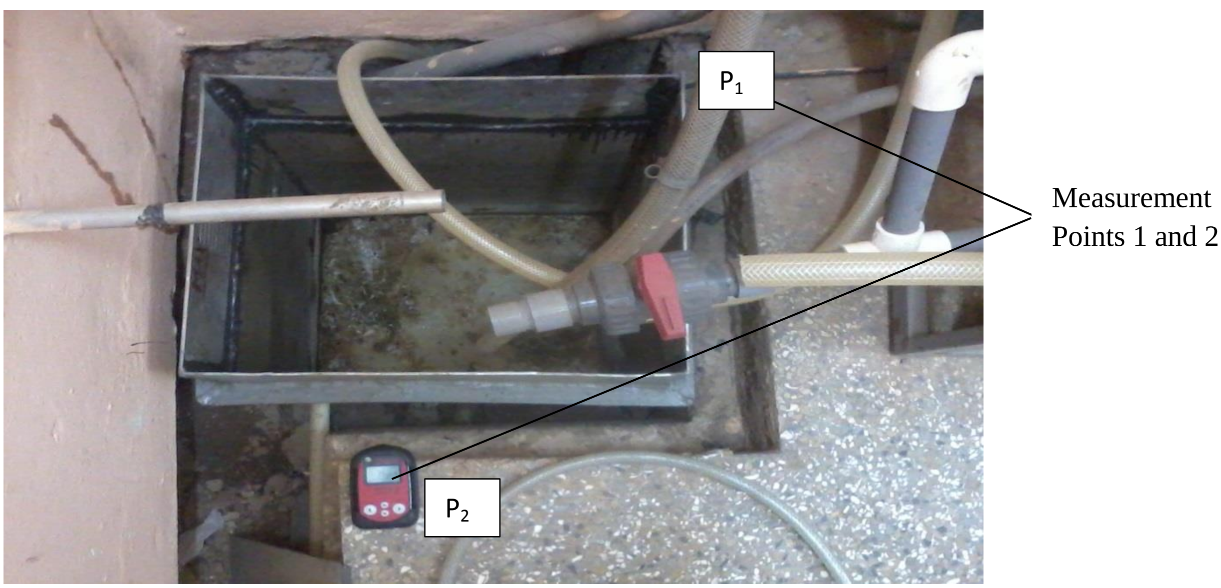


Plate 3.4 Deionizer room showing measurement points one and two

3.3 DOSE RATES & SHIELDING CALCULATIONS

The shielding of the irradiation chamber consists of water, which serves as a primary shield when the source is in the storage position, a concrete barrier of varying thickness then surrounds the irradiation chamber and two lead doors. Fig. 3.3 gives a picture of the dimensional plan of the Radiation Technology Centre. The thickness of the concrete of the irradiation chamber varies but the average thickness is 1.7m. The concrete wall between the irradiation chamber and the de-ionizer room is 1.9m thick. The maze walls are 1.2m apart, 1.5m and 2.0m high at the personnel maze and goods maze respectively.

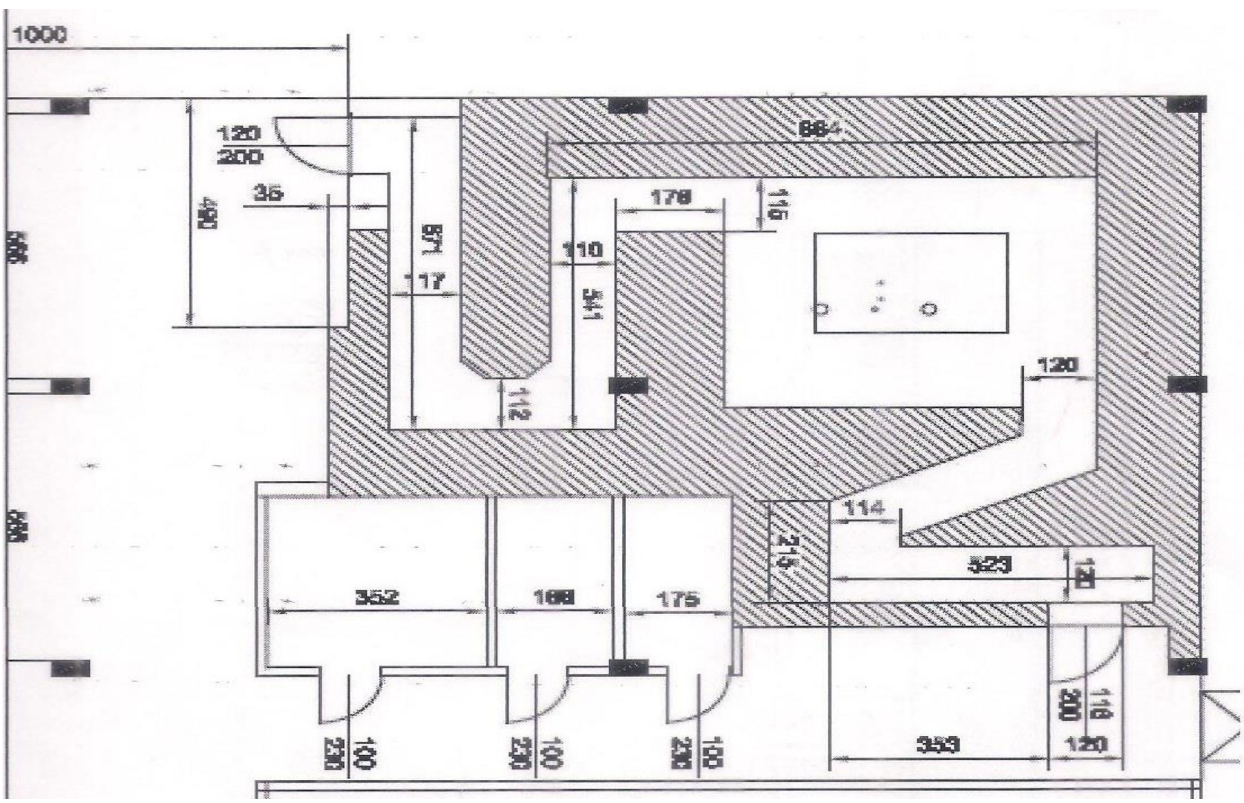


Fig. 3.3 Plan of Radiation Technology Centre (Safety Manual for Radiation Technology Centre, 2010)

Ideally, the dose rate, D_0 from a plane radiation source (Fig 3.4) may simply be calculated from equation 3.6

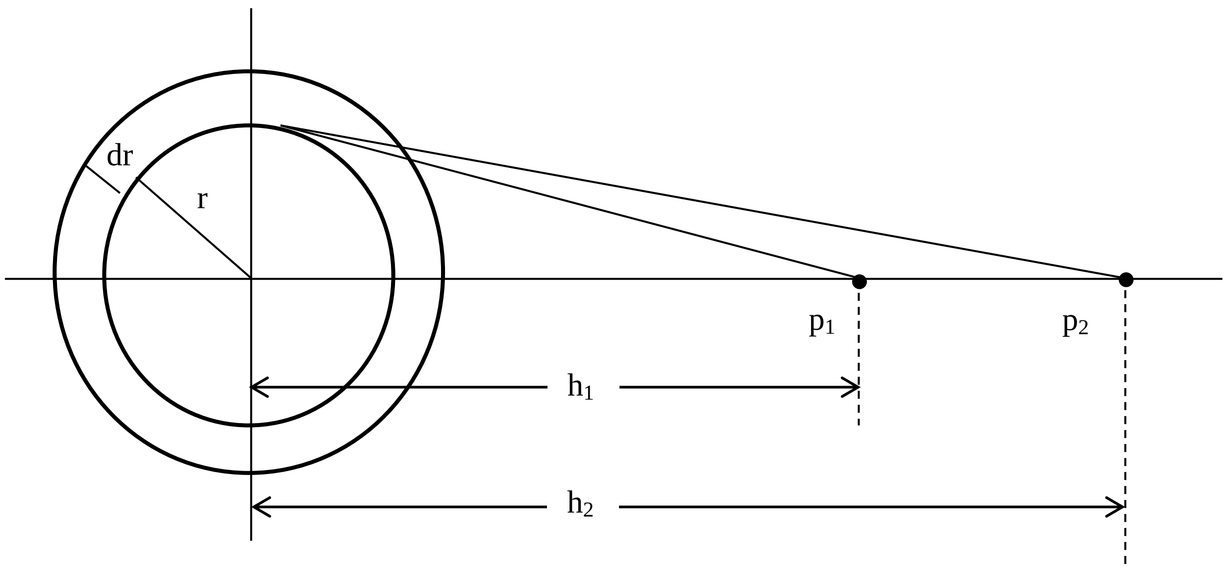


Fig. 3.4 Dose rates due to a plane source

$$D_0 = \int_0^r (\Gamma \times C_a \times 2\pi r dr)/(r^2 + h^2) \qquad (3.6)$$

Where Γ is the gamma constant, C_a is the surface activity, r is the radius of the plane source and h is the distance from the source. It can be simplified to equation 3.7

$$D_0 = \Gamma \times C_a \times \pi \times \ln[\frac{r^2+h^2}{h^2}] \qquad (3.7)$$

Different methods may be applied in calculating dose rate from a radioactive source. One method as applied by Fletcher et al, (2000) is the elliptical source method where an ellipse is superimposed on the plane of the source as in Figure 3.5, where b>>a.

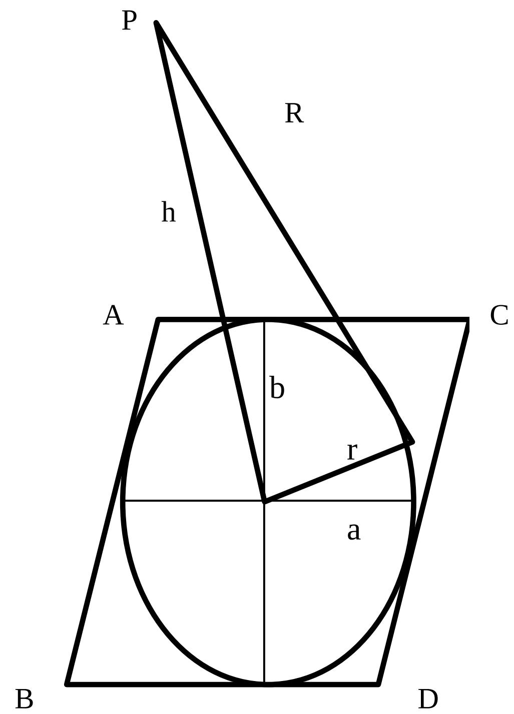


Fig. 3.5 Dose rates due to an elliptical source

Dose rate D_o is calculated from the basic relation;

$$D_0 = \Gamma C_a dA / kR^2 \qquad (3.8)$$

Where Γ is the gamma constant, C_a is the Surface Concentration, dA infinitesimal area of elliptical source, R is the distance from the source to the point of measurement and'

k is a constant (=1).

The dose rate from a rectangular source may also be determined by converting the dose rate from a circular source by multiplying it by a factor of 0.785 obtained from the ratios of their areas.

Another method of calculating dose rates is the line source approximation method which was applied in this work.

3.3.1 F-LINE CODE CALCULATIONS

The F-line code originally developed by Naszodi's software team was re-written by Basar Basarir and Szantos Attila. It computes dose rate for point, linear, planar and cylindrical sources, treating the latter two as combined line sources. Its output reading in krad/h is a dose rate mapping in the irradiation chamber and it requires input parameters of;

1) Shape of source:
2) Number of source rods/pencils:
3) Length of the rods/pencils (in cm):
4) Distance between each rod (in cm):
5) Activity of the first source and proceeding sources if different from first (in Ci):
6) Location of the Middle point for X, Y and Z directions:
7) Fixed plane of Y:
8) Coordinates in X direction, Initial, Calculation Step and End:
9) Coordinates in Z direction, Initial, Calculation Step and End:
10) Factor considering absorption:

Fig 3.6 is a flow chart explaining the algorithm behind the running of the F-line code.

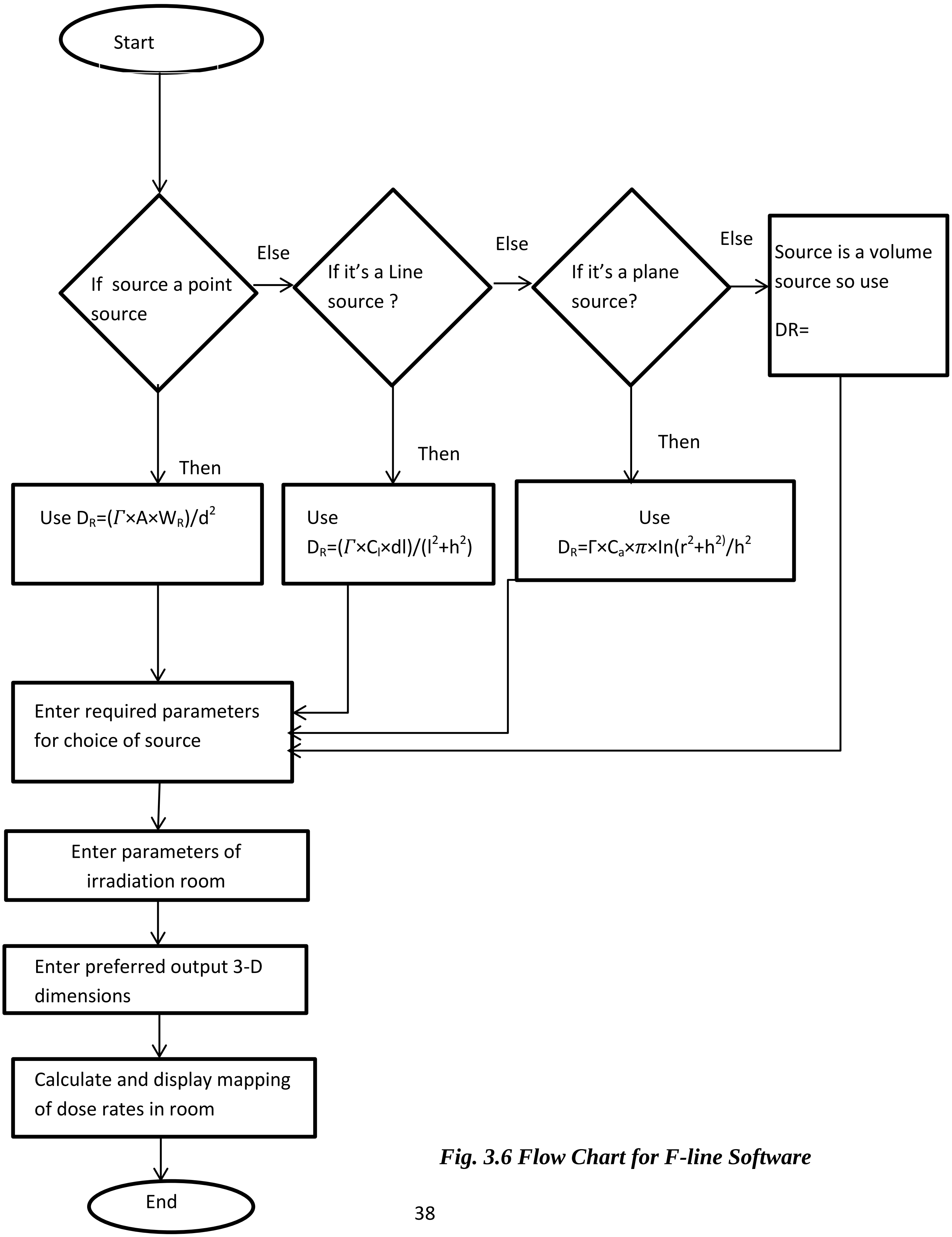


Fig. 3.6 Flow Chart for F-line Software

The F-line code used to generate virtual dose rates at various point in the irradiation chamber is based on the calculation of initial dose rate treating the source as a plane source. Equation 3.9 is the line-source approximation relation used to calculate dose rate 1m from a plane source.

$$D_0 = \Gamma C_i \theta / h \tag{3.9}$$

Where Do is the dose rate, Γ is the gamma constant, C_i is the surface activity and h is the distance from the line source to the point of interest.

The plaque source (68pencils) has a current total activity of 27396.0514Ci equivalent to 1.0137PBq. This implies an average source pencil activity of 402.88Ci equal to 1.4907TBq. Calculations concerning the source are based on the calculations for a plane radiation source. Therefore Activity per unit length of pencil = 3.3053TBq; Activity per unit volume of pencil = 0.1661GBq and $\Gamma_{Co\text{-}60}$ = 2.56678 ×10^{-18}A.kg at 1m per Bq. Each cylindrical pencil has inactive dimension of source 0.016m×0.451m and active dimension 0.011m×0.437m as shown in Figure 3.7.

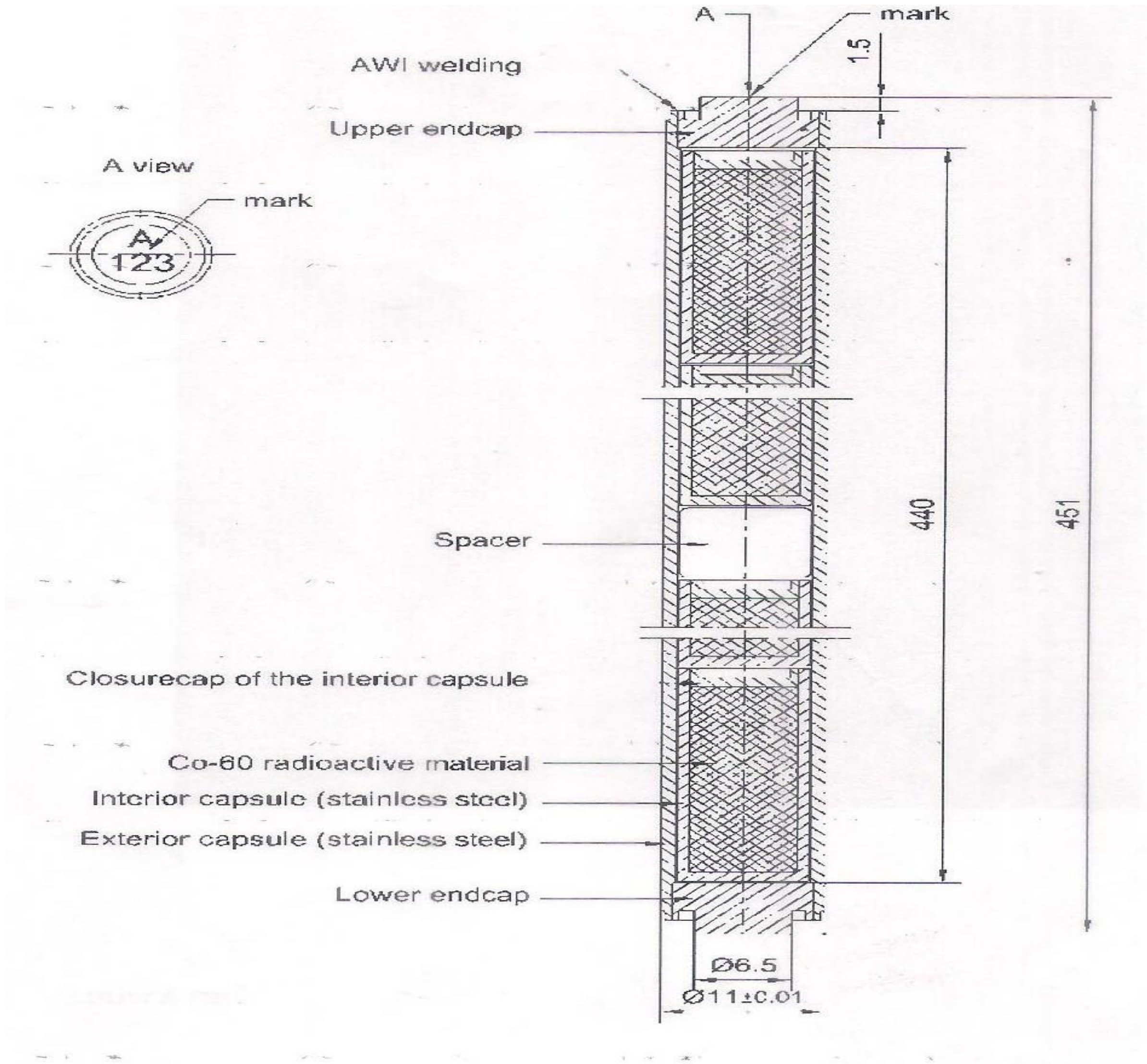


Fig. 3.7 RTC Co-60 Source Pencil (Safety Manual for Radiation Technology Centre, 2010)

3.3.2 EXTRAPOLATION FOR A 500kCi SOURCE

An extrapolation using the current activity of the source for a source of 500kCi (future source strength) required multiplying values measured and computed by a factor of 18.25 to cater for the increment as shown in Equation 3.10.

$$D_{500} = \left(\frac{D_{27.4}}{27.4}\right) \times 500 = 18.25 \times D_{27.4} \tag{3.10}$$

Where $D_{27.4}$ is the dose rate of the source with a current activity or strength as at March 2015 of 27.4kCi at a distance *h*. D_{500} is the calculated dose rate for a 500kCi source.

3.4 DOSE HISTORY (THERMOLUMINESCENT DOSIMETERS)

Accumulative doses of occupationally exposed workers at the facility were obtained from the Radiation Protection Officer, Mr. Peter Davor. Accumulative radiation doses over an average period of three months were recorded using Thermo-Luminescent dosimeter (TLD) badges containing Lithium-Fluoride chips worn by personnel of the facility. The dosimeters were read at the Radiation Protection Institute using the Harshaw 6600 TLD reader. Reports displayed values of personal dose equivalent (Strongly Penetrating Radiation – Hp (10) and Weakly Penetrating radiation – Hp (0.07) in mSv) tallied against the respective personnel.

Results were compiled for a period of 12months and the total accumulative personal dose equivalent was acquired to compute the collective dose.

CHAPTER FOUR

RESULTS & DISCUSSION

This chapter gives an account of the results obtained from both the practical measurements taken and the theoretical values obtained using the F-line code. Dose history results are also accounted for in this section. An analysis of the results, what it represents and the impact it may have on personnel is discussed. Values are also compared with previously conducted measurements and calculations.

4.1 RESULTS FROM PRACTICAL MEASUREMENTS

The Thermo Scientific Rad-Eye G-10 Gamma Survey meter as calibrated at the Secondary Standards Dosimetry Laboratory (SSDL) of RPI on 20th August 2014 had a calibration factor of 1.14. It has an uncertainty of 10% for each measurement taken. (Calibration certificate is in Appendix C)

4.1.1 PERSONNEL & GOODS DOORS

Background measurement taken at the entrance of the facility when the source was down and in the absence of any radioactive material was recorded as 0.080±0.010μSv/h (or 80±10nSv/h). Measurements made at the personnel door produced an average dose rate of 0.090μSv/h. A minimum value of 0.029μSv/h was obtained at position B_5 and a maximum value of 0.143μSv/h was measured at a position A_3. A_3 was adjacent to the Co-60 test source on the wall as shown in Plate 4.1. The relatively high value here may be attributed to radiation from the test source and not from the radiation source within the irradiation chamber. The average dose rate value recorded implies that for a worker operating 8hours a day 5 times a week for 50 weeks a year, a collective dose of 0.180mSv per year would be recorded. Recorded values were close to background readings and were very low, these results were very positive and certified adequate lead thickness of the door. Also, time spent by staff at the location of the personnel door would not be up to 8 hours a day, hence this value is an estimation on the high side.

Plate 4.1 Personnel Door showing position of Test source relative to door position A_3

The goods door had a higher average dose rate reading, of 0.109µSv/h. 0.055µSv/h at C_4 and a maximum of 0.152µSv/h at position A_3. Likewise, a collective dose of 0.218mSv would be received by a worker operating in that area which means that there is no cause for alarm. Likewise, staff do not spend 8 hours a day at this location, this value is therefore also a higher estimate. The higher average dose rate recorded at the goods door may be attributed to leakage radiation being emitted from the cavity below the door.

4.1.2 CONTROL, ELECTRICAL & DEIONIZER ROOMS

Practical measurements made demonstrated that in the controlled areas which included the control room, electrical room and deionizer room, the average dose rates obtained were 0.116µSv/h, 0.089µSv/h and 0.614µSv/h. These rooms directly share a wall with the irradiation chamber. 0.232mSv, 0.178mSv and 1.228mSv are the respective yearly dose exposures projected. The higher reading in the deionizer room may be attributed to the hollow pipes leading into the irradiation chamber which therefore transmit radiation directly into the room. It may also be linked to the covering of the pipe not being adequately sealed. Values show that personnel who may be occupying and working in these rooms remain safe and are not over exposed but movement should be restricted in the deionizer room.

4.1.3 ROOF PLUGS

Roof plugs 1 to 5 produced an average readings 0.135µSv/h equivalent to yearly dose of 0.270mSv. A particular area shown in Plate 4.2 however had a high recording of

8.151µSv/h. This value exceeds the 7.5µSv/h limit as stated by the Nuclear Regulatory Authority (NRA). This may be due to leakage in the concrete shielding of the roof at a point. The 8.151µSv/h recording will give a yearly dose of 16.302mSv which is still below the dose limit of 50mSv per year set by the International Commission for Radiological Protection (ICRP) but can be reduced further. An upgrade of the source to an activity of 18.5PBq (500kCi) which has decayed to 10.14PBq (274KCi) after a period of approximately 4.57yrs will however imply a theoretically computed annual dose of 163mSv which may be a major cause of concern. It does not only exceed the regulatory approved limits but also increases the probability of staff experiencing stochastic effects. Although staff rarely visit the roof top hence spend very little time there except to clean the vicinity or carry out maintenance work, dose rate measurements when the facility is scaled up to 500kCi must be carried out. This position was marked as the point of concern after the discovery of this high value.

Plate 4.2 Gamma Survey Meter recording value at "high point" of roof plug3

4.2 THEORETICAL RESULTS

Values obtained using the F-line software were processed to give theoretical values of scattered and transmitted radiation at various points in the facility. A sum of these gave total dose rates at specified positions.

4.2.1 PERSONNEL & GOODS DOORS

Values obtained in Table 4.1 indicate the total dose rates at the two maze entrances of the Irradiation chamber just before the lead shielding doors. The values were extremely low and considering the source activity, may not actually be the exact dose rates at the locations in a real life situation when the source is up and leakage is being considered.

Table 4.2 give the total dose rate behind the lead doors which indicates the dose rates to personnel at those specific locations. The total dose rate is a sum of the scattered and transmitted radiation at these points in µSv/h. These values indicate that there is sufficient shielding of concrete and lead to attenuate the radiation to locations at the entrances. Values were below the dose rate acceptable limit of 7.5µSv/h for controlled areas.

Table 4.1 Dose Rates due to Scattered and Transmitted Radiations at maze entrances before lead doors

Entrance	Scattered Dose Rate (mSv/h)	Transmitted Dose Rate (mSv/h)	Total Dose Rate (mSv/h)
Personnel Door	0.073	0.0029	0.075
Goods Door	0.067	0.0028	0.069

Table 4.2 Dose Rates due to Scattered and Transmitted Radiation behind lead doors

Entrance	Scattered Dose Rate (μSv/h)	Transmitted Dose Rate (μSv/h)	Total Dose Rate (μSv/h)
Personnel Door	0.078	0.003	0.081
Goods Door	0.072	0.003	0.075

Theoretically, values at the Personnel Door were higher than the goods door. This may be due to the longer maze leading to the Goods Door (9.1m) than that leading to the Personnel Door (8.1m). This is not in agreement with experimental values where the Personnel Door recorded a lower value rather than the Goods Door.

4.2.2 ROOF TOP (COINCIDING WITH SOURCE POSITION IN CHAMBER)

Calculations based on the 1.6m thickness of concrete of the Irradiation chambers' roof produced a dose rate value of 0.08μSv/h. This may be the dose rate in the same area of the roof which coincides with the position of the source in the chamber and is likely an indication of the peak dose rate on the roof. Appropriately increasing this by a factor of 18.25 to cater for a 500kCi source would remain below the prescribed limits.

4.2.3 DEIONIZER ROOM

A dose rate of 0.19μSv/h was obtained as the final dose rate in the deionizer room which directly shares a wall with the main chamber hosting the source. The thickness of the wall between the two rooms is 1.9m and values proved that this is sufficient to attenuate the gamma rays from the chamber when in the operation mode.

4.2.4 OUTSIDE CHAMBER

Outside the irradiation chamber to the left of the facility where the wall thickness is 1.7m thick, a dose rate value of 0.07µSv/h was obtained as the computed value. This value will stand as exposure to the public since it is what is transmitted through the wall. It is lower than background readings in the facility and would not exceed the public exposure limit of 2mSv per year.

4.2.5 EXTRAPOLATION FOR 500kCi

Dose rates predicted for a 500kCi source in specified areas of the facility based on both practical and theoretical results are presented in Table 4.3.

Practically, results show the deionizer room maybe potentially unsafe if an upgraded 500kCi source is installed without increasing the shield thickness. The location of high dose rate corresponds in actual fact to the area around the pipe interconnection between the deionizer room and the adjourning pool in tank in the irradiation chamber. A practical way of reducing this dose rate is to increase the scattering by introduction of lead shots. Concrete or sand can also be used to fill cavities around the tank to reduce this value.

Theoretically, extrapolated results showed that the deionizer room would permit such an upgrade with the current shielding mechanisms in place.

This contradicts practical results because the limit of 7.5µSv/h was exceeded in this vicinity alone except that the occupancy of this location may never go beyond 1/4. Shielding for a source with higher activity would require further calculations.

Table 4.3 Extrapolated Dose Rates for a 500kCi source

Measurement point	Extrapolated Practical Values (µSv/h)	Extrapolated Theoretical Values (µSv/h)
Personnel Door	1.642	1.497
Goods Door	1.989	1.387
Roof	2.464	1.460
Deionizer Room	11.206	3.522

4.3 INTER-COMPARISON OF RESULTS

4.3.1 COMPARISON OF PRACTICAL & THEORETICAL RESULTS

Practical values for all the points of interest were higher than the theoretical values generated as outlined in table 4.4. It is possible that the F-line code produces values of an ideal situation or some relevant factors are ignored while practically, leakage radiation and scatter radiation come into play. Hence the F-line code may not be reliable enough to project exact dose rates based on the specifications of a source. This may explain why the theoretical values were close but not equal to the practically measured values. The deionizer room in particular had a significant difference between measured values and theoretical predictions. When point 2 which is closest to the pipe opening and which had the highest practical readings is taken out, an average of 0.189μSv/h is obtained and this is closer to its theoretical value. The practical and theoretical values at the personnel door also had a difference of 8%.

Table 4.4 Comparison of Practical and Theoretical dose rates using the F-line code

Position	Practical Measurement (μSv/h)	Theoretical Value (μSv/h)
Personnel Door	0.090±0.020	0.082
Goods Door	0.109±0.027	0.076
Roof Plugs	0.135±0.030	0.080
Deionizer Room	0.614±0.550	0.193

4.3.2 COMPARISON OF THEORETICAL RESULTS WITH PREVIOUS COMPUTATIONS

Theoretical calculations carried out by Emi-Reynolds & Akaho in 1994 using the ANISN PC code gave final values of dose rates at the personnel and goods doors for a source strength of 0.5MCi. These were used to predict the dose rates for a rectangular source. Comparison of these results scaled down to the current activity of the source, with theoretical values for a rectangular source obtained in this work is given in table 4.5.

Table 4.5 Comparison of theoretical dose rates with previous computations

	Dose Rates (μSv/h)		
Position	Emi-Reynolds & Akaho (1994)	Source activity scaled down to 27.4kCi	Present Study (Theoretical)
Personnel Door	48.8	2.67	0.082
Goods Door	0.26	0.01	0.076

4.3.3 COMPARISON OF PRACTICAL RESULTS WITH PREVIOUS MONITORING VALUES

Results obtained in 1997 from practical monitoring of the facility by Emi-Reynolds et al (1997) are compared to practical values in this work in table 4.6. Their monitoring took place when the cylindrical source was in use hence were multiplied by a factor of 0.7 to cater for the difference in shape of source.

Table 4.6 Comparison of practical dose rates with those measured by Emi-Reynolds et al (1997)

	Dose Rates (μSv/h)		
Position	Emi-Reynolds et al. for 50kCi (1997)	Scaled down to 27.4kCi × factor	Present Study (Practical)
Personnel Door	0.392	0.150	0.090±0.022
Goods Door	0.388	0.149	0.109±0.027
Control Room	0.011	0.004	0.116±0.020
Electrical Room	<0.01	<0.004	0.089±0.020
Deionizer Room	<0.003	<0.001	0.614±0.550
Roof	0.300	0.115	0.135±0.030

Values of dose rates for the doors were all higher during the use of the cylindrical source than that of the current rectangular source. The control room and roof top however had higher values in this present study. This may be attributed to the fact that the current dimensions of the source alters dose rates to the specified locations.

Comparisons between dose rates measured in this work and by Thembinkosi (2012) using the Rados Dose Rate Meter. Her results were scaled down to the present activity of the source and the calibration factor of the Rados Dose Rate Meter (0.9+0.16) was accounted for as shown in Table 4.7. Thembinkosis' value at the goods door and control room were both identical, practical measurements in this work also show identical measurements in both positions. A high value was also recorded in the deionizer room using the Rad-Eye Gamma Survey Meter and was in agreement (though lower) with the high value measured using the Rados Dose Rate Meter, the other values in the present study were all lower. Both demonstrate a need for further measures to be taken to reduce values at the location.

Table 4.7 Comparison of practical dose rates with those measured by Thembinkosi (2012)

	Dose Rates(μSv/h)		
Position	Rados (Thembinkosi, 2012 without calibration)	Scaled down to 27.4kCi with calibration	Rad-Eye G-10 (Present study)
Personnel Door	0.080	0.049	0.090±0.022
Goods Door	0.050	0.030	0.109±0.027
Control Room	0.050	0.030	0.116±0.020
Electrical Room	0.070	0.043	0.089±0.020
Deionizer Room	4.220	2.559	0.614±0.550
Roof	0.070	0.043	0.135±0.030

4.4 DOSE HISTORY RESULTS

Personal dose equivalent (PDE) values for Strongly penetrating Hp (10) and Weakly penetrating Hp (0.07) radiation are recorded in Appendix D. Hp (10) results were similar to that of the Hp (0.07).

For Hp (10), the minimum and maximum values obtained were 2.11mSv/yr and 3.28mSv/yr respectively. The average collective dose was 2.22mSv. For Hp (0.07), maximum and minimum recordings were 3.78mSv/yr and 2.43mSv/yr respectively with an average value of 2.29mSv. These collective doses of personnel are well below 50mSv per year. This is a further indication that personnel have been operating according to laid down procedures.

CHAPTER FIVE

CONCLUSION & RECOMMENDATIONS

A conclusion of the research and appropriate recommendations to various stakeholders are made in this chapter.

5.1 CONCLUSION

Dose rate monitoring in a Semi-Commercial Gamma Irradiation Facility in Ghana with a rectangular plaque source configuration has been carried out.

In conclusion, both theoretical computations and practical results indicate that total radiation dose rate at the entrances of the mazes may be due mainly to scattered radiation. Theoretical values obtained were all below practical measurements taken. The shielding by lead doors (0.009m) and concrete walls (1.6m-1.9m) of the installed 1.85PBq (50kCi) rectangular plaque Co-60 source irradiation chamber are sufficient to ensure that with the current source, workers are not over exposed and do not exceed the internationally and regulatory authority limit of 7.5µSv/h (i.e.50mSv per year) when the source is out of its storage position and in operation. The introduction of a maze of 8.1m from the personnel entrance to the irradiation room (and 9.1m to the goods entrance) is also adequate and contributes to the low value due to multiple scatter of doses emanating from the irradiation room.

Collective dose of personnel demonstrates that accumulated doses are well below limits and personnel would continue to be safe if the current procedures and safety protocols are strictly followed. The values demonstrate that staff follow best practices. The visual and audio warning mechanisms, interlock systems in place and operational standards of the facility are satisfactory and adequate in preventing over-exposure.

There may however be concerns with an upgrade to 500kCi and further measurements and calculations of shielding (in particular for the deionizer room) would need to be carried out before such an upgrade.

Results also show that the feet (extremities) may be exposed to higher values of radiation in the deionizer and electrical room than the torso assuming there is uniformity of the shielding thickness.

5.2 RECOMMENDATIONS

It is recommended that regular dose measurements should be made at the roof plugs of the irradiation chamber by the authorities of the facility and the regulatory authority to verify the relatively high dose rate readings. Remedial action may also be taken to reduce the levels of dose rates. This may be in the form of packing lead sheets into crevices around the roof plugs to increase shielding and multiple scatter of radiation. Another possible solution may be to place lead shots into cavities around the roof plugs. Also, as an alternative, high density concrete may be poured around the roof plugs to reduce the size of the crevices around them, hence reduce dose rates. The roof top may also be enclosed with glass frames and doors to prevent dust from entering the hoist mechanism especially during Harmattan season since this would call for frequent maintenance.

Access to the roof plugs must also be restricted by placing a latch on the ladder leading to the vicinity.

Though measurements were very low, further precautions can be taken by clogging entrances of openings in the electrical and deionizer rooms through which the wiring and pipes are passed with lead shots. This would further scatter radiation from the cavities and reduce what is being transmitted into those rooms. A maze system may also be applied for the pipes leading to the deionizer room to increase scatter and reduce directly transmitted radiation.

Up to date records of radiation dose assessments in the form of TLD badge reports should be made available at the facility to ease computations. This would help immediately notice any abnormally high recording for appropriate measures to be taken early enough.

Further theoretical studies may be carried out also using different mathematical methods and results compared with practical measurements to ascertain its accuracy, this would help in projection of dose rates for a higher source activity.

Considering the category of the irradiation facility (category IV), the ISO standard 21482 (ionizing radiation supplementary symbol) should be placed on the source rack in addition to the Trefoil at the entrance of the irradiation chamber.

It is also recommended that for further studies, a 3-D mapping of accumulative dose rates in the control, electrical and deionizer room be conducted to obtain contours of dose rates at all positions in the rooms. This would reveal any cracks in the concrete shielding.

REFERENCES

Adu-Gyamfi, A. (2011). *Safety Analysis Report.* Radiation Technology Centre Safety Report.

Agarwal, S. (1998). Radiation Protection in Panoramic Gamma Irradiators. *IAEA Regional Basic Professional Trainig on Radiation Protection*, (553-575)

Al-Ani M.Y and Al-Khalidy F.R. (2006). Use of Ionizing Radiation Technology for Treating Municipal Wastewater. *International Journal of Environmental Research and Public Health, 3(4)*, 360-368.

ANSTO. (n.d.). *Food Agriculture*. Retrieved March 16, 2015, from Australian Nuclear Science and Technology Organisation: http://www.ansto.gov.au/Nuclear Facts/BenefitsofNuclearScience/FoodandAgriculture

ASTM ISO. (n.d.). *Standard Practice for Use of the Ethanol-Chlorobenzene Dosimetry System*, ASTIM51538-09. Retrieved from www.astm.org.

Capital Regional District. (n.d.). Retrieved February 2, 2015, from Capital Regional District: http://www.crd.bc.ca/project/other-projects-initiatives/wastewater-planning/what-is-wastewater

Cember, H., & Johnson, T. (2009). *Introduction to Health Physics, 4th Edition.* The McGraw Hill Co.

Dahlan, K. Z. (n.d.). Physical Methods, Instruments and Measurements. *Radiation Processing, IV*, p. 3.

Emi-Reynolds, G., & Akaho, E. (1994). A Theoretical Evaluation of The Effectiveness of The Biologiccal Shields for The 0.5MCi Irradiation Facility. *Journal of The University of Science and Technology*, 79-89.

Emi-Reynolds, G., Banini, G., & Akaho, E. (1997). Practical Evaluation of the Biological Shielding Effectiveness of the Gamma Irradiation Facility at Kwabenya, Ghana. 19.

Fernandez, A. F., B. Brichard, H. O., Nieuwenhove, R. V., & Berghmans, F. (2015). Gamma dosimetry using Red 4034 Harwell Dosimeters in Mixed Fission Neutrons and Gamma Environments.

Fletcher, J., Emi-Reynolds, G., & Glover, E. (2000). Shielding Calculations for Changing From Circular To a Rectangular Source Configuration for a Pilot Scale Irradiator. *Journal of Applied Science and Technology*, 39-44.

ICRP 75 (1997). *General Principles for The Radiation Protection of Workers*, pp. 37-38.

IAEA-1454 (2010). *IAEA Safety Standard Series SSG No- 8.* Vienna: IAEA.

IAEA SSG-8 (2010). *Radiation Safety of Gamma, Electron and X-Ray Irradiation Facilitie*, pp. 40-41.

International Atomic Energy Agency. (1999). *The Annual Report for 1999.* Vienna.

ISO/ASTM-51538 (2009). Practice for use of the ethanol-chlorobenzene dosimetry system.

Izewska, J., & Rajan, G. (n.d.). Radiation Dosimeters. In *Review of Radiation Oncology Physics: A Handbook for Teachers and Students* (pp. 71-99).

Izewska, J., & Rajan, G. (n.d.). Radiation Monitoring Instruments. In *Review of Radiation Oncology Physics: A Handbook for Teachers and Students* (pp. 85-102).

Joint FAO/IAEA Programme. (n.d.). Retrieved July 2, 2015, from Nuclear Techniques in Food and Agriculture-Sterile Insect Technique: http://www-naweb.iaea.org/nafa/ipc/sterile-insect-technique.html

Krizalkovic, O. (2001). *Chemical Dosimetry in Ionising Radiation.* Ondrej ., Bratislava.

Materials Used in Radiation Shielding. (2015). Retrieved March 11, 2015, from Thomas Net website: http://www.thomasnet.com/articles/custom-manufacturing-fabricating/radiation-shielding-materials

Medical Uses (2007). Retrieved 3 2015, 6, from Radiation Answers Organization: www.radiationanswers.org/radiation-sources-uses/medical-uses.html

McAlister, D. R. (2015) Gamma Ray Attenuation Properties of Common Shielding Materials. Listle, USA.

NRC (2015). *Dose Rate*. Retrieved March 16, 2015, from United States Nuclear Regulatory Commission: http://www.nrc.gov/reading-rm/basic-ref/glossary/dose-rate.html

Nuclear Safety Technical Assessment Guide (2013). *NS-TAST-GD-002 Revision 4*, p. 5.

Optek-Danulat Incorported. (2014). Retrieved 2014, from Optek-Danulat Incorporated website: http://www.optek.com/Turbidity.asp

Pelman, H. (2014). *United States Geological Survey*. Retrieved December 13, 2014, from A United States Geological Survey webpage: http://water.usgs.gov/edu/turbidity.html

Poskus, A. (2012). Attenuation of Gamma Rays.

Radiation Safety Manual. (2015). Stanford University: Stanford University.

Radioactive Source Reduction and Management. (2014). Retrieved March 2015, 6, from US Environmental Protection Agency: http:/www.epa.gov/radiation/source-reduction-management/applications.html

Retrieved March 6, 2014, from Hong Kong Observatory: (2003) www.hko.gov.hk/education/dbcp/radiation/eng/r15.html

Rogozkin, V. (n.d.). *The Free Dictionary*. Retrieved 02 28, 2015, from The Free Dictionary: encyclopedia2.thefreedictionary.com/radiation+shielding

Safety Manual for Radiation Technology Centre (2010). Ghana Atomic Energy Commission: Radiation Technology Centre.

Spector, D. (2014). *About Us: Business Insider*. Retrieved September 20, 2014, from A Business Insider Website: www.businessinsider.com

Terr, D. (2015). *The electromagnetic spectrum*. Retrieved March 2015, http://www.davidterr.com/science-articles/electromagnetic_spectrum.html

Thembinkosi E. (2013). *Experimental Evaluation of the Biological Shielding Effectiveness of the Upgraded Gamma Irradiaiton Facility (GIF).* Mphill Thesis, University of Ghana. Legon-Accra (Unpublished).

Thermo Scientific RadEye G/G-10G Personal Dose Rate Meters. (n.d.). Retrieved January 2015, 30, from Fisher Scientific: http://www.fishersci.com/ecomm/servlet /fsproductdetail_10652_4996961__-1_0

US EPA Radiation Protection. (2013). Retrieved March 2015, from US Environmental Protection Agency: http://www.epa.gov/radiation/understand/gamma .html#emission

US Geological Survey. (2014). Retrieved September 20, 2014, from US Geological Survey: http://water.usgs.gov/edu/earthhowmuch.html.

Workplace Monitoring for Radiation and Contamination. (2004). *Practical Radiation Practical Manual*.

World Nuclear Association. (2015). Retrieved March 16, 2015, from: (WNA) http://www.world-nuclear.org/info/Country-Profiles/Countries-T-Z/USA--Nuclear-Power/

APPENDIX A: SOURCE PENCIL CHARACTERISTICS

Table A.1 Arrangement of source pencils on Steel rack

C155	C158	C215	C181
		C167	C172
C192	C216	C191	C173
C193	C218		
C194	C197	C207	C225
C184	C174		
		C163	C166
C195	C217		
		C208	C226
		C157	C161
C196	C219		
C188	C178	C209	C227
C202	C223		
		C210	C228
C186	C176	C211	C229
		C187	C177
C154	C159		
C204	C220		
		C189	C179
C162	C165		
		C185	C175
C203	C221	C212	C230
C190	C180	C214	C170
C182	C164		
C206	C222	C213	C171
		C156	C160

Table A.2 Table of activities of source pencils manufactured in May 2010

	Initial Activity		Installation Activity (August 2010)		Current Activity (March 2015)	
Serial No	**TBq**	**Ci**	**TBq**	**Ci**	**TBq**	**Ci**
C154	113.65	3071.62	110.02	2973.40	60.63	1638.72
C155	120.96	3269.19	117.09	3164.65	64.53	1744.12
C157	87.07	2353.24	84.29	2277.99	46.45	1255.46
C158	120.30	3251.35	116.45	3147.38	64.18	1734.61
C159	115.31	3116.49	111.62	3016.83	61.52	1662.66
C161	87.90	2375.68	85.09	2299.71	46.89	1267.43
C162	14.95	404.05	14.47	391.13	7.98	215.56
C163	20.60	556.76	19.94	538.95	10.99	297.03
C164	19.61	530.00	18.98	513.05	10.46	282.76
C165	15.62	422.16	15.12	408.66	8.33	225.22
C166	17.28	467.03	16.73	452.09	9.22	249.16
C167	20.60	556.76	19.94	538.95	10.99	297.03
C172	18.93	511.62	18.32	495.26	10.10	272.95
C173	18.43	498.11	17.84	482.18	9.83	265.74
C174	15.72	424.86	15.22	411.28	8.39	226.67
C175	18.26	493.51	17.68	477.73	9.74	263.29
C176	19.27	520.81	18.65	504.16	10.28	277.85
C177	19.75	533.78	19.12	516.71	10.54	284.78
C178	18.68	504.86	18.08	488.72	9.97	269.35
C179	19.61	530.00	18.98	513.05	10.46	282.76
C180	18.00	486.49	17.42	470.93	9.60	259.54
C181	18.09	488.92	17.51	473.28	9.65	260.84
C182	16.57	447.84	16.04	433.52	8.84	238.92
C184	18.09	488.92	17.51	473.28	9.65	260.84
C185	17.75	479.73	17.18	464.39	9.47	255.94
C186	18.09	488.92	17.51	473.28	9.65	260.84
C187	17.92	484.32	17.35	468.84	9.56	258.39
C188	17.87	482.97	17.30	467.53	9.53	257.67
C189	17.03	460.27	16.49	445.55	9.09	245.56
C190	17.53	473.78	16.97	458.63	9.35	252.77
C191	18.38	496.76	17.79	480.87	9.81	265.02

Table A.3 Table of activities of source pencils manufactured in July 2010

	Initial Activity		Installation Activity (August 2010)		Current Activity (March 2015)	
Serial No	**TBq**	**Ci**	**TBq**	**Ci**	**TBq**	**Ci**
C156	108.03	2919.73	106.91	2889.52	58.89	1591.67
C160	120.98	3269.73	119.73	3235.90	65.95	1782.47
C192	15.76	425.95	15.60	421.54	8.59	232.20
C193	15.59	421.35	15.43	416.99	8.50	229.70
C194	17.13	462.97	16.95	458.18	9.34	252.39
C195	16.27	439.73	16.10	435.18	8.87	239.71
C196	16.79	453.78	16.62	449.09	9.15	247.38
C197	15.25	412.16	15.09	407.90	8.31	224.69
C202	17.30	467.57	17.12	462.73	9.43	254.89
C203	15.25	412.16	15.09	407.90	8.31	224.69
C204	15.59	421.35	15.43	416.99	8.50	229.70
C206	15.25	412.16	15.09	407.90	8.31	224.69
C207	17.13	462.97	16.95	458.18	9.34	252.39
C208	16.27	439.73	16.10	435.18	8.87	239.71
C209	16.27	439.73	16.10	435.18	8.87	239.71
C210	15.59	421.35	15.43	416.99	8.50	229.70
C211	15.58	421.08	15.42	416.72	8.49	229.55
C212	16.10	435.14	15.93	430.63	8.78	237.21
C213	15.59	421.35	15.43	416.99	8.50	229.70
C214	16.79	453.78	16.62	449.09	9.15	247.38
C215	16.10	435.14	15.93	430.63	8.78	237.21
C216	15.93	430.54	15.77	426.09	8.68	234.71
C217	16.27	439.73	16.10	435.18	8.87	239.71
C218	17.13	462.97	16.95	458.18	9.34	252.39
C219	15.59	421.35	15.43	416.99	8.50	229.70
C220	16.44	444.32	16.27	439.73	8.96	242.22
C221	16.27	439.73	16.10	435.18	8.87	239.71
C222	15.59	421.35	15.43	416.99	8.50	229.70
C223	15.59	421.35	15.43	416.99	8.50	229.70
C225	14.56	393.51	14.41	389.44	7.94	214.52
C227	15.42	416.76	15.26	412.44	8.41	227.19
C226	15.07	407.30	14.91	403.08	8.22	222.03
C228	15.59	421.35	15.43	416.99	8.50	229.70
C229	15.59	421.35	15.43	416.99	8.50	229.70
C230	14.56	393.51	14.41	389.44	7.94	214.52
C170	15.59	421.35	15.43	416.99	8.50	229.70
C171	15.25	412.16	15.09	407.90	8.31	224.69

APPENDIX B: DOSE RATES IN CONTROLLED AREAS

Table B.1 Dose rate at Personnel Door (μSv/h)

Location	A	B	C	D	E
1	0.082±0.009	0.059±0.009	0.078±0.009	0.105±0.009	0.107±0.015
2	0.064±0.031	0.091±0.019	0.091±0.014	0.098±0.012	0.073±0.006
3	0.148±0.007	0.093±0.012	0.098±0.009	0.093±0.005	0.098±0.012
4	0.105±0.031	0.091±0.007	0.084±0.025	0.091±0.021	0.052±0.009
5	0.084±0.035	0.029±0.015	0.084±0.005	0.096±0.011	0.105±0.017
6	0.093±0.013	0.096±0.015	0.096±0.023	0.078±0.013	0.082±0.009
7	0.078±0.015	0.096±0.011	0.132±0.020	0.105±0.013	0.098±0.003
Average per point	**0.093±0.025**	**0.079±0.024**	**0.095±0.017**	**0.095±0.009**	**0.088±0.018**
Average	**0.090±0.020**				

Table B.2 Dose rate just before Personnel Door (μSv/h)

Location	A	B	C	D	E
1	0.117	0.085	0.111	0.150	0.153
2	0.091	0.131	0.131	0.140	0.104
3	0.212	0.134	0.140	0.134	0.140
4	0.150	0.131	0.120	0.131	0.075
5	0.121	0.042	0.121	0.137	0.150
6	0.134	0.137	0.137	0.111	0.117
7	0.111	0.137	0.189	0.150	0.140
Average per point	**0.135±0.036**	**0.114±0.033**	**0.136±0.024**	**0.136±0.012**	**0.126±0.026**
Average	**0.129±0.029**				

Table B.3 Dose Rate at Goods Door (μSv/h)

Location	A	B	C
1	0.112±0.009	0.078±0.030	0.076±0.014
2	0.111±0.010	0.114±0.034	0.150±0.057
3	0.152±0.014	0.137±0.016	0.106±0.014
4	0.106±0.019	0.083±0.005	0.055±0.009
5	0.140±0.011	0.121±0.005	0.091±0.016
Average per point	**0.124±0.018**	**0.106±0.023**	**0.098±0.032**
Average	**0.109±0.027**		

Table B.4 Dose Rate just before Goods Door (µSv/h)

Location	A	B	C
1	0.160	0.111	0.108
2	0.158	0.163	0.215
3	0.217	0.196	0.152
4	0.152	0.119	0.078
5	0.201	0.173	0.147
Average per point	**0.178±0.026**	**0.152±0.032**	**0.140±0.046**
Average	**0.157±0.039**		

Table B.5 Dose Rate Readings on Roof Plugs (µSv/h)

	Point One	Point Two	Point Three	Point Four	Point Five
	0.091	0.148	0.194	0.137	0.114
	0.103	0.137	0.182	0.125	0.137
	0.114	0.137	0.171	0.125	0.137
	0.137	0.114	0.171	0.137	0.125
	0.125	0.137	0.194	0.125	0.068
Average per point	**0.114±0.016**	**0.135±0.011**	**0.182±0.01**	**0.130±0.006**	**0.116±0.025**
Average	**0.135±0.030**				

Table B.6 Dose Rate Readings in Control Room (µSv/h)

	Point One/µSv/hr	Point Two/µSv/hr
	0.137	0.148
	0.091	0.114
	0.080	0.114
	0.103	0.125
	0.114	0.137
Average per point	**0.105±0.196**	**0.128±0.013**
Average	**0.116±0.020**	

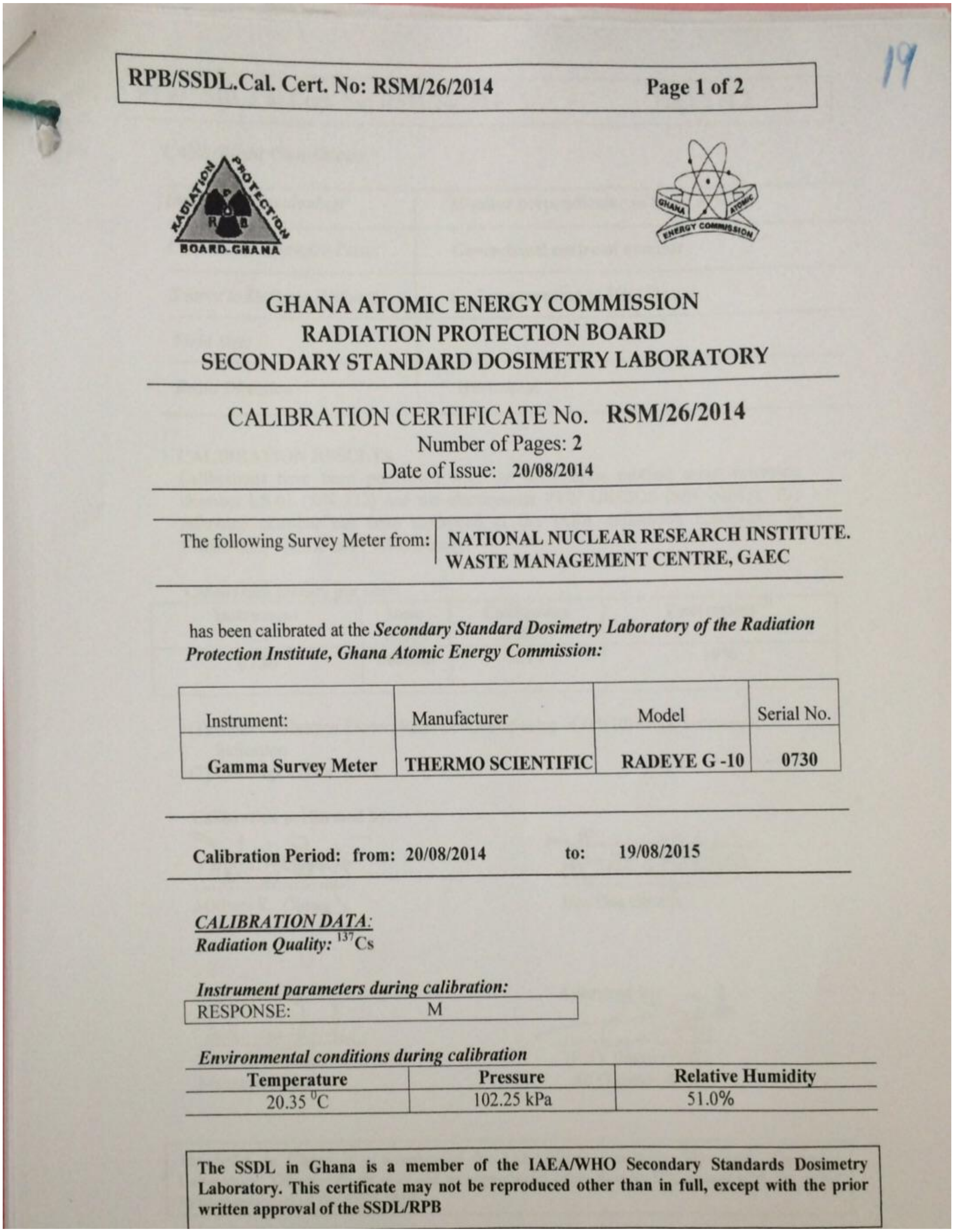
Table B.7 Dose Rate Readings in Electrical Room (µSv/h)

	Point One/µSv/h	Point Two/µSv/h	Point Three/µSv/h
	0.068	0.080	0.114
	0.091	0.080	0.114
	0.091	0.068	0.103
	0.091	0.057	0.091
	0.125	0.080	0.080
Average per point	**0.093±0.01**	**0.073±0.009**	**0.100±0.013**
Average	**0.089±0.02**		

Table B.8 Dose Rate Readings in Deionizer Room (µSv/h)

	Point One/µSv/h	Point Two/µSv/h	Point Three/µSv/h
	0.171	1.243	0.182
	0.194	1.311	0.217
	0.251	1.368	0.182
	0.376	1.436	0.228
	0.308	1.539	0.205
Average per point	**0.260±0.075**	**1.379±0.102**	**0.203±0.018**
Average per room	**0.614±0.550**		

APPENDIX C: RAD EYE G-10 CALIBRATION CERTIFICATE



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GHANA ATOMIC ENERGY COMMISSION
RADIATION PROTECTION BOARD
SECONDARY STANDARD DOSIMETRY LABORATORY

CALIBRATION CERTIFICATE No. RSM/26/2014
Number of Pages: 2
Date of Issue: 20/08/2014

The following Survey Meter from:	NATIONAL NUCLEAR RESEARCH INSTITUTE. WASTE MANAGEMENT CENTRE, GAEC

has been calibrated at the *Secondary Standard Dosimetry Laboratory of the Radiation Protection Institute, Ghana Atomic Energy Commission:*

Instrument:	Manufacturer	Model	Serial No.
Gamma Survey Meter	THERMO SCIENTIFIC	RADEYE G -10	0730

Calibration Period: from: 20/08/2014 to: 19/08/2015

CALIBRATION DATA:
Radiation Quality: ^{137}Cs

Instrument parameters during calibration:

RESPONSE:	M

Environmental conditions during calibration

Temperature	Pressure	Relative Humidity
20.35 ^{0}C	102.25 kPa	51.0%

The SSDL in Ghana is a member of the IAEA/WHO Secondary Standards Dosimetry Laboratory. This certificate may not be reproduced other than in full, except with the prior written approval of the SSDL/RPB

RPB/ SSDL.Cal.Cert. No: RSM/26/2014 Page 2 of 2

Calibration Conditions

Instrument positioning:	**Monitor perpendicular to beam axis**
Calibration reference Point:	**Geometrical centre of monitor**
Source to Detector Distance:	**1 – 7 m according to H*(10) rate**
Field Size:	
Beam Direction	**Horizontal**

CALIBRATION RESULTS

Calibrations have been performed with the substitution method using reference chamber LS-01 (S/N 227) and the electrometer PTW UNIDOS (S/N 20243). The reference chamber has been calibrated at the IAEA calibration Laboratory in Seibersdorf, Austria and PTB-Germany.

Calibration factors per scale

Instrument S/N	Dose range	Calibration factor(1)	Uncertainty(2)
0730	mSv/hr	1.14	10%

(1) Mean Calibration Factor: Ratio of the true value of H*(10) to the instrument indication

(2) Coverage Factor: k = 2.

Calibration performed by:

Michael K. Obeng

Ben Doe Gbekor

Supervised by:

Dr. J. K. Amoako
Manager, HPIC

Approved by:

E. O. Darko (Prof.)
Ag Director, RPI

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APPENDIX D: PERSONNEL DOSE HISTORY (2013)

Table D.1 Personal Dose Equivalent (mSv) : Strongly Penetrating Radiation - (Hp 10)

Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Total
0.23				0.52			0.70		0.66			2.11
0.22				0.59			1.03		0.71			2.55
0.21				0.48			0.77		0.86			2.32
0.20				0.51			0.67		0.82			2.20
0.21				0.45			0.58		0.55			1.79
0.24				0.52			0.68		0.73			2.17
0.25				0.64			0.73		0.63			2.25
0.20				0.24			0.60		0.63			1.67
0.22				0.77			0.92		0.46			2.37
0.21				0.42			0.77		0.62			2.02
0.21				0.26			0.78		0.96			2.21
0.47				0.84			0.78		0.60			2.69
0.22				0.50			0.75		0.89			2.36
0.21				0.58			0.76		0.85			2.40
									Total Cumulative Dose			31.11
									Average (mSv/yr)			2.222

Table D.2 Personal Dose Equivalent (mSv) : Weakly Penetrating Radiation - (Hp 0.07)

Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Total
0.17				0.42			0.58		1.02			2.19
0.16				0.58			0.80		0.77			2.31
0.20				0.65			0.89		1.11			2.85
0.14				0.36			0.64		0.70			1.84
0.18				0.45			0.53		0.66			1.82
0.17				0.48			0.97		0.80			2.42
0.22				0.68			0.77		0.65			2.32
0.17				0.34			0.57		0.97			2.05
0.17				0.90			0.96		0.41			2.44
0.15				0.39			0.68		0.57			1.79
0.15				0.39			1.11		0.66			2.31
0.41				0.78			1.12		0.51			2.82
0.15				0.43			1.09		0.82			2.49
0.15				0.57			0.96		0.78			2.46
									Total Cumulative Dose			32.11
									Average (mSv/yr)			2.294