ASSESMENT OF THE SHIELDING INTEGRITY OF Co – 60 GAMMA- RAY SCANNER AT AFLAO BORDER, GHANA.

This thesis is submitted to the:

DEPARTMENT OF NUCLEAR SAFETY AND SECURITY
SCHOOL OF ALLIED AND NUCLEAR SCIENCES
UNIVERSITY OF GHANA, LEGON

By

EDWIN CAPACITY AGBEMAFO
(10506731)
Bsc. (U.C.C, Cape Coast), 2008

In partial fulfilment of the requirements for the Award of the degree of

MASTEROF PHILOSOPHY

IN

RADIATION PROTECTION.

JULY, 2016

DECLARATION

This thesis is the result of research work undertaken by Edwin Capacity Agbemafo of the Department of Nuclear Safety and Security, University of Ghana, Legon, under the supervision of Dr. J. Owusu Banahene and Professor J.J Fletcher.

Sign ....
(Edwin Capacity Agbemafo) Date
Student

Sign ....
(Dr. J. Owusu-Banahene) Date
Principal Supervisor

Sign ....
(Professor J.J Fletcher) Date
Co-Supervisor

DEDICATION

This work is dedicated to my lovely wife Precious Mawunya Ahortor and my daughter Eyram Debrascha Ardeen-Capacity.

ACKNOWLEGMENT

I would like to express my profound gratitude to the Almighty God who gave me the ability to complete this work.

I am very grateful to my supervisory team; Dr. J. Owusu-Banahene and Prof. J.J Feltcher for their invaluable support, advice and encouragement especially during my research and thesis writing.

Special thanks go to my Head of Department, Dr. Augutine Faanu, also to Lecturers and Course mates at the Department of Nuclear Safety and Security.

My appreciation also go to Mr. Lawluvi Henry and Mr. Akrobortu Emmanuel all of Nuclear Regulatory Authority of Ghana, for their motivations and encouragements during my stay on campus.

My heartfelt appreciation goes to my siblings Messrs Kwaku Kpodo, Emmanuel Kpodo, Yawo Amable Kpodo, Christian Agbemafo, Felicia Agbemafo and Justine Agbemafo - who, as always, have supported and encouraged my education.

Finally, I would like to thank the management and staff, especially Samuel Mensah and Mr. Carl Lokko of NICK TC Scan Ltd, Aflao Border, for granting me the permission to carry out this research work on their facility.

LIST OF ABBREVIATIONS

ALARA-	As Low As Reasonably Achievable
BSS-	Basic Safety Standards
CCTV-	Closed Circuit Television
Gy-	Gray
HVL-	Half Value Layer
IAEA-	International Atomic Energy Agency
ICRP-	International Commission on Radiological Protection,
IDR-	Instantaneous Dose Rate
IPEM-	Institute of Physics and Engineering in Medicine
MeV-	Mega-electronvolt
mSv	milli-sievert
NCRP-	National Council on Radiation Protection and Measurements
POI-	Point Of Interest
SAD-	Source Axis Distance
SRS-	Safety Report Series
TADR-	Time Averaged Dose-equivalent Rate
TBq-	Tera-Becquerel
TVL-	Tenth Value Layer
USA-	United State of America

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

ABSTRACT

This study examines the current state of the shielding integrity of the 38.7 TBq Co-60 gamma ray scanner with an average energy of 1.25 MeV operated by NICK TC Scan Limited, which has been in use for destination inspection at Aflao Border of Ghana, for the past six years, (2010-2016). The facility uses a high energy ionizing radiation in its operation; therefore continuous adequacy of the installed biological shielding is critical to the protection and safety of the workers and the general public. The workload of the facility has increased since its commissioning, requiring the review of the status of the installed shielding. Theoretical calculations for dose rates and barrier thicknesses based on tenth – value- layer (TVL) concept and NCRP 151, 2005 recommendations, were done around the scanning facility using the current operational data. The results were then compared with the measured dose rates and the shielding thickness constituted during the commissioning stage, and international standards. Calculated dose rate at commissioning state ranges from 0.6μSv/hr to 2.4 μSv/hr with an average dose rate of 1. 43μSv/hr and that of the current operational state ranges from 1.1 μSv/hr to 2.6 μSv/hr with an average dose rate of 1.54μSv/hr, indicating an increase of 7.9%. Even though the dose rates were all below the recommended dose limit of 20μSvh^{-1} by NCRP, there has been an increase in dose to the staff and the general public. It has been observed that, the workload has increased three-fold from the commissioning stage to current operational state over the past six years. The assessment done on the installed shielding using the current operational data indicates that the shielding is inadequate in providing protection for the general public and the workers against X-ray radiation source of energy of at least 6MeV, and therefore the facility in its current state cannot be used to house a linear accelerator of energy up to 10MeV.

CHAPTER ONE

INTRODUCTION

1.0. Overview

This chapter gives a brief explanation of the research work carried out.

It touches on the background, the problem statement, the objectives of the research, relevance and the scope of the research work.

1.1. Background

Non-intrusive inspection of cargo containers has become a key issue in recent years to parry terrorist activities and to easily verify the shipping content at customs in order to curb contraband goods. Such systems are mainly based on X- and gamma-ray radiography which produces inside images of the cargos.

There is a worldwide need for effective scanning of cargo containers in order to detect possible contraband goods such as illicit drugs, explosives, nuclear materials or weapons, as well as verification of declared manifests. The manual inspection of large containers is not practical because of the time constraints and the high labour requirements for unpacking and repacking the cargo content. Therefore, there is a need for nonintrusive scanning systems.

Most non-intrusive cargo screening (Industrial radiography) systems are based on the use of a radiation (X- or gamma-rays). Those systems can provide high-resolution intensity images of the cargo contents and are well suited for detecting metal-based objects such as weapons. The images obtained from the systems are easy to interpret due to the high contrast shapes obtained from the scanning. The most used technologies are based on X and gamma-rays systems.

Industrial radiography is an established practice that provides benefits concurrent with radiation risks. The radiation protection objective, therefore, is to keep the risks As Low as Reasonably Achievable (ALARA) while maximizing benefits.

The purpose of shielding is to protect workers as well as the general public from the harmful effects of ionizing radiation. Therefore, adequate shielding design is essential during planning and building of such scanner facilities. (NCRP 47, 2002)

Effective shielding for the scanner units remains a crucial consideration when installing scanner facilities. This is to reduce primary and scatter radiation, hence, limiting radiation exposure to staff and the general public.

The radiation shields and design of the scanner facility describe how the required facility structure should be determined. The radiation shields of the rooms adjacent to the scanning equipment are usually thick to be able to attenuate high energy radiation associated with scanning equipment.

It is important to ensure that the shielding design is installed properly since retrofitting made after the room has been completed are very expensive.

When designing the radiation shield, the workload of the facility, the intended use and the occupancy of the room adjacent to the scanner facility are taken into consideration.

It is often impractical to make an overall experimental determination of the adequacy of the shielding adequacy prior to the completion of the building and installation of the radiation source. However, shielding voids may be detected by use of suitable portable gamma-ray detectors.

Poor design of scanner shielding can result in unnecessary exposures to the staff and the general public. These radiation exposures to the ionizing the radiation can result in two main detrimental health effects, these are Deterministic and Stochastic effects.

Deterministic effects occur at relatively large doses and are called deterministic because they are certain to occur, if the dose exceeds a threshold level. Deterministic effects are the result of various processes, mainly cell death and delayed cell division, caused by exposure to large radiation doses.

Severity of a particular deterministic effect in an exposed individual increases with the dose above the threshold for the occurrence of the effect.

Radiation exposure can also induce delayed effects such as malignancies and hereditary effects, which are expressed after a latency period and may be epidemiologically detectable in a population. This cancer induction, referred to as stochastic effects, usually takes place over the entire range of doses without a threshold level with the probability of occurrence of cancer being higher for higher doses and severity of cancer that may result from irradiation is independent of dose.

Stochastic effects may ensue if an irradiated cell is modified rather than killed, since modified cells may develop into a cancer after a prolonged delay.

If a cell damaged by radiation exposure is a germ cell whose function is to transmit genetic information to progeny, it is conceivable that hereditary effects of various types may develop in the descendants of the exposed individual.

The broad aim of optimization is to ensure that, the magnitude of the individual doses, the number of people exposed, and the likelihood of incurring exposures where these are not certain to be received, are all kept As Low As Reasonably Acceptable (ALARA), economic and social factors being taken into account.

The process of optimising protection is essentially source-related and should first be applied at the design stage of any project. This is done by ensuring that the installations and radiation sources are provided the best practical protection and safety measures possible.

For the purpose of implementing the principle of optimization and taking into account the exposures from different sources, the design of the shielding shall be based on dose limit.

The Basic Safety Series (BSS) defines dose limits the value of the effective dose or the equivalent dose to individuals in an exposure situations that is not to be exceeded.

When planning for the construction of gamma and X-ray scanner facilities, the dose limits are the doses in the controlled and supervised areas for which the facility is designed. (BSS 2014).

It is important that for effective radiological protection system, the principles of distance, time and shielding are all employed to ensure that all radiation doses are kept below acceptable dose limits.

Scanning rooms should also comply with both national and international regulations that deal with shielding requirement to render an installation safe from the radiation protection view point. The design also takes into consideration the structural building codes.

1.2 Statement of the Problem

Effective shielding for scanner units remains a crucial consideration when installing or remodelling containerized scanning facilities. This is to reduce primary and scatter radiation, hence, limiting exposure to staff and the general public.

The facility at the Aflao border uses a 3.8 TBq (103.4 Ci) Co-60 source. The enormous radiation energy from this source during beam-on state needs to be shielded in order to prevent undue exposure of the workers and the general public, therefore continuity and the integrity of the installed shielding of the facility are very important.

Since commissioning of the scanner units, no work has been done to re-evaluate the shielding designs. Workload of the facility may have increased with time, which may result in increased scan per day; number of working days per week which will require that the number of tenth value layers be increased.

In addition, concrete under radiation exposure, heats up and this drives the water out of its internal structure, resulting in swelling, cracking and spalling.

So as the facility ages, its shielding can develop cracks and voids due to the exposure to the gamma-rays.

It is advisable that during this deteriorating phase, an on-site concrete testing service be carried out at regular intervals.

It is against this backdrop that there is the need to conduct research on this facility to determine the current protection status of the installed barriers. The proposed research would undertake radiological assessment on the shielding to determined wether the installed shielding still provides the needed protection of the staff and the general public.

1.3. Objectives

In installing shield for radiation facility, it is advisable that the shielding design stage include consideration of possible future needs for new equipment, higher radiation energies and increased workloads.

The primary objectives of this research are:

- Assess the shielding integrity of the Co-60 scanning facility at Aflao Border by performing shielding calculations using current operational data of the facility, and to determine whether or not there has been any deterioration in the shielding since commissioning.
- Verify whether the shielding provides the needed protection for the staff and the general public in its current operational states.
- Analyse the suitability of the current shielding to house a higher energy up to 10 MeV X-ray scanners.
- Make appropriate recommendations to address any radiological health and safety problems identified.

1.4. Relevance of the Study

Cargo scanning facilities use ionizing radiation. It is therefore important that steps are taken to prevent overexposure to these radiations which usually results in radiation hazard, and this requires adequate shielding and periodic evaluation of the shielding adequacy of the scanning facilities.

The unit has been in operation for six years till now (2010 – 2016), the operational states of the facility has changed with significant variation in workload and with new structures built around the facility.

This study is relevant because re-evaluation of the shielding integrity of both theoretical and verification by measurement will provide the basis for determining the current status of protection and safety of staff and members of the public.

This work will also serve as a reference data for the Nuclear Regulatory Authority in enhancing the regulatory process.

1.5. Scope

This study is focused on assessment shielding integrity of 38.1TBq Co-60 gamma ray destination inspection facility operated by NICK TC Scan Limited at the Aflao border town of Ghana. Dose rates would be measured at critical areas around the facility. These rates would be compared with international values, and recommendations be made if there is any adverse findings in the operation of the facility.

1.6. Structure of the Thesis

This thesis work covers five chapters. Chapter One deals with the background to the study, problem statement, objectives of the study, relevance and justification of the research and scope of the work. In Chapter Two, a review of related literature in radiation is discussed. Chapter Three contains Research Materials and Methods used to assess the shielding integrity of the Co-60 gamma ray scanner. Chapter Four presents results and discussions on the findings from the research. Chapter Five presents Conclusions of the study and Recommendations from the findings of the research work.

The reference section provides the related literature citations used.

CHAPTER TWO

LITERATURE REVIEW

2.0. Introduction

This chapter presents a detailed overview of the work done in this research area in published literature. It discusses the history of shielding, decay scheme and properties of gamma rays, key considerations in structural design of gamma ray scanners, theory of shielding for gamma and X-ray scanners.

2.1. A brief history of radiation shielding

The hazards of x rays were recognized after some months of Roentgen's 1895 discovery, but dose limitation by time, distance, and shielding was at the discretion of the individual practitioner until about 1913. Only then were there organized professional efforts to establish guides for radiation protection, and not until about 1925 were there instruments available to quantify radiation exposure. In his survey of organization for radiation protection, Taylor (1979) begins with British and German efforts at establishing guidance for X-ray shielding. In 1913, the German Radiological Society on X-Ray Protection Measures issued recommendations that 2 mm of lead shielding was needed, regardless of generator voltage, workload, or filtration. In Britain, the Roentgen Society addressed radiation protection, stressing operator protection, the need for beam collimation, and the importance of scattered X rays. No explicit recommendations on shielding requirements were issued.

In 1921, the British X-Ray and Radium Protection Committee issued broad guidelines, both physical and administrative, on radiation protection in X-ray facilities. For diagnostic examinations, 2 mm of lead screening was recommended for the operator, as well as gloves with effectively 0.5 mm of lead shielding. For superficial therapy (up to 100 kV X- rays), 2 mm of lead shielding was recommended.

For deep therapy (in excess of 100 kV X- rays) 3 mm of lead shielding was recommended. Again, filtration and workload were not addressed. (Mutscheller, 1925) introduced important concepts in x-ray shielding. He expressed the erythema dose, *ED*, quantitatively in terms of the beam current *i* (mA), exposure time *t* (min), and source-to-receiver distance *r* (m), namely,

$$ED = 0.00368\frac{it}{r^2} \qquad 2.1$$

independent of X-ray energy. Years later, as observed by Taylor (1981), unit erythema dose was equated to an exposure *X* of about 600 R. Thus, in modern terms,

$$X = 2.2\,\frac{it}{r^2} \qquad 2.2$$

2.2 Destination Inspection.

Destination Inspection (DI) which falls under medico-legal exposure by virtue of the incorporation of radiological imaging without medical indication is a service whereby goods and import declaration are verified before exportation and also on arrival in the importing country.

The benefits of destination inspections include but not limited to:

- Avoidance of the unpacking and repacking of goods
- Inspection is completed quickly, leading to faster clearance
- Reduction in congestion at ports, harbours and country borders
- Increased in Customs revenues

- Detection of discrepancies between goods and declarations
- Reduced smuggling and fraud
- Improved security
- Increased compliance.(COTECNA, 2008)

2.3. Scanning Technology

Radiological imaging, which is carried out in destination inspection technology applies to non-destructive methods of inspecting and identifying goods in transportation system. It exploits a radiation source (X-ray, gamma rays or neutrons) to penetrate a cargo container and produce an image of its contents. The images are typically detecting the transmitted radiation on the other side of the cargo by different techniques: fluoroscopy or transmission which shows the object in superimposition. Computed tomography (CT) normally serves fore luggage control as it digital images without superimposition; and scatter radiations are used to identify chemical components. These images are assessed and matched with entry declaration invoices, bills of entry and packing list to determine whether they appear to reasonably represent the declared goods. Irregular or unexpected object may warrant further scanning and inspection to determine whether the objects are dangerous. Implied in this process is the need for image interpretation, which is typically performed by humans. Cargo scanning imagery may attain different levels of accuracy, depending on the type and specification of the selected technology. High penetration equipment, for instance, can both identify the nature of a cargo and allow quantity checks. (SGS, 2006). (Figure 2.1).

Figure 2. 1: Typical Non - Intrusive System Image (GAMMA RAY)

2.4. Radioactive Decay Scheme of Co-60

Co-60 is a radionuclide produced in a nuclear reactor by neutron activation of Co-59.

Co -60 is a beta emitter with maximum energy of 0.31MeV. Its half-life is 5.27 years and decay into an excited state of Ni-60. The excited nucleus of Ni-60 undergoes two photon emission in cascade, with energies 1.17 MeV and 1.33MeV. The average energy of the two cascades gamma photons is 1.25MeV. (Figure 2.2)

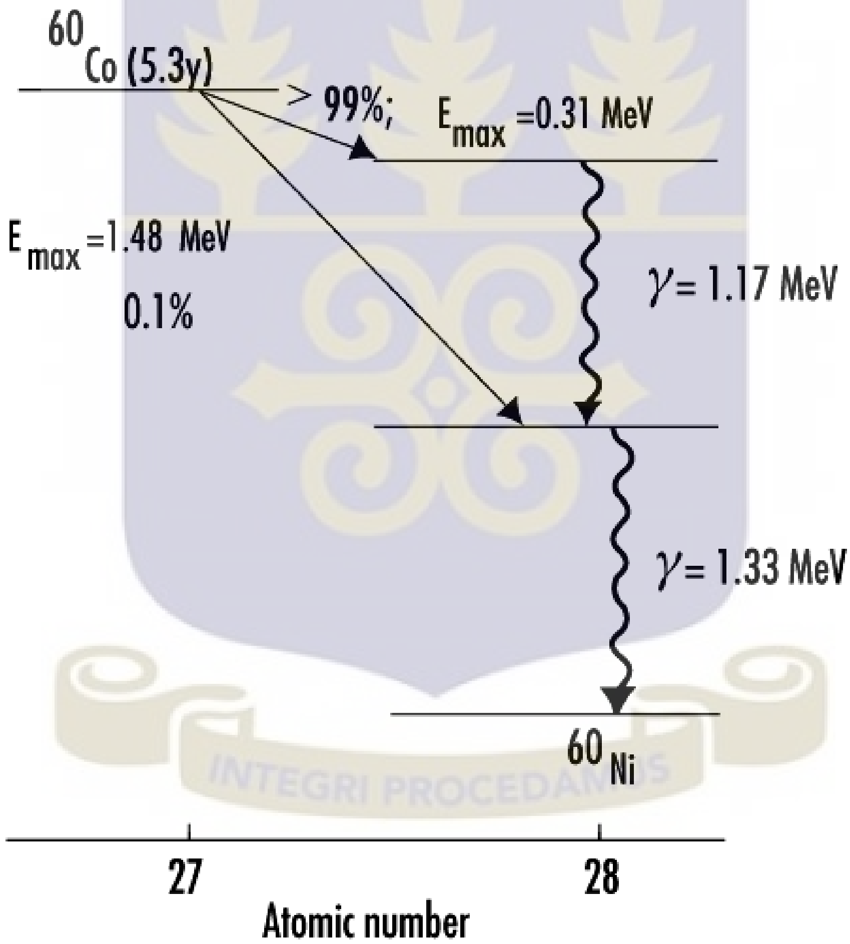

Figure 2. 2: Radioactive decay scheme for Co-60

2.5. Shielding materials

Radiation shielding is based on the principle of attenuation, which is the ability to reduce a wave's or ray's effect by blocking or bouncing particles through a barrier material. Charged particles may be attenuated by losing energy due to interactions with electrons in the barrier, while X-ray and gamma radiation are attenuated through either photoemission, scattering, or pair production. Neutrons can be made less harmful through a combination of elastic and inelastic scattering, and most neutron barriers are constructed with materials that encourage these processes.

The materials typically used for shielding purposes are ordinary concrete, heavy concrete, lead, steel, polyethylene, paraffin, earth and wood.

2.5.1. Concrete

Concrete has many advantages and is commonly used. It gives good X-ray and neutron shielding (concrete contains hydrogen), good structural strength and is relatively inexpensive. The required thickness to attenuate the incident photons to one tenth, the tenth-value layer (TVL), in concrete is much larger than for neutrons and therefore an adequate shielding for photons is more than enough for photoneutrons. For new buildings to house radiation scanner facilities, concrete will usually be the material of choice since it is the least expensive. However, if space is at a premium it may be necessary to use a higher density building material. (IAEA SRS 47, 2006)

2.5.2. Lead

Lead has a very high density and is a very good X- and gamma ray shielding material, but it will need support by a second material to handle the neutrons, which it is nearly transparent to, albeit that the high-energy neutrons get attenuated by inelastic scattering. Lead is also toxic and more expensive than concrete.

2.5.3. Steel

Steel is also expensive compared to concrete, but is not toxic. It is more efficient than concrete as an X-ray and gamma ray shielding material, but less efficient than lead. It is also nearly transparent to neutrons, but is a good structural material for shielding the neutron.

2.5.4. Polyethylene and Paraffin

These two materials are considered together since they are very similar. Paraffin, sometimes called paraffin wax, has the same percentage of hydrogen (14.3%) as polyethylene and is less expensive. However, it has lower density and is flammable so it is usually avoided in any permanent barriers. Polyethylene is perhaps the best neutron shielding material available, but it is relatively expensive. It is available both pure and loaded with varying percentages of boron to increase the thermal neutron capture. It is usually used for neutron shielding, and where space is at a premium (e.g., in doors) around ducts or in ceiling shields where insufficient height is available for less costly methods. It is easy to fabricate and relatively strong. Standard Borated polyethylene (BPE) contains 5 % boron by weight. (Fadlalla, 2010).

2.5.5. Earth

Earth is also commonly used as a shielding material by placing the accelerator room partially or entirely underground. Earth is not a well-defined material and its density can vary considerably. While earth composition is quite variable, it is not too different elementally from concrete and it is sufficient to consider it as equivalent to concrete with a density of 1.5 g cm^{-3}. A common problem with a room that is below is that, there may be a diagonal path up through mostly earth where the shielding is insufficient. Sand that is in contact with the earth will normally be damp enough that it can be treated the same as earth. Dry sand, however, such as bagged sand, needs some hydrogenous material beyond it in order to be an adequate neutron shield. (Fadlalla, 2010).

Table 2. 1: Shielding Materials and Their Densities. Source: IAEA Safety series report no. 47, 2006.

Building material	Density (kg·m3)	Comment
Concrete	2350	Will vary with mineral content
Barytes concrete	3400 - 3500	Most commonly used for dense concrete but expensive
Iron-ore with Ferrosilicon	4000 - 5400	Range of densities which depend on proportions of ore mixture to sand
Ledite	3844 and 4613	Pre-moulded high density interlocking blocks
Clay bricks	1600	May be used for installations up to 500 kV with supplementary lead or
Breeze blocks	1100 - 1400	Steel shielding
Earth fill	1600	May be useful if bunker is below ground level
Steel	7900	Normally used as a supplementary shielding on an existing treatment room

2.6. Shielding of Gamma Radiation

When gamma radiation is incident on a finite thickness of material, there exists some probability that the radiation will interact in the material and later be attenuated.

Thus, any of the common gamma interaction processes may result in secondary photons that have a finite probability of reaching the dose point. The extent to which such secondary photons add to the fluence or dose at the dose point is usually described through the use of an appropriate build-up factor.

Build-up factors may refer to various quantities of interest, such as photon fluence, photon energy fluence, exposure, or dose, and the values among all are somewhat different. Much of the available build-up data relates to determination of exposure or kerma in a small air volume envisioned to be located within the shielding medium of interest.

These data are also suitable for evaluation of dose to water or other low-Z material of interest.

The dose build-up factor is a dimensionless quantity that represents the ratio of total dose (including the dose from secondary photons) at the dose point to primary photon dose at the same point. The primary photon dose naturally comes from original photons that have penetrated the shielding material without interacting. Magnitudes of build-up factors vary widely, ranging from a minimum of 1.0 to very large values, depending on source and shield characteristics.

2.6.1. Good geometry shielding situation

When a narrow parallel beam of photons passes through a relatively thin shield, and if the dose point is many beam diameters away from the exit surface of the shield, we have a situation referred to in photon shielding as good geometry. This means, simply, that virtually all of the photons arriving at the dose point will be primary photons, and the dose, D, or dose rate, at a point of interest outside the shield, is related to the unshielded dose, D_o, or dose rate, at the point by

$$D = D_o e^{-\mu x} \quad 2.10$$

where:

μ is the linear attenuation coefficient for the photons of the energy of interest in the shield material, and x is the shield linear thickness.

2.6.2. Point isotropic source

The most popular source geometry involved in many calculations is the point isotropic source. While no real source is a true point, many sources are sufficiently small in dimensions that they can be treated mathematically as point sources. In practice, if the distance from source to dose point exceeds about three times the maximum source dimension and self-attenuation within the source volume is not a concern, the errors resulting from treating the source as a point will not exceed a few percentages. The assumption that the source is isotropic means that radiation of concern is emitted uniformly in all directions throughout 4π geometry. Consider a source of monoenergetic gamma radiation that emits *S* gamma rays per second and that is situated at a distance r (cm) from the dose point, and a shield of thickness T (cm) through which the gamma radiation passes before reaching the dose point.

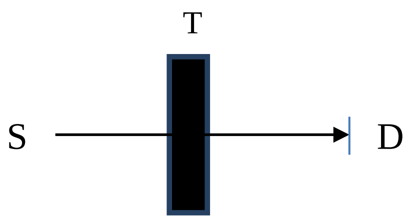

Figure 2.3: Point isotropic source shielding configuration

where

D is the dose point

S is the radiation source

2.6.2.1. Unshielded dose rate

The unshielded dose rate at the dose point is given by

$$\dot{D} = \frac{kSE\frac{\mu}{\rho}}{4\pi r^2} \qquad 2.11$$

where E is the photon energy, MeV,

$\frac{\mu}{\rho}$ is the mass energy absorption coefficient for the material at the dose point, cm^2 g^{-1}

$4\pi r^2$ is the beam area

k is a collective constant to convert energy fluence rate to dose rate; if the dose rate is in gray/hour, k will have a value of 5.76 x 10^{-7}.

2.6. 2.2 Shielded primary photon dose rate

The primary photon dose rate is attenuated exponentially, and the dose rate from primary photons, taking account of the shield, is given by

$$\dot{D} = \frac{kSE\frac{\mu}{\rho}e^{-\mu T}}{4\pi r^2} \qquad 2.12$$

where μ is the linear attenuation coefficient for the photons in the shield material. This expression does not account for the build-up of secondary radiation and will generally underestimate the true dose rate, especially for thick shields and when the dose point is close to the shield surface.

2.6.2.3 Shielded dose rate accounting for build-up

The added effect of the build-up is taken into account by incorporating a point isotropic source dose build-up factor, B, gives

$$\dot{D} = \frac{kSE\frac{\mu}{\rho}Be^{-\mu T}}{4\pi r^2} \qquad 2.13$$

The magnitude of the build-up factor depends on the photon energy, the shield material and thickness, the source and shield geometry, and the distance from the shield surface to the dose point. In most cases, dose build-up factors for point isotropic sources have been determined under the assumption that both the source and the dose point reside within an infinite volume of the shield material. As a consequence, shielded doses evaluated using such build-up factors tend to be conservative for most practical situations in which the dose point is outside the shield and not subject to backscattering from shield material behind the dose point. (Shultis & Faw 1996).When one wishes to determine the shield thickness to yield a specific dose rate, equation (2.13) cannot be solved explicitly for T because the value of B depends on T. Solutions can be obtained by making educated guesses for the value of T, looking up the corresponding values of B, and solving for the dose rates; results can be plotted, and the correct value of T determined for the desired value of dose rate. Alternatively, we can use an analytical form of the build-up factor that can be incorporated into equation (2.13) and, through an iterative process using a computer or calculator, and solve for the desired thickness. There are a number of algebraic expressions that have been used to represent B.

Among the most popular is an expression referred to as Taylor's form of the build-up factor, given by

$$B = A_1 e^{-\alpha\mu T} + (1-A_2)e^{-\beta\mu T} \qquad 2.14$$

Where A_1 and A_2μ and β are constants for a given energy and shield material.

2.7. Shielding for X-ray

Shielding for X-ray equipment is considered under the two categories;

- source shielding and
- structural shielding.

Source shielding is usually provided by the manufacturer of the X-ray tube housing.

For non-medical x-ray equipment, the tube housing and other parts of the X-ray apparatus, such as the transformer, are shielded to reduce the leakage X-ray radiation through the housing to acceptable levels.

Protective tube housing designed to limit the amount of leakage radiation are provided for all X-ray machines.

Leakage radiation, as used in these specifications for tube housings, means all radiation coming from the tube housing except for the useful beam.

Structural shielding for an X-ray facility provides protection from the useful or primary x-ray beam, from leakage radiation and from scatter radiation. It encloses both the x-ray equipment and the object being irradiated.

The amount of scatter radiation depends on the X-ray field size, energy of the useful beam, the effective atomic number of the scattering media and the angle between the incoming useful beam and the direction of scatter.

A key design parameter is the facility workload (W):

$$W = E \; x \; N_v \; x \; N_p \; x \; k \qquad 2.15$$

where W is the weekly workload, usually given in mA-min per week;

E is the tube current multiplied by the exposure time per view, usually given in mA s;

N_v is the number of views per cargo irradiated;

N_p is the number of cargo per week and

k is a conversion factor (1 min divided by 60 s).

Another key design parameter is the use factor U for a wall (or floor or ceiling) n. The wall may be protecting any occupied area such as a control room, office or waiting room.

The use factor is given by:

$$U = \frac{N}{N_v} \qquad 2.16$$

where, N is the number of views for which the primary X-ray beam is directed toward wall n.

The structural shielding requirements for a given X-ray facility are determined by the following:

- the maximum tube potential, in kilovolts-peak (kVp), at which the X- ray tube is operated.
- the maximum beam current, in mA, at which the X- ray system is operated
- the workload (W), which is a measure, in suitable units (usually mA-min per week), of the amount of use of the X-ray system
- the use factor (U), which is the fraction of the workload during which the useful beam is pointed in the direction of interest
- the occupancy factor (T), which is the factor by which the workload should be multiplied to correct for the degree or type of occupancy of the area to be protected
- the maximum permissible dose equivalent rate (P) to a person for controlled and non-controlled areas (typical absorbed dose limits are 1 mGy for a controlled area in one week and 0.1 mGy for a non-controlled area in one week)
- type of shielding material.
- the distance (d) from the source to the location being protected.

With these considerations included, the value of the primary beam transmission factor B in mGy per mA-min at one metre is given by:

$$B = \frac{Pd^2}{WUT} \qquad 2.17$$

Shielding of the x-ray facility must be constructed so that protection is not impaired by joints; by openings for ducts, pipes and so on, that pass through the barriers; or by conduits, service boxes and so on, embedded in the barriers. The shielding should cover not only the back of the service boxes, but also the sides, or be extended sufficiently to offer equivalent protection. Conduits that pass through barriers should have sufficient bends to reduce the radiation to the required level. Observation windows must have shielding equivalent to that required for the partition (barrier) or door in which they are located.

Radiation scanning facilities may require door interlocks, warning lights, closed circuit television or means for audible (e.g. voice or buzzer) and visual communication between anyone who may be in the facility and the operator.

Protective barriers are of two types:

- Primary protective barriers, which are sufficient to attenuate the primary (useful) beam to the required level
- Secondary protective barriers, which are sufficient to attenuate leakage, scattered and stray radiation to the required level.

In a design of secondary protective barrier, the required thickness to protect against each component is calculated separately. If the required thicknesses are about the same, an additional HVL is added to the greatest calculated thickness. If the greatest difference between the calculated thicknesses is one TVL or more, the thickest of the calculated values will suffice.

Scattered radiation intensity depends on scattering angle, energy of the useful beam, field size or scattering area, and subject composition.

When designing secondary protective barriers, the following assumptions are made:

- When X-rays are produced at 500 kV or less, the energy of the scattered radiation is equal to the energy of the useful beam.
- After being scattered, the X-ray energy spectrum for beams generated at voltages greater than 500 kV are degraded to that of a 500 kV beam, and the absorbed dose rate at 1 m and 90 degrees from the scatterer is 0.1% of that in the useful beam at the point of scattering.

The transmission relationship for scattered radiation is written in terms of the scattering transmission factor (B_s) with units of mGym^2 (mA-min)$^{-1}$:

$$B_s = \frac{P d_{sec}^2 d_{sca}^2}{\left(\frac{F}{400}\right) \alpha W T} \qquad 2.18$$

where P is the maximum weekly absorbed dose rate (mSv),

d_{scat} is the distance from the X- ray tube's target and the scanner,

d_{sec} is the distance from the scatterer to the point of interest that the secondary barriers are meant to shield,

α is the ratio of scattered radiation to incident radiation,

F is a factor accounting for the fact that X ray output increases with voltage.

Smaller values of *B* require thicker shields.

2.8. The classification of work areas

Approach to exercising control of the source of exposure and over the workers who are occupationally exposed has been recommended by the ICRP (1990). It involves formally designated proximal localities to the source into controlled and supervised areas (ICRP 1990). A controlled area is one in which access may be restricted during periods of irradiation, requiring workers to follow well-established procedures and practices aimed specifically at strictly controlling radiation exposures. In identifying a controlled area, it may be useful to make use of existing physical boundaries, such as the external walls of the facility and/or room walls.

A supervised area shall be designated in any area not already designated as a controlled area, but where occupational exposure conditions need to be kept under review even though specific protection measures and safety provisions are not normally needed.

The ICRP 1990 recommend that the classification of work areas are best based on operational experience and judgement, taking into account both expected exposure levels and of the likely variations in these exposures. Designation of areas may be defined in terms of the dose rate at the boundary, with the aim of ensuring that anyone outside a designated area will not need to be regarded as occupationally exposed. Thus, doses received outside the designated areas should be low enough to ensure that in normal conditions, the level of protection for those who work in the facility will be less than the dose constraint of 1 mSv per year.

Values of dose rate based on a fraction of the relevant dose limit have often been used in the past for defining the boundaries of controlled areas. However, careful consideration should be taken of the length of time for which the dose rate remains at or above the defined level and the risk from potential exposures. Periodic review of the exposure situation may necessitate the need for extra protective measures or changes to the boundary of a work area.

There should be signs at entrances to controlled area to indicate to staff, especially to maintenance staff, the special procedures currently applied in the area and that a radiation source is likely to be present (ICRP 1999).

2.9. Occupancy Factors

Evaluation of strategically shielded scanner facility must certainly involve an estimation of occupancy for all adjacent areas. The occupancy factor for each selected monitoring site should represent the average time that the maximally exposed individual is present when the radiation is present.

NCRP No. 147 (2004) provides a list of suggested occupancy factors that are to be used as a guide in calculations when accurate occupancy data are not available.

2.10 Use Factor

The use factor describes the different beam orientation used for treatment when calculating the required barrier thickness for each beam orientation. If conventional treatment techniques are to be used, NCRP Report 49 suggests a use factor of 1 for the floor with the beam pointing vertically down and0.25 for each wall and ceiling if specific values are not available. These use factors may depend on the particular use of the facility and also on the energy used. For example, a facility performing a large number of total body irradiations may have a use factor greater than 0.25 for one wall, and lower for other walls. (NCRP, 2004)

2.11. Assessment of Occupational Exposure

During the implementation or evaluation of a radiation protection plan there should be tracking of changes to plan outcome indicators enabling the opportunity for feedback. This routinely occurs for occupational exposure in the form of individual monitoring or in the form of a radiation surveying.

In the characterization of occupation exposure, the main indicators to be examined are the level of the collective dose and the distribution of individual doses (IAEA 2002). The assessment of collective dose from occupation exposure is usually based on the recorded dose from individual monitoring. Such information provides confidence and may provide data of use in reviewing the radiation protection program. Workers exposed to radiation from sources not related to, or required by their work shall receive the same level of protection as if they were members of the public.

A formal individual dose assessment would normally be performed should monitoring indicate that the corresponding annual effective dose exceeds 1mSv, and should certainly be conducted in total annual effective doses estimated to be above 5 mSv (IAEA 1999).

In the case of shielding design, goals are used in the evaluation of the effectiveness of barrier construction for the required level of protection. Ideally one should perform a radiation survey when the source is of greatest activity, sampling nearby controlled and uncontrolled areas. In addition to providing information on the control of occupational exposure, a program of individual monitoring may be helpful in confirming the classification of work areas. For example, when combined with data on the frequency of the jobs performed, their duration, measured dose rates and the number of workers exposed. In practice, it is possible to achieve an accuracy of about 10% at the 95% confidence level for measurements of radiation fields in good laboratory conditions. The NCRP recommend the setting of a weekly shielding design goal for a area at an equivalent dose value of 0.1 mSv per week and 0.02 mSv per week for an uncontrolled area.

Measurements should be made at a number of occupied distances from the source, including outside the facility (e.g. on the roof or an adjacent footpath). The measurements should be taken from the source to the nearest likely approach of the sensitive organs of a person to the barrier (Podgorsak, 2000). The NCRP recommend a measurement point of no closer than 30 cm to a wall. For the floor below, not higher than 1.7 m from that floor level and, for ceiling transmission, a distance of at least0.5 m above the floor of the room above. (Podgorsak, 2006)

If, as a result of a radiation survey, supplementary shielding is added to the protective barriers, the survey should be repeated to evaluate the adequacy of the shielding after the modification.

2.12.0 Calculation method

For radiation safety purposes, the barriers thicknesses must be designed to attenuate the primary, leakage and scatter radiations.

2.12.1 Primary barrier shielding.

The primary barrier is the shielding used for the attenuation of the primary beam. It is givens as, (McGinley, 2002)

$$Bp = \frac{P(d+SAD)^2}{WUT} \qquad 2.19$$

P = the design limit for a public area which is $0.02 mSvhr^{-1}$

d = the source axis to the point behind barrier

SAD = source axis distance

W = workload

U = use factor

T = occupancy factor

The workload W, the average absorbed dose of radiation produced by a source in a specified time (mostly in a week) and expressed as;

$$W = d_s \, x \, S_n \, x \, d_n \qquad 2.20$$

Where

d_s is dose per scan

S_n is number of scan per day

d_n is number of scan days per week.

The required number (n) of tenth layer value (TVL) of concrete needed to shield the primary radiation from the source is determined by the equation.

n = - $(logB_p)$ 2.21

therefore the thickness(t_p)of concrete required to attenuate the primary beam to the desired level is:

$t_p = n \, x \, TVL \, (for \, 1.25 \, Mev \, primary \, \gamma \, ray \, in \, concrete)$ 2.22

2.12.2 Secondary barrier shielding.

2.12.2.1. Scattered radiation

The secondary barrier must shield against both the leakage and scatter radiations. As described in section 2.2.3, at megavoltage energies the dominant photon interaction is the Compton effect. In the Compton interaction, the photon is not absorbed, rather it is scattered at an angle and with reduced energy. Both the leakage and scatter radiations undergo Compton scattering.

The leakage beam scatters multiple times, and thus loses significant energy, while traversing the dense material (typically lead) in the head of the radiation generator. The scatter radiation, by definition, is scattered at least once before reaching the secondary barrier.

Two main sources of scatter radiation are possible:

- scatter radiation produced by scattering of the primary beam within the cargo,
- scatter radiation produced by scattering of the primary and cargo scatter radiations with the walls and fixtures of the scanning tunnel.

A secondary barrier with thickness adequate to attenuate scatter radiation from the primary beam will generally be sufficient for cargo scatter radiations, since multiple scattering interactions reduce photon energy and penetrability significantly.

Barrier thicknesses are calculated for the leakage and patient-scatter beams independently due to the difference in energy between them. Leakage radiation, having undergone multiple scattering within the head of the scanning machine, is typically of a much lower energy than the cargo-scatter radiation, which is scattered less often before reaching the walls.

The barrier transmission (B_s) required to shield against scattered radiation from the container is given by the equation (NCRP 151, 2005), (McGinley, 2002)

$$B_s = \frac{Pd_{sec}^2 d_{sca}^2}{(\frac{F}{400})\alpha WT} \qquad 2.23$$

Where

α is the scatter fraction or the fraction of the primary beam absorbed dose that scatters at a particular angle.

$dsca$ is the distance from the primary radiation source to the scattering material, usually taken as the SAD of the machine.

$dsec$ is the distance from the scattering point to the point of interest (POI).

F is the incident field area (in cm^2) on the scatterer.

The number (n) of TVL required to shield the scattered radiation from the container is given by, (McGinley, 2002)

$$n = \log(\frac{1}{B_s}) \qquad 2.24$$

The required thickness, t_s, for the attenuation of the scattered radiation is

$$t_s = n\ x\ TVL\ (for\ 1.25\ Mev\ primary\ \gamma\ ray\ in\ concrete) \qquad 2.25$$

2.12.2.2. Leakage radiation shielding

Based on NCRP 151, 2005, the required transmission barrier is given by the equation

$$B_l = \frac{1000Pd_s^2}{WT} \qquad 2.26$$

P is the design limit for a public area which is $0.02mSvhr^{-1}$

d_s is the source axis to the point of interest behind barrier

W = workload

T = occupancy factor

The number (n) of TVL required shielding the leakage radiation from the scanner head is given by

$$n = \log(\frac{1}{B_l}) \qquad 2.27$$

The required thickness, t_l, for the attenuation of the leakage radiation is

$$t_l = n\ x\ TVL\ (for\ 1.25\ Mev\ primary\ \gamma\ ray\ in\ concrete). \qquad 2.28$$

2.13.0 Average instantaneous dose rate calculations.

In the IAEA SRS 47 report, the instantaneous dose rate (IDR) is defined as the direct reading of a survey meter that gives the dose per hour, averaged over a minute.

2.13.1 Instantaneous dose rate due to the primary radiation (D_p)

$$IDR_P = \frac{60DR_mB_e}{d^2} \qquad 2.29$$

Where

DR_m is the maximum dose rate output at 1.0m is 0.8Gy/min.

B_e is the primary barrier transmitted factor given by the relation

$$B_e = \frac{1}{10^{t_e/\,TVL}} \qquad 2.30$$

Where t_e is the thickness of the primary barrier

2.13.2 Dose rate from the scattered radiation.

The dose from the scattered radiation is given by,

$$IDRs = \frac{60\alpha\, DR_m(\frac{F}{400})B_{es}}{d_{sec}^2 d_{sca}^2} \qquad 2.31$$

Where $B_{es} = \frac{1}{10^{t_e/\,TVL_S}}$ 2.32

α is the scatter fraction or the fraction of the primary beam absorbed dose that scatters at a particular angle.

$dsca$ is the distance from the primary radiation source to the scattering material. Usually taken as the SAD of the machine.

$dsec$ is the distance from the scattering point to the point of interest (POI).

F is the incident field area (in cm^2) on the scatterer. (NCRP 151, 2005)

2.13.3 Dose from the leakage radiation

The dose from the leakage radiation is given by

$$IDR_l = \frac{60\alpha\, DR_m(\frac{F}{400})B_{el}}{d_{sec}^2 d_{sca}^2} \qquad 2.33$$

Where $B_{el} = \frac{1}{10^{t_e/\,TVL_l}}$ 2.34

α is the scatter fraction or the fraction of the primary beam absorbed dose that scatters at a particular angle.

$dsca$ is the distance from the primary radiation source to the scattering material. Usually taken as the SAD of the machine.

$dsec$ is the distance from the scattering point to the point of interest (POI).

F is the incident field area (in cm^2) on the scatterer.

Total dose rate for leakage and scatter radiation is defined by the equation;

$$IDR_T = IDR_l + IDR_s \qquad 2.35$$

2.14.0 Time Averaged Dose-Equivalent Rates

When designing radiation shielding barriers it is usual to assume that the workload will be evenly distributed throughout the year. Therefore, it is reasonable to design a barrier to meet a weekly value equal to one-fiftieth of the annual shielding design goal (NCRP, 2004). However, further scaling the shielding design goal to shorter intervals is not appropriate and may be incompatible with the ALARA principle. It is more useful if the workload and use factor are considered together with the *IDR* when evaluating the adequacy of a barrier.

For this purpose, the concept of time averaged dose equivalent rate (TADR) is used.

The TADR is the barrier attenuated dose-equivalent rate averaged over a specified time or period of operation. TADR is proportional to *IDR*, and depends on values of W and U. There are two periods of operation of particular interest to radiation protection, the week and the hour. (NCRP 151, 2005)

The maximum dose in-any-hour ($TADR_h$) due to leakage and scattered gamma radiation is given by:

$$\frac{W_h IDR_T U}{DR_m} \qquad 2.36$$

Where W_h is workload in one hour and the rest of the symbols have their usual meaning as stated earlier in the text.

2.15.0 Calculation of required barrier thickness for X – ray of energy 6 MeV.

2.15.1 Primary barrier shielding.

The primary barrier is the shielding used for the attenuation of the primary beam.

It is given as $B = \frac{Pd^2}{WUT}$ (McGinley, 2002)

P is the design limit for a public area which is $0.02 mSvhr^{-1}$

d = the source axis to the point behind barrier

SAD = source axis distance

W = workload

U = use factor

T = occupancy factor

The workload W, the average absorbed dose of radiation produced by a source in a specified time (mostly in a week) and expressed;

$$n = -(logB) \qquad (2.31)$$

therefore the thickness(t_p) of concrete required to attenuate the primary beam to the desired level is:

$t_p = n\ x\ TVL\ (for\ 6\ MeV\ primary\ gamm\ ray\ in\ concrete)$

2.16.0 Secondary barrier shielding

2.16.1 Scattered radiation

The barrier transmission (B_s) required to shield against scattered radiation from the container/cargo is given by the equation (NCRP 151. 2005)

$$B_s = \frac{P d_{sec}^2 d_{sca}^2}{(\frac{F}{400}) \alpha W T} \qquad 2.33$$

Where

α is the scatter fraction or the fraction of the primary beam absorbed

dose that scatters at a particular angle.

$dsca$ is the distance from the primary radiation source to the scatteringmaterial.

$dsec$ is the distance from the scattering point to the point of interest (POI).

F is the incident field area (in cm^2) on the scatterer.

The number (n) of TVL required to shield the scattered radiation from the container is given equation n = - $(log B_s)$

The required thickness, t_s, for the attenuation of the scattered radiation is

$t_s = n\ x\ TVL\ (for\ 10\ Mev\ primary\ x - ray\ in\ concrete$

CHAPTER THREE

MATERIALS AND METHODS

3.0 Introduction

This section describes the method and the materials used in the assessment of the scanner.

The layout of the facility, distance measurements, dose measurements at critical locations around the facility, primary and secondary barrier thickness calculations are describe in this section.

3.1 Materials

The gamma ray scanner facility at Aflao is one of the scanning facilities operated by NICK TC Scan limited of Ghana.

The facility was built in October, 2010. The primary shield is made of concrete of density 2350 kgm^{-3}. Its thickness was measured to be 500 mm.

The secondary shield is also made of concrete of thickness 430 mm. The scanning tunnel is 30 m long

The scanning method is such that the scanned container or cargo remains still whilst the scanning unit moves at a speed of 12m/min. The scanning time for each cargo is 90 seconds.

Safety systems installed at the facility helps determine whether anyone is in the scanning tunnel area or not during scanning.

A motion sensor at both ends of the tunnel triggers the retraction of the source into its shield when it detects an intrusion during beam-on state.

CCTV cameras are also installed in the tunnel for monitoring the tunnel before and during the scanning process.

Other safety systems at the facility include; safety confirmation devices, an alarm, radiation symbol on the door, and traffic marshals at both ends of the tunnel.

3.2 Technical specification of the facility.

Manufacturer: Beijing Huahixing Science Tech-Development Co. Ltd.

Model: TC-SCAN FMG

Serial Number: HFMG01405032

Scan speed: 6-18m/min

Power rating 30KVA

Target W x H: 3m x 4.6m

Date of manufacture: January, 2006

Radionuclide: Co-60

Half-life: 5.7 years

Initial activity: 3.8 TBq (103.4 Ci)

Dimension of the tunnel: L x B: 30m x 18 m

3.3 Area of Survey

Figure 3.1 shows the outline of the NICK TC Scan facility at the Aflao Border. The areas where the survey was carried out are the Manager's Office, the Secretary's Office, Control Room, Check-in, Reception, Washrooms and the Marshal post.

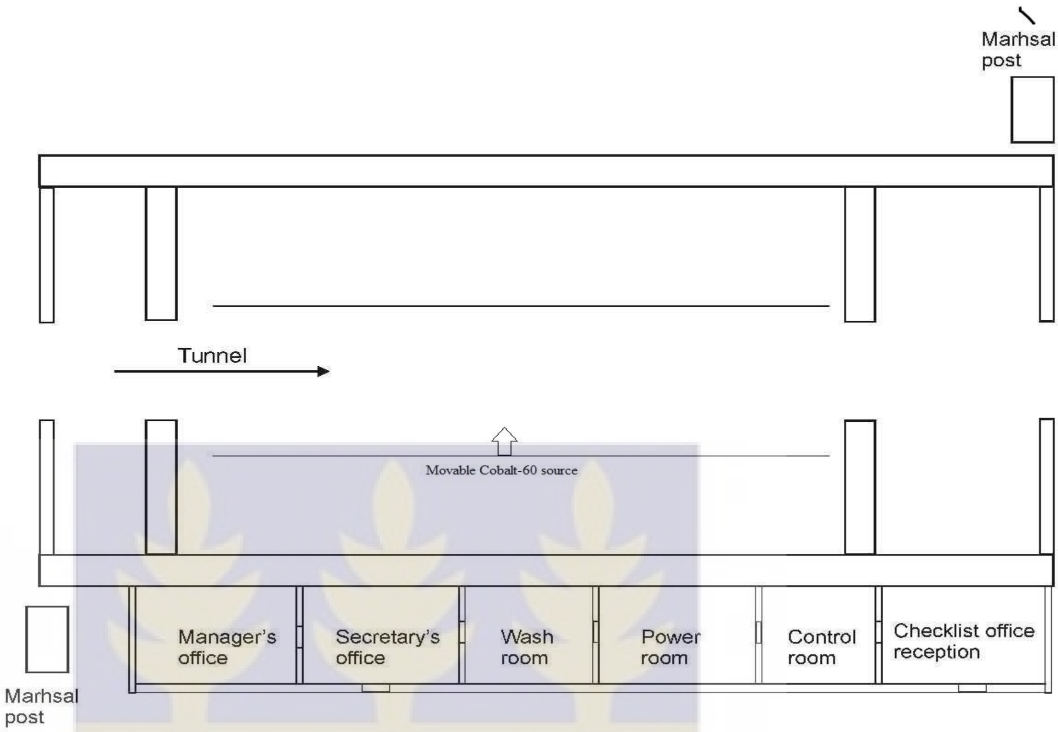

Figure 3. 1: Layout of the NICK TC Gamma ray scan facility.

3.4 Method

Measurements of distances were taken using surveyor's metre, from the source axis and the scatterer.

The following points of interest were considered

- Source axis to the point of interest for the primary barrier shielding.
- Source axis to the scattered for the scattered radiation shielding..
- Container to the point of interest for scattered radiation shielding.

3.5.0 Instruments used

3.5.1 Dose rate meter

The survey meter used in this work is a Rad-eye G10 which was manufactured in Germany by Thermo electron Scientific Inc.

It is a portable radiation detection and measurement instrument used to check personnel, equipment and facilities for radioactive contamination or to measure external or ambient ionizing radiation fields. The instruments are designed to be handheld, are battery powered and of low mass to allow easy manipulation. Other features include an easily readable display, in counts or radiation dose, and an audible indication of the count rate. These features, among others makes this detector one of the most used detectors.

The Rad-eye G10 gamma survey meter (Fig 3.2) is specifically designed for gamma survey from background up to personal safety levels and used by first responders, in the nuclear power industry, radiography and for radiation protection purposes.

Figure 3. 2: Rad eye G-10 gamma survey meter

Technical specifications of the rad-eye G10 survey meter

Model	Rad-Eye G10
Serial number	46506/76 F-Nr0738
Country of Origin	Germany
Detector	Energy compensated GM tube
Measuring range	0.5 µSv/h -100 mSv/h
Energy range	50keV- 3MeV

3.6. Shielding barrier calculations

Two types of radiation reach the walls of the room and must be accounted for in shielding design calculations are;

- Primary beam (either attenuated through the container or unattenuated) and
- Secondary radiation arising from leakage through the shielded head of the radiation generator and from scattered radiation produced by the interaction of the primary beam within the container. Since the energy of the primary and secondary beams differ significantly, they must be considered independently and require separate shielding design calculations.

3.6.1. Calculation of the workload for the gamma ray source.

The workload was calculated using equation (2.20)

3.6.2. Primary Barrier Shielding for the Gamma Source

The barrier attenuation factor need to shield against primary radiation was calculated using equation (2.19).

Since there are no structures around the space behind the primary barrier, the only group of people affected by transmitted beam outside the barrier is the public.

The number of tenth-value layers of concrete required to shield against primary radiation was also determined using the equation (2.21)

The thickness, $t_{p,}$ of concrete required to shield against the primary radiation is given by t_p = number of TVLs x TVL in concrete for 1.25 MeV primary γ concrete.

The TVL for concrete (density 2350 kgm^{-3}) for gamma radiation is 218 mm.

3.6.3.0 Secondary barrier shielding for the 1.25 MeV Gamma-ray Scanner.

3.6.3.1. Leakage Gamma Radiation

The areas outside the shielding where people who are affected by leakage radiation are located are the Manager's office, the secretary's office, control room, check-in, the reception, the washrooms and the marshal post.

The required attenuation to shield against leakage radiation from the scanner head for the above named areas was calculated using equation (2.26).

The number of tenth value layer of needed to shield against the leakage gamma radiation was calculated using equation (2.21)

The thickness of required to shield against leakage radiation was calculated using equation (2.22).

3.6.3.2 Scattered gamma radiation

The areas outside the shielding where people who are affected by scattered radiation are located are the manager's office, the secretary's office, control room, check-in, the reception, the washrooms and the marshal post.

The needed barrier thickness to effectively shield the scattered gamma ray from the container was determined using equation (2.23).

The number of tenth value layer of needed to shield against the scattered gamma radiation was calculated using equation (2.21)

The thickness of required to shield against scattered radiation was calculated using equation (2.22). ie

$t_s = number\ of\ TVLs\ x\ TVL\ (for\ 1.25\ Mev\ primary\ \gamma\ ray\ in\ concrete)$

Where the TVL for 1.25 MeV in concrete of density 2350 kgm^{-3} for scattered radiation is 151 mm.

3.7. Average instantaneous Dose Rate Calculation for the primary Gamma ray radiation

Primary barrier transmission factor required was calculated using equation 2.30i.e.

$$B_e = \frac{1}{10^{t_e/TVL}}$$

Where t_e is thickness of the primary barrier employed which was measured to be 500 mm.

TVL for the gamma ray for the concrete used is 218 mm.

The dose rate for the primary radiation was determined using equation (2.29)

$$D_p = \frac{60DR_m B_e}{d^2}$$

Where

DR_m is the maximum dose rate output at 1.0 m is 0.8Gy/min.

3.8. Instantaneous dose rate from leakage gamma ray radiation.

The expected transmission factor was determined using

$$B_{el} = \frac{1}{10^{t_e/\,TVL}}$$

Where the barrier thickness used for the leakage is 430 mm, and the TVL for the concrete used for the leakage radiation is 218 mm.

The dose rate from the leakage radiation was calculated using equation

$$IDR_l = \frac{60\alpha\, DR_m(\frac{F}{400})B_{el}}{d_{sec}^2 d_{sca}^2}$$

3.9. Instantaneous dose rate from scattered gamma ray radiation.

The needed scatter transmission factor from the container was determined using the equation

$$B_{es} = \frac{1}{10^{t_e/\,TVL_S}}$$

Where t_e is given as 430 mm; and the TVL for the scatter radiation for the concrete used is 151 mm.

The dose rate due to the scatterer was calculated using equation 2.31

ie

$$IDRs = \frac{60\alpha\, DR_m(\frac{F}{400})B_{es}}{d_{sec}^2 d_{sca}^2}$$

Where

$$B_{es} = \frac{1}{10^{t_e/\,TVL_S}}$$

3.10.0 Barrier thickness consideration for X-ray radiation of energy 6MeV

3.10.1 Barrier thickness for primary X-ray beam of energy 6MeV

The barrier needed factor required to shield against primary X-ray was determined using equation (2.17)

$$Bp = \frac{Pd^2}{WUT}$$

P is the design limit for a public area which is 0.02mSv/hr

d = the source axis to the point behind barrier

SAD = source axis distance

W = workload (Gy/wk)

U = use factor

T = occupancy factor

The tenth value layer was also calculated using equation

n = - $(logB_p)$

where Bp is attenuation barrier determined.

The needed shielding thickness was calculated using equation

t_p = number of TVLs x TVL in concrete for 6MeV primary X- ray radiation.

The TVL for concrete (density 2350 kgm^{-3}) for x-ray radiation is 343 mm.

3.10.2 Barrier thickness for leakage radiation for 6MeV X-ray radiation.

The areas outside the shielding where people who are affected by leakage radiation are located are the manager's office, the secretary's office, control room, check-in, the reception, the washrooms and the marshal post.

The barrier transmission, *B*, needed to attenuate the radiation to the scatterer acceptable level was calculated using the equation

$$B_S = \frac{P d_{sec}^2 d_{sca}^2}{(\frac{F}{400}) \alpha W T}$$

where

P is the maximum weekly absorbed dose rate (mGy),

d_{scat} is the distance from the x ray tube's target and the scanner,

d_{sec} is the distance from the scatterer to the point of interest that the secondary barriers are meant to shield,

α is the ratio of scattered radiation to incident radiation,

F is a factor accounting for the fact that x ray output increases with voltage. Smaller values of *B* require thicker shields.

CHAPTER FOUR

RESULTS AND DISCUSSION

4.0 Introduction

The results obtained from the research are presented and discussed in this section. The discussions are in the context of the international standards found in the reviewed literature.

4.1 Primary barrier thickness for the gamma ray scanner.

Table 4. 1: Calculated and measured primary barrier thickness for the gamma ray scanner

POI	*P (Sv/wk)*	*d/m*	*W*	*U*	*Be*	*nTVLs*	*Calculated thickness (mm)*	*Designed Thickness (mm)*
Behind the primary barrier	2.0E-5	15.8	0.06	0.25	3.08E-2	1.51	329.3	500.00

The group of people that are affected by transmitted beam through the primary barrier is the public, since there are no structures behind the barrier.

The nearest distance of approach behind the primary barrier is 15.8 m, and the calculated barrier thickness needed to protect the occupant in this area is 329.43 mm of concrete. The installed barrier thickness for this purpose was measured to be 500.00 mm, an indication of over-shielding. This consideration might be given to make provision for increase in workload and or for higher energy source. And also for the purpose of radiation protection, it is more desirable to over-shield a radiation source than to under shield it. This gives an added protection against the source.

4.2.0 Secondary barrier thickness for the Co-60 scanner

The radiation transmitted by the secondary barrier (leakage and scatter) would affect people located at the following places, Manager's Office, Secretary’s Office, Washroom, Control Room, Check In Office, Reception, Waiting Area, Marshall Post and the corridor.

4.2. 1. Leakage Barrier Thickness for the Co-60 Gamma-ray Scanner

Table 4. 2: Leakage barrier thickness compared with designed specification

POI	*P (Sv/hr)*	*d/m*	*T*	B_l	*nTVLs*	*Calculated Thickness (mm)*	*Designed thickness (mm)*
Manager's Office	1.00E-04	8.27	1.00	6.33E-03	2.20	479.25	430
Secretary's Office	1.00E-04	8.60	1.00	6.85E-03	2.16	471.85	430
Washroom	1.00E-04	8.00	1.00	5.93E-03	2.23	485.54	430
Control Room	1.00E-04	12.50	1.00	1.45E-02	1.84	401.03	430
Check In Office	1.00E-04	13.20	1.00	1.61E-02	1.79	390.72	430
Reception	2.00E-05	14.69	1.00	4.00E-03	2.40	398.94	430
Waiting Area	2.00E-05	16.75	0.25	2.08E-02	1.68	366.74	430
Marshall Post	1.00E-04	11.85	1.00	1.30E-02	1.89	411.15	430

The average secondary shielding employed by the structural shielding engineers against leakage gamma radiation was 430 mm of concrete. The result for the calculated leakage barrier thickness for the gamma-ray scanner based on the current operational data is tabulated in Table 4.2.

From the table, the Manager's Office requires a concrete shield thickness of 479.25 mm against the leakage gamma radiation. This is against the 430 mm of concrete shield installed by the shielding engineers. There is therefore a shielding deficit of 49.25 mm of concrete. In order for the occupant of the office to receive adequate protection from leakage radiation, 49.25 mm of concrete needs to be added to the current barrier thickness.

The Secretary's office requires a shielding of thickness 471.85 mm for him to be adequately protected against the leakage radiation. Comparing this value with the installed shield shows a deficit of 41.85 mm, which must be added to the current thickness in order that the secretary is well protected.

To effectively shield the washroom from any leakage radiation from the scanner shielding head requires a wall thickness of was 485.54 mm. Comparing this thickness with the concrete shield employed during the construction of the facility shows a shielding difference of 55.54 mm which must be added to 430 mm employed during the construction of the facility.

The control room is at a distance of 12.50 m from the gamma ray scanner source axis and would require a leakage shielding thickness of 401.03 mm. Comparing this with shielding employed the expert during the construction of the facility shows an over shielding thus giving the occupants of the control room an added protection.

The calculated thickness expected to adequately shield the check-in office against the leakage gamma radiation is 390.72 mm. There is an excess shielding of 39.28 mm, an indication that the occupants of check-in office are well protected against leakage radiation from the source.

The reception is at distance of 14.69 m from the gamma ray source-axis requires a concrete shielding barrier thickness of 398.94 mm. The installed shielding thickness is 430 mm, showing an extra concrete shield thickness of 31.06 mm.

The corridor which is 12.70 mm from the gamma ray source-axis requires a concrete shield of 419.15 mm. Comparing this with the designed thickness shows that occupant of this area are adequately protected from the leakage radiation. This reason for this might be due to the fact that the designed shield needs to protect both far and near occupants.

According to the leakage shielding calculations against the gamma ray, the marshal post requires an effective concrete thickness of 411.15 mm. Comparing this with that employed by the shielding experts shows an excess shielding of 18.85 mm concrete. This means that for a worker occupying the office is adequately protected from the leakage radiation from the source a shielding, therefore does not require any additional concrete to be added to the current shield.

The waiting area is a public access area where the customers or members of the public who come to the facility sit. A member of the public located at this area who is at a distance of 16.75 m requires a concrete shielding thickness of 366.74 mm in order to receive adequate protection against the leakage radiation. The average secondary barrier concrete thickness employed by the structural shielding engineers was 430 mm.

The average concrete shielding thickness required to adequately protect all the various points of interest likely to be affected by the leakage radiation based on the current operational data was calculated to be 428.70 mm. This value is approximately equal to the secondary leakage barrier shield thickness of 430 mm installed during the construction of the facility. Therefore, the shielding at its current state provides the needed protection against leakage radiation from the source head.

4.3 Scattered Barrier Thickness for the Co-60 Gamma-ray Scanner

Table 4. 3: Scatter Barrier thickness compared with designed specifications.

POI	P (Sv/hr)	d/m	T	Bs	nTVLs	Thickness Calculated (mm)	Thickness Designed (mm)
Manager's Office	1.00E-04	8.27	1.00	6.96E-03	2.16	325.78	430
Secretary's Office	1.00E-04	8.60	1.00	7.53E-03	2.12	320.64	430
Washroom	1.00E-04	8.00	1.00	6.51E-03	2.19	330.13	430
Control Room	1.00E-04	12.50	1.00	1.59E-02	1.80	271.60	430
Check In Office	1.00E-04	13.20	1.00	1.77E-02	1.75	264.45	430
Reception	2.00E-05	14.69	1.00	4.39E-03	2.36	355.97	430
Waiting Area	2.00E-05	16.75	0.25	2.28E-02	1.64	247.84	430
Marshall Post	1.00E-04	11.85	1.00	1.43E-02	1.85	278.60	430
Corridor	2.00E-05	12.70	0.25	1.31E-02	1.88	284.15	430

In order that the Manager's Office would be adequately protected against scattered gamma radiation from the container, the calculated concrete shielding thickness of 325.78 mm is required. However, the average thickness employed as secondary barrier shielding to contain both leakage and scattered radiation is 430 mm, which shows an extra concrete shield of 104.22 mm. Based on this, the manager can be said to be well protected from the scattered radiation from the container.

The secretary's office needs a concrete shield of 320.64 mm for effective protection against radiation beam from the scatterer (container) which is at a distance of 8.60 m away. Comparing this calculated shield with that installed for this purpose shows an extra shielding thickness of l09.36 mm concrete which gives an added protection to secretary.

From the control room, a concrete of thickness 271.60 mm is require for an effective protection against the gamma ray radiation scattered by the container. Here, there is a difference of extra 158.40 mm compared with the installed shield, indicating an over shielding for a worker occupying the control room to receive the needed protection against scattered radiation. This can be explained by the fact that the shielding barrier was for both the leakage and scatter radiation.

The calculated shield thickness to protect the washroom against the gamma radiation scattered by the container was 330.13 mm. Therefore, any person vising the washroom during beam-on state receives the needed protection against scattered radiation.

From the container to the check-in office is 2.25 m and requires a barrier thickness of 264.45 mm to adequately protect its occupant from the scattered beam from the container. Comparing with the employed concrete of 430 mm shows a difference of 165.55 mm. This is due to the fact that leakage radiation is more penetrating than scattered radiation and that the structural shielding engineers might have considered leakage as more important than the scattered radiation in constituting the average secondary barrier shielding, since this shield is to protect the occupants from both leakage and scattered radiation.

The scattered barrier thickness required to adequately protect the receptionist against the scattered radiation by the container was calculated as 355.97 mm concrete. This shows a shielding difference of 74.03 mm compared with that of 430 mm employed by the shielding engineers, giving the receptionist an added protection against the scattered radiation.

The waiting room was located at a distance of 16.75 m from the scatterer. The public in this area requires a concrete shielding thickness of 330.13 mm for them to receive adequate protection from the gamma radiation scattered by the container. Therefore additional shield of 99.87 mm was constituted when the shielding experts constructed the facility. The 330.13 mm is purposely meant to contain only the scattered radiation but in installing the secondary barrier shielding, consideration was given to leakage as well as scattered radiation and this perhaps explains why an average secondary barrier shielding of 430 mm was put in place to cater for both radiations since leakage is more important than scattered radiation and a thicker shield must be preferred to a smaller one in such a circumstance.

4.4. Dose rates for the secondary barrier thickness of the gamma ray scanner.

Table 4. 4: Calculated dose rates for the secondary barrier thickness for the gamma ray scanner.

POI	*d/m*	*IDR_s (μSvh^{-1})*	*IDR_l (μSvh^{-1})*	*IDR_T (μSvh^{-1})*	*$TADR_h$ (μSvh^{-1})*
Manager's Office	8.27	5.44E-05	4.48E-04	5.03E-04	2.62E-06
Secretary's Office	8.60	5.03E-05	4.15E-04	4.65E-04	2.42E-06
Washroom	8.00	5.82E-5	4.79E-04	5.37E-04	2.80E-06
Control Room	12.50	2.38E-05	1.96E-04	2.20E-04	1.15E-06
Check In Office	13.20	2.14E-05	1.76E-04	1.97E-04	1.03E-06
Reception	14.69	1.72E-05	1.42E-04	1.59E-04	8.30E-07
Waiting Area	16.75	1.33E-05	1.09E-04	1.23E-04	6.38E-07
Marshall Post	11.85	2.65E-05	2.18E-04	2.45E-04	1.28E-06
Corridor	12.70	2.31E-05	1.90E-04	2.13E-04	1.11E-06

Based on the concrete shielding design thickness of 430 mm, the calculated dose rate at the manager's office was 2.62 μSvh^{-1}. When compared with the recommended limit of 20 μSvh^{-1} as specified in NCRP (2005) shows that the secondary shielding is adequate to prevent the manager from unnecessary dose from the radiation beam.

From the calculated dose rate for the position of the secretary's office, it can be stated that the secretary received far less dose from leakage and scattered beam than the stipulated minimum dose.

The dose rate outside the secondary shielding to the control room based on the 430 mm concrete shield thickness of the secondary barrier was calculated to be 1.15 μSvh^{-1}, which falls below the 20 μSvh^{-1} as specified by NCRP (2005).

This means that the occupant of this room does not need any additional shielding to be well protected.

Based on the concrete shielding design thickness of 430 mm, the calculated dose rate at the reception was 0.83μSvh^{-1}. When compared with the recommended limit of 20 μSvh^{-1} as specified in NCRP (2005) shows that the secondary shielding is adequate to prevent the occupants of the reception from unnecessary dose from the radiation beam.

The waiting area receives a dose of 0.63 μSvh^{-1}.

This is below the maximum recommended limit of 20 μSvh^{-1} as specified in NCRP (2005) shows that the secondary shielding is adequate to prevent the persons found at this area during the beam-on state from unnecessary dose from the radiation beam.

4.5. Comparison between measured and calculated dose rate for the gamma-ray scanner.

Table 4. 5: Comparison between measured and calculated dose rates for the gamma-ray scanner.

POI	*d/m*	*Calculated*	*Measured*	*Deviation*
Manager's Office	8.27	2.62	1.77	0.85
Secretary's Office	8.60	2.42	1.34	1.08
Washroom	8.00	2.80	2.48	0.32
Control Room	12.50	1.15	1.00	0.15
Check In Office	13.20	1.03	0.93	0.10
Reception	14.69	0.83	0.72	0.11
Waiting Area	16.75	0.63	0.61	0.02
Marshall Post	11.85	1.28	0.78	0.50
Corridor	12.70	1.11	1.00	0.11

From Table 4.5, it can be seen that there is a small deviation between each pair of values, nonetheless, they both fell below the limit of 20 μSvh^{-1} as specified in NRCP 151, 2005.

The secondary barrier has not deteriorated much enough to pose any radiation hazard to staff and the general public.

The calculated dose rate for the primary barrier was 0.62 μSvh^{-1} whilst that recorded by the dose rate meter was 0.93 $\mu Svh^{-1.}$ These two values are all less than the specified maximum limit of 20 $\mu Svh^{-1,}$ therefore the primary barrier still provides the necessary protection for the public against the gamma ray radiation from the cobalt -60 source.

4.6. Barrier thickness calculation for X-ray of energy 6MeV

Table 4. 6: Calculated primary barrier thickness for X-ray of energy 6 MeV.

POI	*P* (*Sv/hr*)	*d/m*	*U*	*T*	B_e	*nTVLs*	*Calculated thickness (mm)*	*Designed Thickness (mm)*
Behind the primary barrier	2.0E-5	15.8	0.06	0.25	3.08E-2	1.51	421.29	500.00

4.7. Scatter and leakage radiation barrier thickness barrier for 6MeV X-ray source.

Table 4. 7: Calculated scattered radiation barrier thickness barrier for 6MeV X-ray source.

POI	*P (Sv/hr)*	*d/m*	*T*	*Bs*	*nTVLs*	*Thickness* *Calculated (mm)*	Designed *(mm)*
Manager's Office	1.00E-04	8.27	1.00	6.96E-03	2.16	368.92	430
Secretary's Office	1.00E-04	8.60	1.00	7.53E-03	2.12	363.11	430
Washroom	1.00E-04	8.00	1.00	6.51E-03	2.19	373.85	430
Control Room	1.00E-04	12.50	1.00	1.59E-02	1.80	307.57	430
Check In Office	1.00E-04	13.20	1.00	1.77E-02	1.75	299.48	430
Reception	2.00E-05	14.69	1.00	4.39E-03	2.36	403.11	430
Waiting Area	2.00E-05	16.75	0.25	2.28E-02	1.64	280.67	430
Marshall Post	1.00E-04	11.85	1.00	1.43E-02	1.85	315.50	430
Corridor	2.00E-05	12.70	0.25	1.31E-02	1.88	321.78	430

Table 4. 8: Calculated leakage radiation barrier thickness barrier for 6MeV X-ray source.

POI	*P* (*Sv/hr*)	*d/m*	*T*	B_l	*nTVLs*	*Calculated Thickness (mm)*	*Designed Thickness (mm)*
Manager's Office	1.00E-04	8.27	1.00	4.75E-03	2.32	648.22	430
Secretary's Office	1.00E-04	8.60	1.00	5.14E-03	2.29	638.73	430
Washroom	1.00E-04	8.00	1.00	4.44E-03	2.35	656.26	430
Control Room	1.00E-04	12.50	1.00	1.09E-02	1.96	548.11	430
Check In Office	1.00E-04	13.20	1.00	1.21E-02	1.92	534.90	430
Reception	2.00E-05	14.69	1.00	3.00E-03	2.52	704.00	430
Waiting Area	2.00E-05	16.75	0.25	8.57E-04	3.07	855.66	430
Marshall Post	1.00E-04	11.85	1.00	9.75E-03	2.01	561.05	430
Corridor	2.00E-05	12.70	0.25	5.60E-04	3.25	907.25	430

The values in the Tables 4.6, 4.7 and 4.8 are calculated for 6MeV X-ray using the current work load of1080Gy/wk and other current operational data for the selected point of interest around the facility. The results show that even though, the primary shield can provide the necessary protection for the public, the existing wall thickness for the secondary radiation attenuation is not enough to give the needed protection for the main occupant of the facility.

The average secondary barrier required is 672.69 mm (ie the average of the averages of the leakage and scattered values), and the installed barrier thickness is 430 mm. This means that an additional concrete of 242 mm thickness is required to enable the current shield to provide the needed protection against radiation from an X-ray of energy of at the least 6MeV.

Therefore, the shielding at its current state cannot be used house X –ray energy 6MeV.

4.9. Comparison of commissioning shielding requirement with the shielding requirement at the present state.

Table 4. 9: Calculated values leakage barrier thickness for the Gamma ray scanner at commissioning stage.

POI	*P (Sv/hr)*	*d/m*	*T*	B_l	*nTVLs*	*Calculated thickness (mm)*	*Designed thickness (mm)*
Manager's Office	1.00E-04	8.27	1.00	1.90E-02	1.72	375.24	430
Secretary's Office	1.00E-04	8.60	1.00	2.05E-02	1.69	367.83	430
Washroom	1.00E-04	8.00	1.00	1.78E-02	1.75	381.53	430
Control Room	1.00E-04	12.50	1.00	4.34E-02	1.36	297.02	430
Check In Office	1.00E-04	13.20	1.00	4.84E-02	1.32	286.70	430
Reception	2.00E-05	14.69	1.00	1.20E-02	1.92	418.83	430
Waiting Area	2.00E-05	16.75	0.25	6.23E-02	1.21	262.73	430
Marshall Post	1.00E-04	11.85	1.00	3.90E-02	1.41	307.13	430
Corridor	2.00E-05	12.70	0.25	3.58E-02	1.45	315.14	430

Table 4. 10: Calculated values for scattered barrier thickness for the Gamma ray scanner at commissioning stage

POI	*P (Sv/hr)*	*d/m*	*T*	B_s	*nTVLS*	*Calculated Thickness (mm)*	*designed Thickness (mm)*
Manager's Office	1.00E-04	8.27	1.00	2.09E-02	1.68	253.73	430
Secretary's Office	1.00E-04	8.60	1.00	2.26E-02	1.65	248.60	430
Washroom	1.00E-04	8.00	1.00	1.95E-02	1.71	258.08	430
Control Room	1.00E-04	12.50	1.00	4.77E-02	1.32	199.55	430
Check In Office	1.00E-04	13.20	1.00	5.32E-02	1.27	192.40	430
Reception	2.00E-05	14.69	1.00	1.32E-02	1.88	283.92	430
Waiting Area	2.00E-05	16.75	0.25	6.85E-02	1.16	175.80	430
Marshall Post	1.00E-04	11.85	1.00	4.29E-02	1.37	206.55	430
Corridor	2.00E-05	12.70	0.25	3.94E-02	1.40	212.10	430

The facility started operation in 2010 with the average workload of 360Gy/week, ie 20 scans per day.

From the Tables 4.9 and 4.10 the barrier thickness needed to adequately protect the workers and the general public is approximately 290 mm. But the structural engineers have installed a thickness of 430 mm, an excess concrete of 140 mm. It can be seen that there was an over-shielding of the source during its early stage of operation. However, in about six years from commissioning, the workload has increased by almost three-fold, (from 360Gy/wk to 1080Gy/wk). This requires that the half-value layer of the concrete be increased in order to provide maximum protection of people around the facility.

Table 4. 11: Calculated dose rate at commissioning stage.

POI	*d/m*	*IDRs* *µSv/hr*	IDR_l *µSv/hr*	IDR_T *µSv/hr*	$TADR_h$ *µSv/hr*
Manager's Office	8.27	1.81E-05	4.48E-04	4.67E-04	2.43E-06
Secretary's Office	8.60	1.68E-05	4.15E-04	4.31E-04	2.25E-06
Washroom	8.00	1.94E-05	4.79E-04	4.99E-04	2.60E-06
Control Room	12.50	7.94E-06	1.96E-04	2.04E-04	1.06E-06
Check In Office	13.20	7.12E-06	1.76E-04	1.83E-04	9.54E-07
Reception	14.69	5.75E-06	1.42E-04	1.48E-04	7.70E-07
Waiting Area	16.75	4.42E-06	1.09E-04	1.14E-04	5.92E-07
Marshall Post	11.85	8.83E-06	2.18E-04	2.27E-04	1.18E-06
Corridor	12.70	7.69E-06	1.90E-04	1.98E-04	1.03E-06

Table 4. 12: Comparison of calculated dose rate at commissioning stage with the calculated dose rate the current stage.

POI	*Commissioning stage (µSv/hr)*	*Current state (µSv/hr)*
Manager's Office	2.43	2.62
Secretary's Office	2.25	2.42
Washroom	2.60	2.80
Control Room	1.06	1.15
Check In Office	0.95	1.03
Reception	0.77	0.83
Waiting Area	0.59	0.64
Marshall Post	1.18	1.28
Corridor	1.03	1.11

Table 4.12 compares the dose rate at commissioning stage and the current state. Comparing each pair of entries, it can be seen that there has been increment in dose rate over the past six years. This is mainly due to increase in the workload as a result of increase in the average number of scan per day.

However, these increases in dose rates to the individuals still fall below the limit 20µSv/hr. (NCRP 151, 2005), but due to the cumulative nature of radiation doses, it is advisable that management of the facility takes steps to optimise dose to staff as well as the general public by keeping an optimal scan per day.

CHAPTER FIVE

5.1 Conclusions.

The assessment of the shielding integrity was carried out using commissioning data and the current operational data. Dose rate were measured beyond the primary and secondary barriers. Both the average measured and calculated dose rate for primary barrier were below the 0.02 $mSvhr^{-1}$ limit specified by NRCP 151, 2005. This shows that the shielding is adequate for the protection of the workers.

The secondary shielding is also enough for the persons at the premises of the Co-60 scanner, since the measured dose rates were below the specified limit of 0.02 $mSvhr^{-1}$

It was however, observed that due to increase in workload over the past six years, there has been increase in dose to the staff and the general public.

The assessment done on the install shielding using the current operational data indicates that the shielding is inadequate in providing protection for the general public and the workers against X-ray radiation source of energy 6MeV.

5.2 Recommendation

As the result of the research carried on the facility, the following recommendations are made:

To Management of the NICK scan facility,

- Optimise dose by reducing the weekly workload by keeping an optimal number of scans per week, preferably 800Gy/wk.
- Ensure that door to the tunnel is always closed during scanning time.
- Provision of local rules and procedures at the workplace to be followed by all staff and it should be supervised by a qualified Radiation Safety Officer (RSO) at the facility.

To regulatory authority –Nuclear Regulatory Authority, Ghana.

- Encourage all practices no matter the level of exposure to develop an ALARA culture for the facility.
- Ensure regular shielding integrity checks on all practices within the country, at least once in every two years.
- Enforce the need for the facility to conduct random checks in aspects of workplace monitoring.

REFERENCES.

Biggs, P.J. (1995). Obliquity factors for Co-60 and 4, 10 and 18 MV X-rays for concrete, steel and lead and angles of incidence between 0° and 70°.

Bjorkholm, P. (2003). Cargo screening: Selection of modality. *Port Technology*

Blizard EP, Abbott, LS, (1962). Reactor handbook, Volume III, Part B, Shielding. New York: John Wiley and Sons.

Bottan, G. (2009). Principles of cargo screening –II, Cargo Security International.

Brockhoff RC, Shultis JK, Faw RE. (1996). Skyshine line-beam response functions for 20- to 100-MeV photons. Nucl Sci Eng 123:282–288.

Buffer, A. (2004). Contraband detection with fast neutrons. *Radiation Physics*

Cember, H,(200). Introduction to Health Physics, 3rd Edition, McGraw-Hill, New York.

COTECNA, (2008) Publications, Canada.

Elsayed, E., Young, C., Minge, X., Hao, Z., & Yada, Z. (2009). Port of entry inspection: Sensor deployment policy optimization. *IEEE Transactions on Automation Science and Engineering*, *6*(2) 265-276.

Fadlalla M., (2010). The radiation protection code of practice in teletherapy facilities. Sudan Academy of Science (SAS), Atomic Energy Council.

Fano U, Spencer LV, Berger MJ. (1959). Penetration and diffusion of x rays. Encyclopedia of physics. Vol. 38, Part 2. Berlin: Springer-Verlag;

Faw R. E, Shultis J. K.(1993). Absorbed dose buildup factors in air for 10 -100 MeV Photons, Nuclear Science Engineering, 114:76-80.

Firestone R.B, Baglin, C.M., Frank-Chu.(1999). Table of Isotopes, 8th Edition, Wiley, New York.

Firestone RB, Shirley VS. (1996). Table of isotopes. New York: John Wiley and Sons.

G. Emi-Reynolds and A.W.K Kyere, (1999). Regulatory and radiological Assessment of a Co-60 Teletherapy at Korle-Bu Teaching Hospital. Journal of Applied Science and Technology.Vol. 4 Nos. 1 and 2

Goldstein H, Wilkins JE, Jr. (1954). Calculation of the penetration of gamma rays. White Plains, NY: Nuclear Development Associates; NYO-3075.

Goldstein H. (1959). Fundamental aspects of reactor shielding. Reading, MA: Addison-Wesley.

Grande, P. and O. R. M. C., (1971). Data for Protection Against Ionizing Radiation from External Sources: Supplement to ICRP Publication 15, International Commission on Radiological Protection, Pergamon Press,

Gui AA, Shultis JK, Faw RE. (1997). Response functions for neutron skyshine analysis. Nucl Sci Eng 125:111–127.

Harima Y. (1983). An approximation of gamma-ray buildup factors by modified geometrical progression. Nucl Sci Eng 83:299–309.

Hubbell JH, Berger MJ. (1966). Photon attenuation and energy absorption coefficients. Tabulations and discussion. Washington, DC: National Bureau of Standards; Report 8681.

Hubble J.H. (1969). Photon cross sections, attenuation coefficients, and energy absorption coefficients from 10 keV to 100 GeV. Washington, DC: National Bureau of Standards; SRDSNBS 29.

IAEA (1996b). International Basic Safety Standards for Protection Against Ionizing Radiation and for the safety of radiation sources. Safety Series, No. 115, Vienna, Austria.

IAEA (2004). Code of Conduct on the Safety and Security of Radioactive Sources. Vienna

IAEA (2006a). Radiation Oncology Physics: A Handbook for Teachers and Students.

ICRU (1985). Dose and volume specification for reporting intracavitary therapy in gynaecology. Report 38

IAEA (2011). No. SSG-11 Radiation Safety in Industrial Radiography. Vienna, Austria.

IAEA (2014). International Basic Safety Standards,(BSS) General Safety Requirements Part 3. Radiation Protection and Safety of Radiation Sources, International Atomic Energy Agency, Vienna.

Jaeger RH.(1968). Engineering compendium on radiation shielding. Vol I, shielding fundamentals and methods. New York: Springer-Verlag.

Kenneth, J., Shultis, Richard E. Faw, (2000). Radiation Shielding. La Grange Park, Illinois: American, Nuclear Society, Inc.

McGinley PH, Miner MS. (1995). A history of radiation shielding of X-ray therapy rooms. Health Physics 69:759 –765.

Mutscheller A. Physical standards of protection against Roentgen-ray dangers. Roentgenol Radiation Therapy 13:65–70; 1925.

Muckenthaler F.J.,Mynatt F.R and Stevens PN. (1969). Development of two-dimensional discrete ordinates transport theory for radiation shielding. Oak Ridge, TN: Union Carbide Corporation; USAEC Report CTC-INF-52.

NCRP (1976). Structural shielding design and evaluation for medical use of X-rays and gamma-rays of energies up to 10 MeV. Report No. 49

NCRP (1993). National Council on Radiation Protection and Measurements. Limitation of Exposure to Ionizing Radiation, NCRP Report No. 116, Washington, DC.

NCRP (2003). Radiation Protection for Particle Accelerator Facilities, Bethesda, MD

NCRP (2004b). National Council on Radiation Protection and Measurements. Structural Shielding Design for Medical X-Ray Imaging Facilities, Report No. 147, Washington, DC.

NCRP (2005). Structural Shielding Design and Evaluation for Megavoltage X- and Gamma-Ray Radiotherapy Facilities. Report No. 151, Washington, DC.

NCRP, (2004a). Recent applications of the NCRP public dose limit recommendation for ionizing radiation. Statement No. 10, Washington, DC.

Nelson W.R, LaRiviere P.D. (1983). Primary and Leakage Radiation Calculations at 6, 10 and 25 MeV. SLAC-PUB-3039.

P. H. McGinley, (2011). Radiation Shielding Evaluation, Georgia Institute of Technology.

Podgorsak, E.B. (2005). Radiation Oncology Physics: A Handbook for Teachers and Students, International Atomic Energy Agency, Vienna.

Professor P W Horton, (2000). An Introduction to Installation Shielding in Radiotherapy, Royal Surrey County Hospital Guildford, UK

R. D. Mosteller, (2002). Validation Suites for MCNP, DOE, Los Alamos National Laboratory LA-UR-02-0878.

Shultis JK, Faw RE. (2000). Radiation shielding. La Grange Park, IL: American Nuclear Society.

SGS (2011). Société Générale de Surveillance publications, SA

Simmons GL. (1973). An adjoint gamma-ray moments computer code, ADJMOM-1. Washington, DC: National Bureau of Standards; NBS Technical Note 748.

Simpkin DJ. (1989). Shielding requirements for constant-potential diagnostic x-ray beams determined by a Monte Carlo calculation. Health Phys 56:151–154.

Simpkin DJ. (1995). Transmission data for shielding diagnostic x-ray facilities. Health Phys 68:704 –709.

Spencer LV, Chilton AB, Eisenhauer CM. (1980). Structure shielding against fallout gamma rays from nuclear detonations. Washington, DC: National Bureau of Standards, U.S. Government Printing Office; Special Publication 570.

Spencer LV, Fano U. (1951). Penetration and diffusion of x-rays: calculation of spatial distributions by polynomial expansion. J Res Nat Bur Stds 46:446
.

Spencer LV. (1962). Structure shielding against fallout radiation from nuclear weapons. Washington, DC: National Bureau of Standards; Monograph 42.

Stevenson, J (2010). Linac Based Photofission Inspection System Employing Novel Detection Concepts, Nuclear Instruments and Methods in Physics Research.

Suntharalingam, N., Podgorska, E. B., TÄolli, T (2005). Brachytherapy: Physical and Clinical Aspects, International Atomic Energy Agency, Vienna, Austria

Taylor JJ.(1954). Application of gamma-ray build-up data to shield design. Washington DC: Westinghouse Electric Corporation, Atomic Power Division; USAEC Report WAPD-RM-217.

Taylor LS. (1981). The development of radiation protection standards (1925–1940). Health Phys 41:227–232.

V. M. Systems, (2012). Shielding Against Primary and Leakage X-rays, Palo Alto, CA: Varian Associates, Inc.

Wang X, Faw RE. (1995). Transmission of neutrons and secondary gamma rays through concrete slabs. Radiation Protect Dosimetry 60:212–222.

APPENDICES

Appendix A

Table A. 1: Tenth Value Layers (TVL) for Co-60 and some selected Energies for concrete of density 2350kgm^{-3}

Radiation Source/	*Co-60*	*X-ray*		
Energy	1.25 MeV	6MeV	10MeV	15 MeV
Primary gamma ray / x - ray	218	343	389	432
Leakage gamma/ x-ray	218	279	305	330

Appendix B

Table B. 1: First TVL (mm concrete) for Container scatter radiation at some selected scatter angles.

Scattered angle /°	*Co-60*	*X-ray*		
	1.25 MeV	6MeV	10MeV	15 MeV
15	218	343	389	432
30	213	279	305	330
45	197	229	23	237
60	189	205	209	299
90	151	171	173	174
135	128	144	144	145

Appendix C

Table C. 1: Summary of recommended/legal effective dose limits and designed dose limits

Dose limit	IAEA	USA	United Kingdom
Occupational exposure dose limit	20 mSv per year averaged over 5 consecutive years and 50 mSv in any single year	Implied annual limit of 10 mSv, cumulative dose of age × 10 mSv, and 50 mSv in any single year	20 mSv in a year or 100 mSv in 5 consecutive years and 50 mSv in any single year
Design limit for occupational exposure		Fraction of 10 mSv annually.	6 mSv in a year IDR is 7.5 $\mu Sv \cdot h^{-1}$
Public dose limit	1 mSv in a year	Infrequently, 5 mSv annually, and continually	1 mSv annually
Design limit for public area		1 mSv annually 20 µSv in any hour	0.3 mSv in a year IDR is <7.5 µSv·h–1 TADR is <0.5 $\mu Sv \cdot h^{-1}$ TADR2000 <0.15 $\mu Sv \cdot h^{-1}$

Appendix D

Table D. 1: Scatter fraction of dose α, at 1 m, for 400 cm^2 incidence beam.

Angle/^{o}	Co-60	6 MeV		10 MeV		18 MeV		24MeV	
		Max a	α at 1.5cm	Max α	α at 2.5 cm	Max α	αat 2.5 cm	Maxα	α at 2.5 cm
10	1.1E–2	1.68E–2	1.04E–2	1.69E-2	1.66E–2	2.43E-2	1.42 E–2	2.74E–2	1.78E–2
20	8.0E–2	1.15E–2	6.73E –3	1.03E–2	5.79E–3	1.172E-2	5.39E–3	1.27E–2	6.32E–3
30	6.0E–3	5.36E–3	2.77E-3	6.73E–3	3.18E–3	7.13E–3	2.53E–3	7.21E–3	2.74E–3
45	3.7E–3	2.97E–3	1.39 E–3	3.25E–3	1.35E–3	3.05E–3	8.64E-4	3.06E–3	8.30E–4
60	2.2E–3	1.74E–3	8.24E–4	1.84E–3	7.46E–4	1.42E–3	4.24E–4	1.37E–3	3.86E–4
90	**9.1E–4**	**7.27E–4**	**4.26E–4**	**7.14E–4**	**3.81E–4**	**3.75E–4**	**1.89E–4**	**3.53E–4**	**1.74E–4**
135	5.4E–4	4.88E–4	3.00E–4	3.70E–4	3.02E–4	2.59E–4	1.24E–4	2.33E–4	1.20E–4
150	1.5E–4	3.28E-4	3.16E–4	2.87E-4	2.74E–4	2.26E–4	1.20E-4	2.12E–4	1.13E–4

Appendix E

Table E. 1: Occupancy factor for some selected areas

Type of area	NCRP	IPEM
Offices, Reception areas, Shops, Control room Children's play areas, Staff rooms	1	1
Corridors	1/4	0.2
Toilets, bathrooms, Outside areas with seating	1/16	0.1
Stairways, unattended waiting rooms,	1/16	0.05