SHIELDING DESIGN FOR POSITRON EMISSION TOMOGRAPHY FACILITY

BY

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A dissertation submitted in partial fulfilment of the requirements for the degree of

Master of Science in Medicine

in the

Department of Medical Physics
Faculty of Medicine, Medunsa Campus
University of Limpopo

January 2007

ACKNOWLEDGEMENTS

This work was produced with the assistance and guidance of the following people to whom I would like to express my sincere gratitude.

- Mr. O.R. Van Reenen, my supervisor, for his guidance, advice and support throughout this study.
- Dr. A.C. Chamberlain, my co-supervisor, for his support and guidance, which is highly appreciated.
- Dr. G. Kemp of PET Labs Pharmaceuticals (Pty) Ltd, at Little Company of Mary Hospital, who made this investigation possible.
- Mr. J.L Roux of PET Labs Pharmaceuticals (Pty) Ltd, at Little Company of Mary Hospital, for his assistance in making the literature material available.
- Mr. W. A. Elsaigh of Civil Engineering Department at Pretoria University, for his assistance with regard to concrete and bricks characterization.
- My wife Rihab and my son Ahmed who sacrificed the most so that I could peruse my interest.
- My mother and late father who raised me knowing the value of education.
- My creator.

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ABSTRACT

With the recent advent of readily available tracer isotopes, there has been marked increase in the number of hospital-based and free-standing positron emission tomography (PET) clinics. PET facilities employ relatively large activities of high-energy photon emitting isotopes, which can be dangerous to the health of humans and animals. This coupled with the current dose limits for radiation worker and members of the public can result in shielding requirements. This research contributes to the calculation of the appropriate shielding to keep the level of radiation within an acceptable recommended limit.

Two different methods were used including measurements made at selected points of an operating PET facility and computer simulations by using Monte Carlo Transport Code. The measurements mainly concerned the radiation exposure at different points around facility using the survey meter detectors and ThermoLuminescent Dosimeters (TLD). Then the set of manual calculation procedures were used to estimate the shielding requirements for a newly built PEF facility.

The results from the measurements and the computer simulation were compared to the results obtained from the set manual calculation procedure. In general, the estimated weekly dose at the points of interest is lower than the regulatory limits for the Little Company of Mary Hospital. Furthermore, the density and the HVL for normal strength concrete and clay bricks are almost similar.

In conclusion, PET facilities present somewhat different design requirements and are more likely to require additional radiation shielding. Therefore, existing shields at the Little Company of Mary Hospital are in general found to be adequate and satisfactory

and additional shielding was found necessary at the new PET facility in the department of Nuclear Medicine of the Dr. George Mukhari Hospital.

By use of appropriate design, by implying specific shielding requirements and by maintaining good operating practices, radiation doses to staff and the public can be kept to acceptable limits.

Contents

CHAPTER 1

INTRODUCTION

1.1 Introduction

The clinical practice of nuclear medicine involves the administration of trace amounts of compounds labelled with radionuclides that are used to provide diagnostic information for a wide range of diseases. The power of nuclear medicine lies in its ability to provide carefully sensitive measures of a wide range of biologic processes in the body. The nuclear medicine technique involves injecting a compound, which is labelled with a gamma-ray-emitting or positron-emitting radionuclide, into the body. The radiolabelled compound is called a radiopharmaceutical. As a result of the decay of the radionuclide, gamma rays are emitted. The energy of these gamma rays are such that a significant number can exit the body without being scattered or attenuated. An external position-sensitive radiation detector can detect the gamma rays to form an image of the distribution of the radionuclide within the tested organs of the patient [1].

Significant advances continue to occur in the development of both more powerful diagnostic techniques and more effective treatment. There are two broad classes of nuclear medicine imaging techniques. The first technique is called Single Photon Emission Computed Tomography (SPECT). The second technique is Positron Emission Tomography (PET). PET represents a very important nuclear medicine technique that

has improved resolution compared to SPECT. The positron-emitting radiotracers can be labelled to appropriate molecules for study of fundamental biochemical processes [1], to identify tumours in the body.

The use of PET entails handling of radioactive materials and exposure to radiation. Exposure to high level of radiation can be dangerous to the health of humans and animals. For this reason, the department of health requires certain measures to ensure minimum radiation exposure to personnel, the general population and animals. It is therefore necessary to provide the PET facility with an appropriate shielding to keep the level of radiation within acceptable recommended limit. With the growing interest in the use of PET technology for research and clinical diagnosis, it becomes necessary to evaluate the shielding required for such facilities. The focus of this study is to determine the shielding requirements for a new PET facility. The work will include both theoretical calculations and experimental measurements.

1.2 Objectives and scope

In general, this study addresses issues related to the design and implementation of safe operation practices for handling PET isotopes. The main objectives of this study are:

- To determine the shielding barrier requirements for a PET facility using a numerical approach.

- To assess the numerical methods used for the shielding barrier calculation by comparing calculated and measured results. The measurements will be conducted on an operating PET facility.

- To design the shielding barrier requirements for the new PET facility that will be established in an existing building at the MEDUNSA Campus of University of Limpopo.

The barrier determination will include calculations for single-floor facilities as well as multi-story buildings as the later include additional calculations for the upper and the lower floors. It is worth noting that aspects related to cyclotron and their shielding as well as the operation of radiopharmacy is beyond the scope of this study.

1.3 Brief description of the work

The theoretical calculation for the shielding design of the PET facility will be conducted using both numerical methods as well as computer simulations. Monte Carlo N-Particle (MCNP) a general-purpose simulation programme will be used. The recommendations and guidelines provided by the National Council for Radiation Protection are implemented in these calculations [2]. The results of the numerical methods and the computer simulation will be validated by comparing to experimental results obtained on an operating PET facility in the Little Company of Mary Hospital. The measurements will mainly concern the radiation exposure at different points around the facility using survey meter detectors and ThermoLuminescent dosimeters (TLD). The validated approach will thereafter be used to design the shielding requirements for a new facility in the department of Nuclear Medicine of the Dr. George Mukhari Hospital located at the MEDUNSA Campus of University of Limpopo.

CHAPTER 2

POSITRON EMISSION TOMOGRAPHY

2.1 Introduction

Positron Emission Tomography (PET), also called PET imaging, or PET scanning, is a nuclear medicine method widely used in diagnosis by producing quantitative physiological imaging. The PET is one of the most sensitive methods to image trace amounts of molecules in vivo. It is a three-dimensional imaging technique, used in man and animal, designed to measure the level of metabolic activity within the organ [3].

In recent years, the use of PET imaging technique has increased due to the worldwide installation of new PET facilities [4]. On the other hand the PET facilities are using the relatively low cost sodium iodide PET tomographs and the capability of the facilities for sharing ^{18}F produced by a single cyclotron between several PET centres [5].

This chapter discusses the physics of PET, positron-emitting radionuclides and radiotracers, PET radiation detectors, and aspects with respect to preparation and injection of the patient for PET scanning.

2.2 PET scanning technique

The first primarily used commercial PET scanner was introduced in 1975. In the 1970's and 1980's PET was mainly used for research. During the initial PET development, the principal clinical investigation involved the heart and brain. In the early 1990's PET expanded into hospitals, diagnostic clinics, mobile systems and physician practices as the medical community began to realise the benefits of using PET scanning [6]. The current technologies used in the diagnosis, management and treatment of cancer include; Computer Tomography (CT) and Magnetic Resonance Imaging (MRI), PET and single photon emission computed tomography (SPECT). PET enables the physician to obtain non-invasive information that is unavailable from other technologies or procedures. PET as molecular analysis adds information about the chemical activity of normal and abnormal tissue while the other techniques only detect the anatomic defects in the tested organs [1].

PET technique used most often to detect cancer and to examine the effect of cancer therapy by characterising biochemical changes in the cancer. This scan can be performed on the whole body, brain, lung, as well as head and neck. PET as a biologic imaging technique does not replace anatomic imaging, but adds the characterisation of simple molecular processes that are taking place in normal or diseased tissue within the body [1]. The recent interest in PET as a diagnostic imaging technique originates from three factors [3]:

- Appropriate radiotracers,
- Coincidence detection and
- Study reimbursement.

A combination of PET and CT scanners constitute high-end devices, which combine a diagnostic-grade CT scanner with a dedicated PET machine in the same housing. This allows automatic registration of the physiologically determined PET image with

anatomical information in the CT image. The attenuation and scatter corrections are based on an attenuation map generated from the CT image [7].

2.3 Physics of PET

Proton-rich nuclei decay by either electron capture or by positron emission. These alternative decay pathways may exist in the same nuclei and by alternate forms of decay in some atoms, such as in naturally occurring positron emission. In many radionuclides used for PET imaging, most decay pathways are predominantly by positron emission alone [1]. When a positron is expelled from the nucleus of an atom, it travels only a short distance. The positron loses its kinetic energy in collision with atoms of surrounding matter and comes to rest, usually within a few millimetres of the site of its origin in body tissue. The positron then combines with the negative electron in annihilation reaction, in which their masses converted into energy through Einsten's equation $E = mc^2$ [5]. Since the mass of the positron and electron are exactly the same, the annihilation process converts the mass of each particle into pure electromagnetic energy of 511keV from each particle [1]. Therefore a pair of 511keV gamma rays is produced, with gamma rays being emitted in nearly opposite direction, 180 +/- 0.5 degrees apart (refer to Figure 2-1). The minor variation from exactly 180 degrees apart is due to the kinetic energy of the particles at the time when the annihilation event occurs [5]. Each PET radionuclide emits its positron with a different energy, higher energy positrons having greater path lengths and greater mean ranges for the travel of the positron. Table 2.1 list the maximum and the main ranges of some of the most useful PET radionuclides, along with the half-life of each. In those PET radionuclides that have very large mean ranges, the positron may travel a substantial distance (a few millimetres) before the annihilation event occurs. This large particle range results in slight miss positioning of the positron-emitting atom. Therefore a scan performed with ^{18}F (2.4 mm maximum and 0.2 mm average range) will produce a higher resolution image than ^{82}Rb, which has a 15.6 mm maximum and 2.0 mm average range [1].

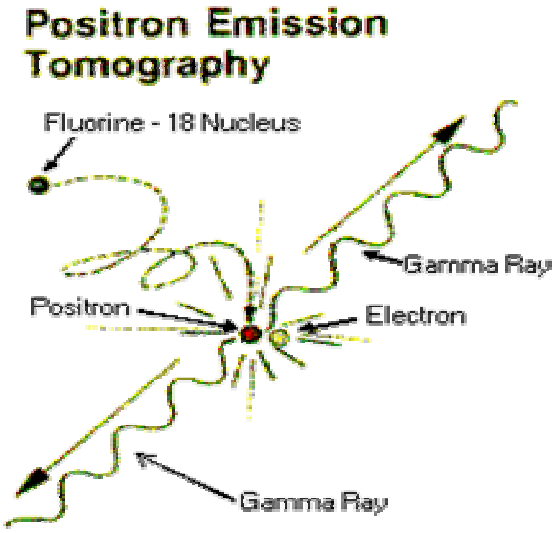

Figure 2.1: positron undergoing an annihilation reaction with electron.

Table 2-1: Positron radionuclides for PET [1].

Radionuclide	Half-life	Maximum range (mm)	Mean range (mm)
^{11}C	20 min	5.0	0.3
^{13}N	9 min	5.4	1.4
^{15}O	2.1 min	8.2	1.5
^{18}F	110 min	2.4	0.2
^{82}Rb	75 sec	15.6	2.6

2.4 Interaction of photons with matter

Photons are far more penetrating in matter than particulate types of radiation. There is no specific range for these photons, and their interaction with matter is based only on a probability of interaction. In matter, photons may undergo scattering, may have no

interaction with matter, or may be completed absorbed and disappear. Photons have four types of interactions as follows [1]:

- Coherent scattering

Coherent scattering is a scattering process that leaves atoms in the same energy state after the scattered photon departs in the direction different from that incident photon. The energy of the scattered photon is the same as that of the incident photon.

- Photoelectric effect

The photoelectric effect is an interaction that takes place between an incident photon and an orbital electron. For the photoelectric effect to occur, the energy of the incident photon must be greater, but in the order of the binding energy of the orbital electron, and the energy of the incident photon must be totally absorbed. With the photoelectric effect some of the photon's energy is used to break the bond of the electron in its orbit and the remaining energy given to the electron in the form of kinetic energy. The probability of photoelectric interaction occurring depends on the energy of the incident gamma ray and the binding energy of an atom's electrons. Therefore a general relationship for this interaction can be written as:

$$E_0 = E_b + E_k \qquad \text{(Equation 2-1)}$$

Where:

E_0 = the incident photon energy.

E_b = the electron binding energy.

E_k = the electron kinetic energy.

- Compton scattering

Compton scattering is an incomplete absorption of gamma ray's energy or scattering of gamma radiation. The Compton effect involves inelastic interactions of photons with orbital electrons. As in the photoelectric effect, there is the emission of an electron that is ejected from the atom; however, not all the incident gamma-ray energy is absorbed,

and is scattered with lower energy. The energy of the scattered photon is always lower than that of the incident photon, as energy depends on the atomic number of the scattering material, the photon incident energy, and the angle of scatter. The energy of scattered photon is related to the angle of deflection. The minimum energy loss of the scattered photons will be at a shallow scatter angle and can be calculated using the following relationship:

$$E_{min} = \frac{E_0}{\left(1 + \frac{2E_0}{0.511}\right)} \qquad \text{(Equation 2-2)}$$

Where:

E_0 = the incident photon energy (in MeV).

E_{min} = the minimum energy loss

The maximum energy will be at a scatter angle of 180 degree and can be calculated from:

$$E_{max} = \frac{E_0^2}{(E_0 + 0.2555)} \qquad \text{(Equation 2-3)}$$

Where: E_{max} = is the maximum scattered energy.

- Pair production

Pair production is an interaction produced when a photon with energy greater than 1.02MeV passes near the high electric field of the nucleus. The strong electrical force brings about the energy-mass conversion. When the photon comes near the nucleus, it disappears totally and two particles of matter are created, an electron and positron, each possessing the mass equivalent of 0.511MeV. For this interaction to occur, the initial photon must possess 1.02MeV or more energy. Any additional energy of the incident photon is converted into kinetic energy, which is given to the positron and electron, thus

conserving energy and momentum. The electron interacts with surrounding atoms, possibly causing ionisation and excitation. The positron loses some of its energy through interactions and ultimately undergoes an annihilation interaction (refer to section 2.3).

2.5 Attenuation and transmission of photons

Interactions of gamma photons with matter combine into the linear attenuation coefficient (μ), which can be defined as the probability of attenuation per distance travelled through the material. Therefore μ has the unit of 1/distance (cm^{-1}). The general attenuation equation is as follow:

$$I = I_0 \, e^{-\mu x} \qquad \text{(Equation 2-4)}$$

Where:

I = Reduced intensity of radiation field.

I_0 = Initial intensity of radiation.

x = Distance travelled by the radiation.

In Equation 2-1 I_0 is reduced by the exponential function of μ and x, to give I. The μ is related to the half-value layer (HVL) of the material as in Equation 2-2. The half-value layer is the thickness of the absorber necessary to reduce the intensity of the radiation to half its initial intensity [1].

$$\mu = \frac{0.693}{HVL} \qquad \text{(Equation 2-5)}$$

2.6 PET radiation detector

A PET scanner consists of circular array of detector elements that are arranged to detect the essentially simultaneous arrival of these two photons. The point of photon emission can then be taken to lie somewhere on the line connecting the two detectors. After enough events are collected, reconstruction by either filtered back projection or algebraic means results in an image of the distribution of the positron-emitting isotope. Detecting 0.511Mev coincidence photons in PET would be most ideal if the scintillation material had a high density to stop the higher energy photons. Ideally, the material should also be very efficient scintillator and create a large amount of light that would be released from the crystal very quickly. Therefore most PET scanners use Bismuth OxyGermanate ($Bi_4Ge_3O_{12}$ or BGO). Bismuth has a high atomic number and is therefore very good at stopping 511keV photons. Some dedicated PET scanner use other scintillation crystal materials. Leutetium Oxyorthosilicate (Lu_2SiO_5[Ce] or LSO) and Gadolinium OxyorthoSilicate (Gd_2SiO_5[Ce] or GSO) are now available in some commercial PET systems [1].

PET imaging requires the placement of detectors on the opposite sides of the patient in order to simultaneously detect the coincidence photons produced by the annihilation process. Dedicated scanners employ specialized detectors and coincidence circuitry that have been optimised for PET scans. Most PET scanners employ a ring of detectors, and then the ring rotated around the subject in order to obtain all of the needed projections [1].

2.5 PET radionuclide and radiotracer

Positrons are produced by proton-rich nuclei. Therefore to manufacture a positron emitting radiopharmaceutical requires a device that can add protons to the nuclei. Small linear accelerators or cyclotrons provide a source of positively charged protons or deuterons of appropriate energy to create these nuclei. Small medical cyclotrons are typically used for producing PET radiopharmaceuticals. The cyclotron must be accompanied by a radiochemistry laboratory that can manufacture the short live

radiotracer of ^{18}F, ^{11}C, ^{13}N, ^{82}Rb, and ^{15}O. ^{18}F is the only radiopharmaceutical that has a half-life long enough (110 minutes) to be transported substantial distance from the cyclotron. Therefore radiotracers ^{11}C, ^{13}N, ^{82}Rb, and ^{15}O must be imaged on a scanner that is in the same facility where the cyclotron is located [1]. Currently, the most common examination in clinical practice is an ^{18}F FDG Scan of the whole body for oncology studies. Fluorodeoxyglucose is a molecule that is an analogue of glucose. Its chemical composition is 2-fluoro-2-deoxy-D-glucose and commonly abbreviated to FDG. Therefore the fluorine in the FDG is replaced with ^{18}F to produce ^{18}F FDG.

All PET isotopes give rise to two 0.511 MeV photons per emitted positron. In addition, there can be additional gamma rays from the nuclear decay and bremsstrahlung radiation emitted as the positron slows down. Bremsstrahlung production is strongly dependent on the atomic number of the medium in which the positron travels and on the positron energy. It will be strongly curtailed in low Z materials such as water and soft tissues [5]. ^{18}F has the lowest positron energy of the common PET isotopes (see Table 2-1) and therefore, under fixed conditions, will have the lowest amount of bremsstrahlung radiation. ^{82}Rb has the highest positron energy (3.15 MeV) and also is the only isotope to have significant contamination from other gamma rays.

2.6 Patient preparation and injection

Before any radiotracer, such as FDG is injected, the oncologists should explain the examination procedure to patient and answer any questions. To reduce uptake in soft tissue, the patient must be kept in a relaxed state before and after the administration of ^{18}F FDG in either bed or chair. After injection, the patient will be held for 30-60 minutes (depending on the type of scan and the practice of the institution), to allow the drug to localize in the lesion of interest. The distribution of FDG in the patient will vary from patient to patient, but most of the isotope will be located in the brain and the bladder. At the end of the uptake period, the patient should void to clear the radioactivity that has accumulated in the bladder [3].

After one-hour post injection, approximately 20 percent of the injected dose accumulates in the patient bladder. After voiding, the patent is positioned on the tomograph for the procedure. Images are acquired at 6 to 10 bed positions over a period of 30 to 60 minutes intervals [3]. The patient may be released following the procedure if

the dose rate at 1m is less than 6μSv/h or may go to a waiting area while the PET study is reviewed

Positron emitting radioisotopes are used to evaluate glucose metabolism and blood perfusion in normal cell function, as well as the other altered metabolism diseases. PET is a minimally invasive diagnostic procedure using a positron camera (tomograph) to measure the decay of radioisotopes such as FDG. The rate of FDG decay provides biochemical information on glucose metabolism in a given cell. This information is then used to diagnosis various diseases. Existing technology, such as computerized tomography (CT) and magnetic resonance imaging (MRI), supplies information about the anatomic structure of suspected malignancies, primarily their size and location. FDG's advantage in cancer imaging is its ability to differentiate lesions based on cellular biochemical or physiologic function. Detecting increased glucose metabolism within cancer cells is unique to PET technology.

CHAPTER 3

SHIELDING DESIGN

3.1 Introduction

Nuclear medicine procedure includes handling radioactive materials and exposure to radiation. To perform this safely, requires precaution to ensure minimum radiation exposure to staff and the general population and only appropriate exposure to patients. In the past, shielding was often unnecessary in nuclear medicine departments performing only diagnostic studies using other techniques. PET facilities differ from traditional nuclear medicine facilities in that the study employs relatively large activities of high-energy photon emitting isotopes and may also use equipment that has CT scanning capability. This coupled with the current dose limits for radiation workers and members of the public create the necessity for shielding considerations [8].

3.2 Facility design

The PET facility will often consist of three rooms. The first is the uptake room, or sometimes named as the patient preparation room, where ^{18}F-FDG is administered to the patient [9]. In relatively busy PET facilities, the preparation room will have to accommodate more than one patient at a time. This can be a crucial aspect to be considered when performing shielding calculations. A toilet needs to be attached to the preparation room, as patients need to urinate and therefore discharge the radioactivity often accumulated into their bladders. The second is the imaging room (scanning room) where the patient will be positioned on the tomograph for the procedure and will remain for about 30 to 60 minutes [3]. A third room is the control room, attached to the PET imaging room, is needed to operate the machine and control the imaging procedure. Because of the high penetration of annihilation radiation of PET radionuclides all surrounding areas in the vicinity of the PET imaging clinic ought to be considered for shielding calculations. This includes the areas above and below the PET clinic as well as adjacent areas on the same floor.

3.3 Radiation protection requirement

The purpose of radiation protection programme is to monitor individual's contact with radiation and to limit their exposure to as low level as possible. The National Council on Radiation Protection (NCRP) established limits of radiation exposure for worker in radiation areas and for non-radiation workers. For a member of the public the dose limit is 1 mSv/year. On a weekly basis, this means controlling the dose to the level of 20 μSv. There is an additional requirement that the dose rate in areas accessible to members of the public should not exceed 20 μSv in any given hour [2]. Institutional staff members whose assigned duties do not involve exposure to radiation sources are considered to be members of the public. Radiation workers are limited to receiving 20 mSv total effective dose equivalent per year.

In addition, there are dose limits to individual organs (500 mSv/ year), extremities and the skin (500 mSv/ year), and the lens of the eye (150 mSv/year). The dose to the foetus

of a radiation worker who declares herself to be pregnant is limited to no more than 1 mSv/year in the course of the pregnancy as a consequence of occupational exposure to the mother. Operationally, this last requirement is usually implemented by a monthly limitation of 0.5mSv to the foetus. In addition to the specific limitations outlined above, each facility has an obligation to conduct operations so as to maintain doses to both radiation workers and to members of the public As Low As Reasonably Achievable (ALARA) [2].

Radiation protection can be achieved by one of three principles:

- Providing sufficient distance between individuals and the source of the radiation.
- Limiting the time of exposure to the radiation source.
- Interposing a protective barrier between the individuals and the radiation source.

The third principle is often termed as radiation shielding and is seen to be the most convenient and effective in facilities where a shorter exposure time to radiation or / and larger distance are not possible.

3.4 Shielding materials

Most shielding materials can be used for radiation shielding if the provided thickness is sufficient to attenuate the radiation to an acceptable limit. The most commonly used materials for shielding applications are lead and concrete. The choice between lead and concrete as shielding material is usually made for the reason of economy and depends upon the energy of the radiation to be attenuated.

Lead is available in the form of leaded wallboard and as plate and sheet stocks for special construction. Under narrow beam conditions, lead has a mass attenuation coefficient of 0.153 cm^2/g and a half value layer of 0.398 cm and concrete has a mass attenuation coefficient of 0.0877 cm^2/g and a half value layer of 3.4-4.3 cm at 511 MeV. The effective half-value layer of lead for shielding photons of this energy under broad beam conditions has been reported in the range of 0.41 to 0.55 cm [9], [2] and [10], and 3.4 cm for normal strength concrete [6] and [3].

Calculations based on the above-mentioned values will not provide sufficient shielding since they neglect scatter buildup factors. In addition, even Tenth-Value Layers (TVL's) that are derived from broad beam measurements, such as those provided by the National Council on Radiation Protection and Measurements [2] may not correctly estimate shielding requirements. More precise calculations can be made either by using build-up factors or transport codes. In a build-up model, the transmitted radiation intensity through a barrier can be calculated from Equation 2.4 (refer to section 2.5). The appropriate build-up function for the shield geometry should be used. The build-up in concrete is more significant than that in lead. Courtney [9] evaluated gypsum wallboard as a shielding material at 0.511 MeV, but found it to have a transmission factor of 91% even at a thickness of 11.4 cm. Thus, gypsum wallboards normally make a negligible contribution to shielding in the facility.

The standard reference for practical shielding design is the NCRP Report No. 49 [2]. This standard does not provide data for 0.511 MeV photons. However the AAPM Task Group [3] provided values of broad beam transmission factors for lead, concrete, and iron that are based on consistent Monte Carlo calculations. They found an insignificant difference between the TVL and Monte Carlo results for lead up to a 1.0 cm thickness. With increasing the thickness of lead beyond the point of 1.0 cm, the TVL actually overestimates the amount of lead required as compared to the Monte Carlo calculation. Iron had a similar result while a substantial difference between the TVL and Monte Carlo results for concrete is recorded. Table 3.1 Summarises the Monte Carlo transmission factors for lead and concrete.

Table 3.1: Broad beam transmission factors at 0.511 MeV in lead and concrete [3].

Thickness (mm for lead and cm for concrete	Transmission factors	
	Lead	Concrete [a]
0	1.0000	1.0000
1	0.8912	0.9583
2	0.7873	0.9088
3	0.6905	0.8519
4	0.6021	0.7889
5	0.5227	0.7218

6	0.4522	0.6528
7	0.3903	0.5842
8	0.3362	0.5180
9	0.2892	0.4558
10	0.2485	0.3987
12	0.1831	0.3008
14	0.1347	0.2243
16	0.0990	0.1662
18	0.0728	0.1227
20	0.0535	0.0904
25	0.0247	0.0419

[a] Concrete density = 2.35 g/cm^3.

It is worth noting that the shielding of the radiation rooms shall be constructed that the protection is not weakened by joints, openings for ducts, etc. passing through the barriers, or by conduits, serving boxes, etc. embedded in the barriers. Doors (or other mean of access to the room) as well as observation windows also require spatial considerations to ensure adequate protection.

3.5 Factors affecting shielding requirements

There are several obvious factors that affect the amount of shielding required for PET facilities, these factors include:

- The number of patients imaged.
- The amount of radiotracer administered per patient.
- The length of time that the patient remains in the facility.
- The location of the facility.

3.6 Radiation protection in PET facility

Most of the doses received by the technologist occur during the transport and positioning of patient, followed by the preparation and assay of the dose [11]. A number of investigators have determined the whole body effective dose received by technologists per unit injected activity (average value of 0.018 μSv/MBq) and per procedure (average value of 9.3 μSv) [11], [12], [13], and [14]. Also because of the high

dose constant associated with positron-emitting radionuclides, hand doses for individuals drawing up and administering PET radiopharmaceuticals can be relatively high. Tungsten syringe shields can reduce the hand dose by 88 percent, but the additional weight (approximately 0.8 kg) can make injections difficult [3]. Other ways to reduce hand dose are to use automatic administration systems and to divide the injection responsibilities among the staff. The staff should develop procedures to minimise the time spent near the radioactive patient. As much as possible, information collection, explanations, and blood collection or other tests should be performed before radioactivity has been administered. Remote monitoring of the patients using video cameras can also be used to reduce the time technologists and nurses spend in close proximity to the patients. In a busy clinic, a technologist or nurse could spend more than an hour a day within the range of a radioactive patient and thereby accumulate more than 7.5 mSv/ year [3]. The reasonable way to reduce this dose is to have enough staff so that the contact time between radioactive patients and any of staff members can be diluted.

During the patient image acquisition, at least one technologist is located at the PET system console where both the patient and the progress of the imaging study can be monitored. Ideally, the console area should be located more than 2 m away from the scanner to reduce the operator dose below the ALARA levels.

3.7 Factors affecting dose rates from radioactive patients

In determination of the radiation dose from the patient to the surrounding areas, the following aspects must be considered:

- Radiation source

The radiation sources that must be considered in the shielding plan includes the doses prior to injection, calibration sources, the patient after injection and the transmission sources in the scanner as well as the scatter and leakage radiation from the CT scanner if PET/CT is used. Once the patients have been injected, they represent a source of

radiation to the staff and members of the public. The calibration sources, which might be in the order of 110 MBq of ^{68}Ge, are normally kept sealed in their own storage. Thus the patient is the primary source of radiation that needs to be considered.

- Dose rate constant

Shielding requirements for un-injected doses and for the calibration sources can be evaluated using the gamma-ray dose constants (Γ). The high energy (0.511 MeV) of annihilation photons from radionuclides that emit positrons, produce specific gamma ray constants considerably higher than those of conventional isotopes. For instance, the Γ of ^{18}F is 3 and 9 times higher than that of ^{131}I and ^{99m}Tc respectively [12]. The appropriate dose rate constant for ^{18}F for shielding purposes is (0.143 μSv m^2/MBq.h). But the dose rate from the patient is reduced by a significant factor because the body absorbs some of the annihilation radiation. AAPM Group [3] recommended a patient dose rate of (0.092 μSv m^2/MBq.h) immediately after administration. This corresponds to an effective body absorption factor of 0.36 [3]. A number of studies have been made of the magnitude of the exposure from ^{18}F-DG [6], [12], [13] and [15]. The results at 100 cm from the patient range from 0.055 to 0.150 (μSv/hr)/MBq. The variation in reported values for ^{18}F can be attributed to both the measurement method and the different experimental conditions. For example, Methe [16] measured the dose rates from different surfaces of the body.

- Administration activity

The amount of administered activity for ^{18}F-FDG studies depend to some extent on the mass of the patient, the duration of the uptake time, and the acquisition mode [3]. A number of studies have reported the administration activity in the range 370–740 MBq of ^{18}F-FDG [9], [16], and [17]. The amount of radiotracer that can be administered differs with the acquisition mode and limits of the PET tomograph.

- Reduction factor

As a result of the PET tracers having a short half-life, the total radiation dose received over a particular time period is less than the product of the initial dose rate and duration time. Therefore a reduction factor is required to account for difference [6]. In other words, the reduction factor is used to account for the decay during the integration time. For example the reduction factor for ^{18}F equals to 0.91, 0.83 and 0.76 at study duration times of 30, 60 and 90 minutes respectively.

- Regulatory limits.

The regulatory authority in South Africa establishes the dose limits in controlled radiation areas and uncontrolled areas open to the general public. Under these regulations, the facility must be shielded so that the effective dose equivalent in uncontrolled areas does not exceed 1 mSv/year. The 1 mSv/year limit implies a weekly dose limit of 20 μSv, and this limit becomes the determining factor for shielding calculations in uncontrolled areas. The occupational dose limit in controlled areas is 20 mSv/year. Most shielding calculations use a target level of 5 mSv/year in controlled areas to be consistent with ALARA recommendations.

- Occupation factor

Occupational exposure means exposure to the individual to ionising radiation in the course of employment in which the individual's normal duties or authorized activities necessarily involve the likelihood of exposure to ionising radiation [2]. For occupational exposure, the occupation factor T usually assumed to be unity, and ¼ for the corridor (outside a radiation room).

- Use factor

Shielding calculation for the facility is designed to keep the PET scanner running as much as possible, so a use factor of 1.0 is used for the scanning room. All scanners of PET provide significant attenuation of the photons emitted within them, but it is difficult to quantify the total attenuation provided as the patient is travelling through the scanner. No correction is made for the scanner attenuation in calculations [2].

CHAPTER 4

METHODS FOR NUMERICAL ANALYSIS FOR SHIELDING CALCULATIONS

4.1 Introduction

The design of PET shielding can be conducted using both existing numerical formulae and /or computer simulations. In principal, the existing formulae are used to calculate the transmission factor and thereafter the barrier thickness based on the specification of the material used for shielding. For computer simulations, several programmes are available such as MicroShield, or Monte Carlo based programmes. In this research the Monte Carlo N-Particle Transport Code version 4B (MCNP) is used. This chapter includes a numerical method entailing six steps and a brief description of the computer programmes is presented.

4.2 Numerical calculation methods

Shielding design can be performed using hand calculation. In general, this calculation includes the steps from (1) through (6):

(1) Determine the distances on the considered facility. This can be obtained from the building plans of the building.
(2) Determine the expected workloads of the facility in terms of number of patients examined per day and the isotope activity used per patient.
(3) Determine the occupancies of areas within the facility and adjacent uncontrolled areas. This will also include consideration of occupancies above and below the facility in multi-floor buildings.
(4) Determine the location and initial activity of the isotope source and the amount of time the source will be present. This includes the injected patient as a source. For ^{18}F source, the activity must be integrated over the appropriate time periods to obtain the total dose delivered by the source.
(5) Calculate the total dose for controlled and uncontrolled areas.
(6) Calculate the transmission factor for all barriers that will be provided.

In the analysis the doses were estimated for weekly exposure for controlled and uncontrolled areas in the facility. The dose can be estimated using one of the following methods:

- Crude approximation of the patient in air without self-absorption. Accordingly the patient is represented as a point source of ^{18}F-FDG solution.
- Dose rate measurements by assuming a phantom or direct from the patient.

Courtney [9] compared between these two methods by using the MicroShield point kernel photon shielding programme to determine the unshielded exposure rates as function of distance between 5 cm and 1000 cm from two sources. In their comparison, one source was used to simulate a person containing a uniform distribution of 185 MBq (5 mCi) of ^{18}F. The body of the person was modelled as cylinder of water with a radius of 15 cm and height of 66 cm. For the second source, the activity was assumed to be similar to the first source, positioned in a 1.0 cm radius sphere of water. This should provide a conservative estimate of the aliquot between the time it is prepared and the time it is injected. The study revealed that the ratio between the two methods is

approximately 1.4 for distance between 100 cm and 800 cm. In the analysis carried out in this research, the weekly dose at point d meters from the patient source is used to determine the transmission factors for all areas in the facility including upper and lower floors.

4.2.1 Exposure rate due to the point source

The exposure rate ($\dot{R}$) due to the patient source can be estimated using Equation 4-1. The resulting activity is assumed generated from a non-absorptive point source. This assumption is not precisely correct but will result in a conservative estimation of the exposure rate.

$$\dot{R} = \frac{\Gamma A_o}{d^2} \qquad \text{(Equation 4-1)}$$

Where:

Γ = Exposure rate constant.

A_o = Administrated activity (initial activity at time of injection).

d = Distance from the point source.

4.2.2 The exposure per study

The calculation involves the estimation of the exposure rate in the absence of complications caused by the physiological redistribution of the radiotracer. In this calculation, the cumulative activity (A) is used instead of the administration activity (refer to Equation 4-1). The cumulative activity is defined as the total number of disintegrations over the time period in the patient for the study. The exposure per study (R) can be calculated as in Equation 4-2:

$$R = \frac{\Gamma A}{d^2} \quad \text{(Equation 4-2)}$$

4.2.3 The cumulative activity

The relationship between the cumulative activity (A) and the initial administrative activity (A_o) is indicated in Equation 4-3 (Keafott et al., 1992)[7]:

$$A = A_o \frac{1 - e^{-(\lambda t_s)}}{\lambda} \quad \text{(Equation 4-3)}$$

Where:

λ = Decay constant.

t_s = Study duration time (integration time).

4.2.4 The reduction factor

The reduction factor account for decay during the integration time (refers to section 3.7).

$$R_t = 1.443\left(T_{1/2} . t_s \right)\left(1 - e^{-0.693 \left(\frac{t_s}{T_{1/2}}\right)} \right) \quad \text{(Equation 4-4)}$$

Where:

R_t = the reduction factor

$T_{1/2}$ = the radionuclide half-life.

4.2.5 The reduction in the activity after integration time

The activity in the patient is reduced exponentially by physical decay. This is due to the delay required by the uptake phase between the administration of the

radiopharmaceutical and the final distribution. The physical decay factor can be calculated as in Equation 4-5:

$$F = e^{-0.693\left(\frac{t_s}{T_{1/2}}\right)} \qquad \text{(Equation 4-5)}$$

Where F is the physical decay factor. The values for t_s and $T_{1/2}$ ought to have the same units.

4.2.6 Total weekly dose

The total weekly dose ($D(t_s)$) at a point spaced (d) meters from the patient source during the integration time (t_s) can be determined as in Equation 4-6:

$$D(t_s) = \frac{\Gamma\, A_o\, R_t\, t_s\, F\, T\, N}{d^2} \qquad \text{(Equation 4-6)}$$

N = Number of patients per time period.

T = Occupancy factor.

4.2.7 Barrier transmission factor

The barrier transmission factor (B) can be calculated as in Equation 4-7:

$$B = \frac{P}{D(t_s)} \qquad \text{(Equation 4-7)}$$

Where:

P = Target dose in protected area.

4.3 Uptake room calculation

Patients undergoing PET scans need to be kept in a quiet resting state prior to imaging to reduce uptake in the skeletal muscles. This uptake time varies from one clinic to another, but is usually in the range of 30 to 90 minutes. The total weekly dose at different points from the patient during the scanning time can be calculated by using Equation 4-6 while the transmission factor for the barrier thickness can calculated by using Equation 4-7.

4.4 Scanner room calculation

The calculations here use a conservative approach as the shielding of the tomograph itself is neglected while calculating for the scanner room. This will lead to similar results to those obtained for the uptake room. The gantry and detectors of the PET tomograph can provide a substantial reduction of the dose rate at some of the walls. This depends on the actual geometry and placement of the tomograph in the room as well as the type of scanning procedures. If information on the tomograph shielding characteristics is available from the vendor, it can be incorporated into the calculation for the walls that are shielded by the scanner and for the floors and ceilings. Activity that is within the scanner bore is nearly 100% shielded, but the axial width of most PET scanners range from 16 cm to 18 cm. Thus, for a 5-bed position scan, the scanner reduces the dose by 20 percent. However, because of the time required to bring the patient into the room and position them for the scan, the effective reduction is realistically about 15% [3]. It should be noted that this reduction would not be included when dealing with the scanner room. The calculations will however take into account that most patients void prior to imaging and therefore remove an approximately 15 percent of the administered activity. The dose rate will eventually reduce by a factor of 0.85 [3].

4.5 Rooms above and below the PET facility

In multi-floor buildings, it is necessary to include the areas above and below the PET facility as well as those adjacent in the same level for shielding calculations. This is because the 0.511MeV annihilation photons are so penetrating. Figure 4-1 shows the generally accepted source and target distances that apply in the case of multi-floor buildings. It can be assumed that the patient (source of the activity) is 1.0 m above the floor. The dose rate is calculated at 0.5 m above the floor for rooms above and at 1.7 m above the floor for rooms below and 0.3 m for the adjacent room in the same level.

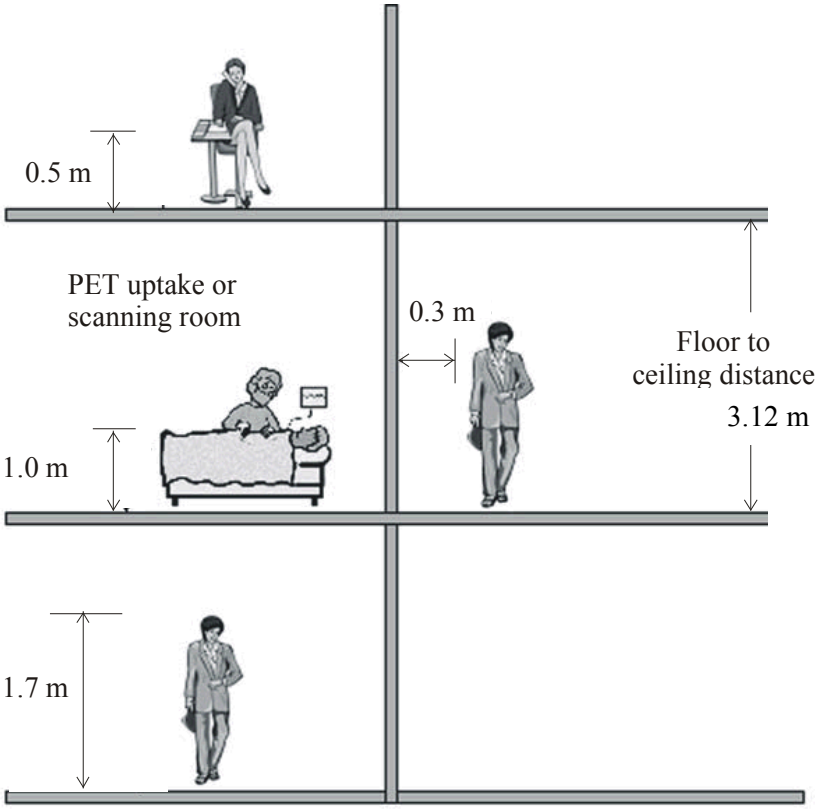

Figure 4-1: Distances to be used for shielding calculations [3].

4.6 Brief description of the Monte Carlo N-Particle transport programme

The Monte Carlo N-Particle (MCNP) transport code is a general-purpose program that can be used for shielding calculations [18]. It is an internationally recognised code for analysing the transport of neutrons and gamma rays and coupled transport. For example, transport of secondary gamma rays resulting from neutron interactions. MCNP can also treat transport of electrons; both primary source electrons and secondarily electrons created in gamma- ray interactions. Specific areas of application include, but are not limited to, radiation protection and dosimetry, radiation shielding, radiography, medical physics, nuclear safety, detector design and analysis, nuclear oil well logging, accelerator target design, fission and fusion reactor design, decontamination and contamination. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first and second degree surfaces and fourth degree elliptical tori [18].

Important standard features that make MCNP very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of variance reduction techniques; a flexible tally structure; and an extensive collection of cross-section data.

4.6.1 Development of Monte Carlo input file

The Monte Carlo method used to simulate the detection process by calculating random particle histories based on the particle cross-section libraries and geometrical information. Each particle history is calculated until the particle ceases to exist. In order to achieve a statistically reasonable and accurate result, the number of particle history simulations has to be fairly large and generally requires a powerful computer. The programme that will be used is version 4C of MCNP [18]. The input parameters mainly consist of:

(1) Geometry part.
(2) Source definition part.
(3) Material part.
(4) Tally part.

4.6.1.1 Geometry part

MCNP treats the problem of geometry primarily in term of regions or volumes bounded by first or second-degree surfaces. Cells are defined by intersections, unions and complements of regions, and contained user defined materials. The union, intersection and complement operations may be thought as logical *or*, *and* and *not* respectively. MCNP uses a three-dimensional (x,y,z) Cartesian coordinate system. All dimensions are in centimetres. All spaces are composed of contiguous volumes or cells. Each cell is bounded by a surface, multiple surfaces, or by infinity. A surface is represented functionally as $f(x, y, z) = 0$. Every surface has a positive side and a negative side. Surfaces are defined by supplying coefficients to the analytic surface equations or using known points on the surface. The code automatically does an extensive internal checking to find possible input errors. In addition, the geometry-plotting capability in the programme helps the user check for geometry errors.

MCNP tracks particles through the geometry, calculates the intersection of a track's trajectory with each bounding surface, and finds the minimum positive distance to an intersection. At the appropriate surface intersection, MCNP finds the correct cell that the particle will enter by checking the sense of the intersection point for each surface listed for the cell. When a complete match is found, MCNP has found the correct cell on the other side and the transport continues [18].

4.6.1.2 Source part

The MCNP generalised user-input source capability allows the user to specify a wide variety of source conditions without making modifications to the code. Independent probability distributions may be specified for the source variables of energy, time, position and direction, and for other variables such as starting cells or surfaces. Information about the geometrical extent of the source can also be input.

4.6.1.3 Material part

Specification of materials filling the various cells in an MCNP calculation involves the following elements:

(a) Unique material number.
(b) Elemental (or isotopic) composition.
(c) Cross-section compositions.

It should be noted that the density is to be specified on the cell card not in the material card.

4.6.1.4 Tally part

The tally cards are used to specify the required output desired from the calculation using the MCNP. There are several tallies in the MNCP. The most frequently used tallies are current at a surface designated as tally F1, average flux at surface designated as tally F2, flux at a point or a ring designated as tally F5, and flux averaged over a cell designated as tally F4. In the work conducted in this research the tally designated as *F8 will be utilised. The function of this tally is to calculate the energy deposition in a pre-specified volume [18].

CHAPTER 5

EXPERIMENTAL PROCEDURE

5.1 Introduction

In this chapter the layout of the PET facility of the Little Company of Mary Hospital and the measurement performed on the facility are presented and discussed. The aim of the measurement is to provide satisfactory experimental data that can be used to verify the numerical methods and computer simulation described in chapter 4.

5.2 Layout of the PET facility

The experiments were conducted in the operating PET facility of the Little Company of Mary Hospital in Pretoria South Africa. The facility is established on the first floor in the hospital main building. The ground floor of this building accommodates the Cyclotron where the ^{18}F is produced. The ^{18}F is safely transported to the first floor as a unit dose to be injected to the patient. The measurements do not include the Cyclotron facility. Figure 5-1 shows the plan view of the rooms forming the facility.

Several points of interest were chosen in the floor accommodating the facility and in the floor below. The points were positioned in the most critical points around the radiation source and in the most occupied areas in the facility (refer to Figure 5-1). The locations of these points were marked properly in the walls. The Points were positioned 1.0 m above the floor level to yield the most critical readings as the radiation source lies in the same level. TLD's were used to measure the accumulated dose and occupational exposure and were placed on the marks on the walls. The survey meter was used to measure the dose rate and placed at 0.3 m away from the wall at the same height of the

marked point. It is worth noting that several points were marked in the ground floor. However, the preliminary readings in these points (points in the ground floor) showed that the radiation dose is negligible and therefore are not considered for shielding.

At present the facility has a single uptake room. A future extension is suggested to add an additional uptake room by dividing the existing room. This will be achieved by using a barrier wall as in Figure 5-1. This implies that the shielding calculation will be considered for three radioactive sources instead of the two existing sources.

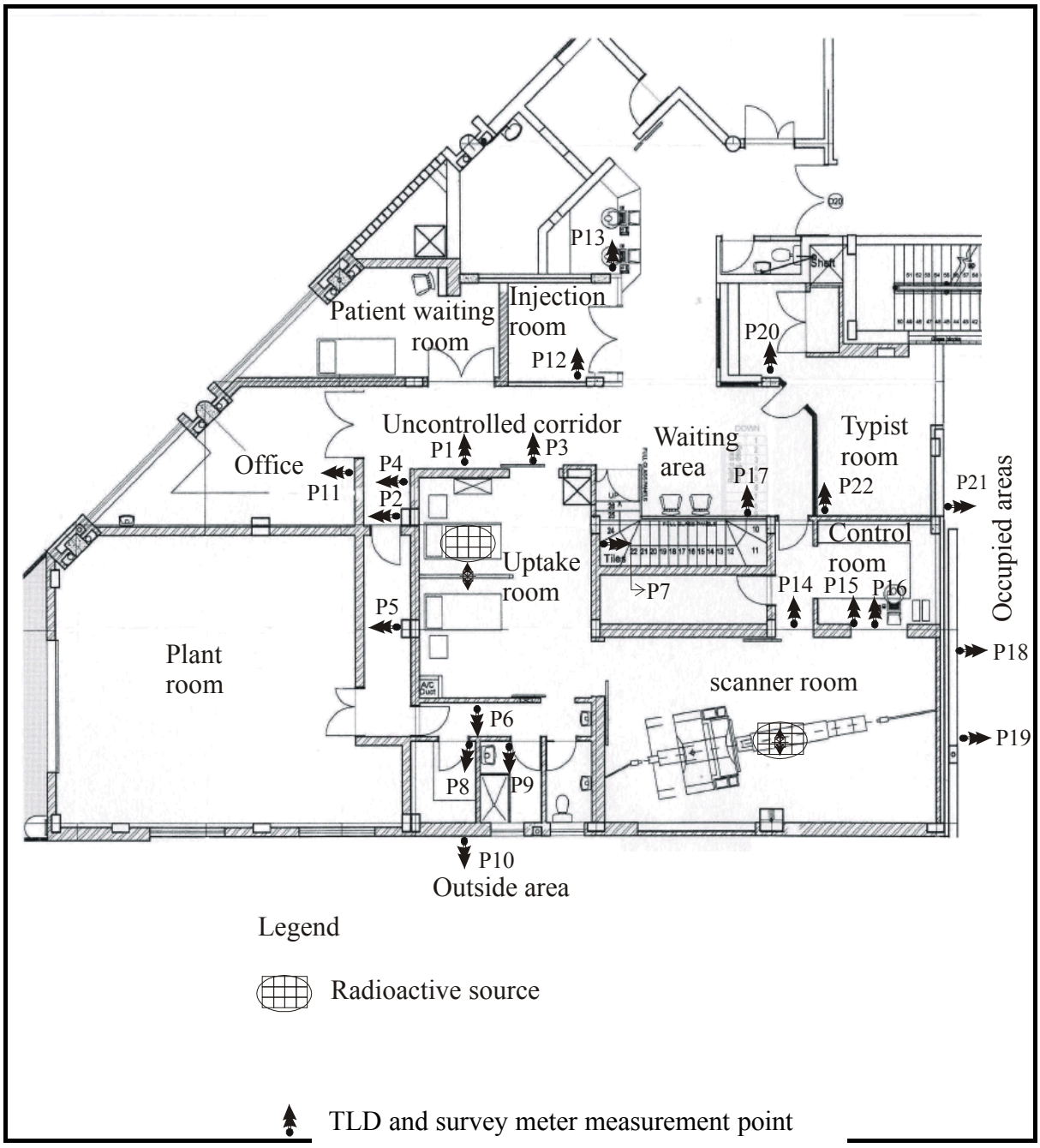

Figure 5-1: Plan view and points of interest of the PET facility and surrounding areas in the Little Company of Mary hospital.

5.3 Experimental method and measurements

The experiments were performed using a radioactive source ^{18}F-FDG contained in a glass vial measuring 5 ml. The initial activity of this source is 370 MBq (10mCi). The first radiation source was placed on the patient bed in the uptake room for 60 minutes, thereafter taken to the scanner room couch for 30 minutes. In the mean time a second radiation source of initial activity 370 MBq (10mCi) was placed on the patient bed, thus having two radioactive sources simultaneously in the uptake room and in the scanner room. It should be noted that the radiation source at the scanner room is less radioactive compared to the source in the uptake room. This will reasonably simulate reality. Several measurements were performed immediately after placing the second radiation source in the uptake room. The measurements approximately take 10 minutes. The experiment was performed 15 times to allow for readings of all points of interest.

The distances of the points of interest were determined using the building plans and confirmed using a measuring tape. Table 5-1 shows the exact distances of the points of interest from the radiation source.

The dose rate and the accumulated dose were measured using the radiation survey monitor and TLD's respectively. The survey meter device used is an Eberline Model ASP2 ion chamber calibrated using ^{68}Ga source. The survey meters read within accuracy of 1 nSv. The TLD chips used are TLD 100. The TLD's were calibrated using ^{90}Sr source. Two sets of TLD's were used. The TLD's of the first set were permanently fixed in all the points of interest from P1 to P22 for a period of a month to measure the occupational exposure. The second set of TLD's was used frequently to measure the dose at selected points. These measurements were necessary to determine the weekly exposure rate so the barrier thickness can be calculated. Preliminary calculations for the weekly exposure dose assisted in eliminating some of the points that were initially designated as points of interest. This reduced the amount of measurements required during the available time of the radiation source. Accordingly, the points P1, P2, P3, P7, P11, P12, P15, P16 and P17 were found to be sufficient for dose and the accumulated dose rate as they indicate reliability dose rats according to acceptable measuring in biological trends.

Seven samples of the bricks used to build the facility were obtained to determine its density. The density is estimated by dividing the weight of the brick sample by its volume. The average density was then estimated using the results from the seven samples. A test was conducted to evaluate the Half Value Layer (HVL) for the bricks. In the test a source and a detector were spaced 1.0 m apart. The level of radiation exposure

for different brick-wall thickness was recorded. The measurements raw data is presented in the Appendix B. The determined HVL will be used to estimate the barrier requirements for the PET facility.

Table 5-1: distances of interest points from the source and the occupancy factor.

Points	*Description*	*Uptake room Distance (m)*	*Scanner room Distance (m)*	*T*	*Points*	*Description*	*Uptake room Distance (m)*	*Scanner room Distance (m)*	*T*
P1	Uncontrolled corridor	2.4	10.8	0.25	P12	T-99m injection room	5.7	12.6	1
P2	Uncontrolled corridor	1.9	11.3	0.25	P13	Reception	8.2	12.6	1
P3	Uncontrolled corridor	2.8	9.8	0.25	P14	Control corridor	6.1	2.6	0.25
P4	Uncontrolled corridor	2.2	11.7	0.25	P15	Control room	10.6	4	1
P5	Controlled corridor	3.2	10.5	0.25	P16	Control room	11.1	4.2	1
P6	Control corridor	4.5	8.3	0.25	P17	Waiting area	7	5.8	1
P7	Controlled steers	4	6.4	0.25	P18	MRI scanner	13.8	5.3	1
P8	Controlled storage	5.6	8.2	0.25	P19	MRI control room	15.2	5.8	1
P9	Shower	6.2	5.6	4.2	P20	Typist room	11	6.3	1
P10	Outside	8	8.8	5.5	P21	Nurse's rest room	13.8	7.8	1
P11	Office	3.8	10.8	5.6	P22	Typist room	9.5	9.5	1

CHAPTER 6

RESULTS AND DISCUSSION

6.1 Introduction

In the previous two chapters existing numerical analysis and experimental procedures to determine the shielding requirements were presented. In this chapter the analytical results from both the numerical methods and computer simulations as well as the experimental results are presented and discussed. The analytical results are compared to the measured results obtained on an operating Positron Emission Tomography (PET) facility. This will further allow the evaluation of the numerical methods.

6.2 Experimental results

6.2.1 General

The building plan and visual investigation of the facility revealed that the walls were built using Clay bricks and mortar. The thickness of all brick walls is approximately 23 cm except for the wall designated Wu1 the thickness is 11.5 (see Figure 6-1). The brick walls were plastered using cement and sand mortar of a thickness varying between 1.0 and 2.5 cm. The floor and the roof were constructed using normal strength concrete having a thickness of 28.5 cm.

6.2.2 Dose rate

Based on preliminary results obtained using numerical calculations, only 9 points were selected for measuring the dose rate (refer section 5.3). These points will be named as points of interest. Table 6-1 indicates the points of interest and measured hourly dose. The weekly dose is calculated thereafter. The reported hourly doses are corrected taking the environmental background reading into account. The average measured background is 0.2 ± (0.07) μSv/h including the readings inside and outside the facility. The value between the brackets represents the standard deviation.

Table 6-1: Measured dose rates for the PET facility.

Interest Point	Measured Hourly dose (μSv/h)	Estimated Weekly dose (μSv/week)
P1	1.26 ± (0.34)	11
P2	0.24 ± (0.09)	2
P3	4.47 ± (0.4)	38
P7	0.21 ± (0.08)	2
P11	0.12 ± (0.09)	3.9
P12	0.38 ± (0.11)	1.5
P15	0.21 ± (0.05)	6.26
P16	0.07 ± (0.11)	2.12
P17	0.01 ± (0.01)	0.29

Table 6-1 indicates that the estimated weekly dose at the points of interest is lower than the regulatory limit (20 μSv/week) except for the P3 where the dose (38 μSv) is significantly larger than this limit. The reason is that P3 is positioned behind an unshielded wooden door. The shielding requirement with regard to the area represented by this point (P3) will be discussed latter in section 6.3. It is interesting to mention that the estimated weekly dose for P2 shows lower dose compared to P11 spite the larger distance of P11 from the radioactive source. This can be explained by the fact that P2 is located at an area that has greater shielding (thicker brick walls).

6.2.3 Annual exposure

Table 6-2 indicates the measured annual exposure of 46 patients and sources with an average administrated activity of 370 MBq (10.2 mCi). The occupational exposure per week, per month and per year was estimated using the measured exposure per study. The value between the brackets represents the error in the TLD reading.

Table 6-2 shows the annual exposure for some points in the facility.

Interest Point	Annual dose (μSv)			
	Measured (per study)	Estimated		
		per week	per month	per year
P1	62.0 ± (0.02)	13.9	55.8	556
P2	14.7 ± (0.11)	3.3	13.2	132
P3	211.4 ± (0.01)	47.6	190.3	1904
P7	13.4 ± (0.09)	3	12.1	120
P11	8.50 ± (0.14)	1.9	7.6	304
P12	36.8 ± (0.03)	8.3	33.1	1324
P15	17.6 ± (0.11)	8.8	15.8	632
P16	7.3 ± (0.19)	4	6.6	160
P17	3.7 ± (0.35)	0.8	3.3	32

Table 6-2 indicates that the estimated annual dose at the points of interest are lower than the regulatory limit (1000 μSv/year) except for the P3 where the dose is significantly larger than this limit. The reason is the same as discussed in Table 6-1. The dose for P12 is slightly higher than the regulatory limit. This was expected because this point is positioned inside the ^{99}Tc injection room where higher doses will be received.

Both dose rate and annual dose per study were measured for several points. The weekly dose then was estimated, and compared with the dose limits. It was confirmed that the estimated dose around uptake and scanner rooms in the present condition was sufficiently lower than the dose limit except for uptake room door which need more shielding of 0.46 cm lead.

6.2.4 Density and Half Value layer for clay bricks

Table 6-3 indicates the measured density and HVL for the clay bricks used in the PET facility under consideration. The standard density and HVL of concrete are also presented to compare to the values measured for bricks [9] and [2]. The value between the brackets represents the standard deviation. The measurements raw data are indicated in Appendix B.

Table 6-3: Density and HVL for Clay bricks.

Material property	Clay bricks	Normal strength concrete
Density (g/cm^3)	2.154 ± (0.02)	2.35
Half-Value Layer (mm)	37.1 ± (2.91)	34

Table 6-3 indicates that the density and the HVL for normal strength concrete and clay bricks are almost similar. Therefore both materials are assumed to result in approximately equal shielding thickness. It should be noted that the shielding capability of a material is not only dependant on the density but also on the chemical composition of the material. However, there is a similarity between the chemical composition of concrete and clay bricks as they both mainly consist of SiO_2 [19] as shown in Appendix C.

6.3 Numerical calculation

The calculation mainly entails two phases: In the first phase, calculation is carried out considering a single uptake room (the present case for the existing facility at the Little Company of Mary hospital). In the second phase the uptake room was divided into two rooms to increase the uptake capacity of the facility (see Figure 6-1).

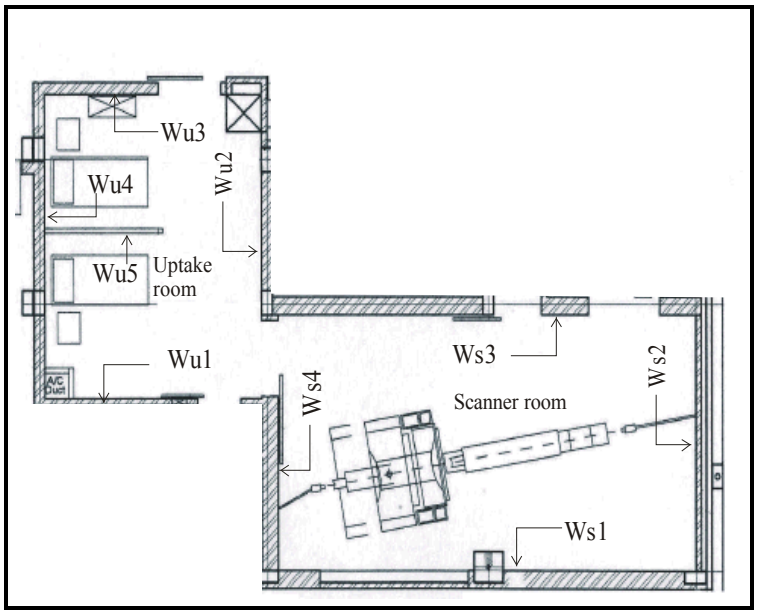

Figure 6-1: Representation of the walls for scanner and uptake rooms.

For each phase, the total weekly dose and transmission factor were calculated and thereafter the required shielding thickness is determined. The shielding thickness was determined for the rooms adjacent, above and below the scanner and uptake rooms. In the calculation, required shielding thickness for concrete was estimated using the values as recommended by the AAPM Task Group (2006) [3] presented in Table 3-1 (refer to section 3.4) and also calculated using the HVL of concrete. The brick thickness is calculated using two methods:

(1) The measured HVL according to Equation 2-1 and Equation 2-2 (refer to section 2.5).

(2) The proportionality between bricks and concrete. The brick thickness is calculated according to Equation 6-1:

$$eq\ x_{bricks} = \frac{eq\ x_{concrete}}{x_{concrete}} . x_{bricks} \qquad \text{Equation 6-1}$$

Where:

$eq\ x_{bricks}$ = Equivalent brick wall thickness.

$eq\ x_{concrete}$ = Equivalent concrete thickness estimated according to the values as recommended by the AAPM Task Group (2006)[3]

x_{bricks} = Brick thickness calculated using the HVL.

$x_{concrete}$ = Concrete thickness calculated using the HVL

In this analysis, the ratio for the concrete thickness (as in column 7 of Table 6-5) was selected to estimate the thickness for brick walls. The resulted brick wall thickness is justified to be viable. This is because of close similarity between the densities of bricks and concrete. In addition, the similarity of the chemical composition of clay bricks and concrete (both mainly contain silica) makes this evaluation even more reasonable.

For all calculations carried out, the following assumptions were made:

(1) No attenuation exists between the source and the point of interest.
(2) The PET facility scans eight patients per day (40 patients per week).
(3) Initial administrated activity is 370 MBq (10 mCi).
(4) The uptake time is 60 minutes.
(5) The scanning time is 30 minutes.

6.3.1 Results of the numerical calculations

6.3.1.1 Considering single uptake room (phase I)

Table 6-4 indicates the calculated total weekly doses for the uptake and scanner rooms. The total weekly dose was calculated for all points of interest as shown in Figure 5-1 (refer to section 5.2). The distances of these points from the radioactive source and the relevant occupancy factors are indicated in Table 5-1 (refer to section 5.3). The weekly doses for the uptake room and for the scanner room were calculated using Equations 4-4, 4-5 and 4-6 (refer to section 4.2.6). The total weekly doses in the last column of Table 6-4 are the arithmetic sums of the weekly doses calculated for the scanner room and the uptake room. The calculations presented in this table are for the PET facility considering a single uptake room (refer to section 6.3). The value between the brackets represents the standard deviation. A Sample of the calculation is presented in Appendix A.

Table 6-4: Total weekly doses for uptake and scanner rooms (Single uptake room).

Interest point	Weekly dose (μSv)		
	Uptake room	Scanner room	Total
P1	49	0.8	50.0 ± (4.1) (*)
P2	78	0.7	79.0 ± (8.9)
P3	36	0.9	37.0 ± (2.6)
P4	58	0.7	59.0 ± (5.3)
P5	28	0.8	28.4 ± (1.7)
P6	14	1.3	15.3 ± (0.59)
P7	18	2.2	19.9 ± (0.82)
P8	9.0	1.3	10.4 ± (0.35)
P9	7.3	2.9	10.2 ± (0.13)
P10	4.4	1.2	5.60 ± (0.08)
P11	78	3.1	81.0 ± 0.48
P12	35	2.3	37.1 ± (1.26)
P13	17	2.3	19.1 ± (0.37)
P14	7.6	13.3	20.9 ± (0.68)
P15	10	22.5	32.6 ± (0.94)
P16	9.2	20.4	29.6 ± (0.81)
P17	23	10.7	33.4 ± (0.29)
P18	5.9	12.8	18.5 ± (0.06)
P19	4.9	10.7	15.6 ± (0.17)
P20	9.3	9.1	18.4 ± (0.12)
P21	5.9	5.9	11.9 ± (0.24)
P22	13	4.0	16.5 ± (0.35)

(*) Refer to Appendix A for sample of calculation

Table 6-4 indicates that the calculated weekly-unshielded doses (considering no walls or any shielding materials between source and detectors) for the points adjacent to the uptake rooms are higher than the points adjacent to the scanner room. This can be attributed to three reasons. The first is due to the reduction in the radioactivity while the patient is waiting at the uptake room after being injected with the radioactive substance (waiting time of 60 minutes). The second is that the waiting time in the scanner room is 30 minutes (half the time spent in the uptake room). The third reason is the larger size of the scanner room, which decreases the dose (the dose at point is inversely proportioned to the square of its distance from radioactive source).

Table 6-5 indicates the thickness for the brick walls estimated using the HVL method and the equivalent thickness methods (refer to section 6.2). At this stage, the calculations were carried out only for selected points that are shown to impact directly on assessing the shielding requirements of the PET facility. The weekly target doses for controlled and uncontrolled areas are determined in accordance to the South African regularly authority (refer to section 3.7). The total weekly doses are calculated as indicated in Table 6-4. The transmission factor is calculated using Equation 4-7 (refer to section 4.2.7) and the required concrete thickness is calculated using the HVL and also interpolated using the values indicated in Table 3-1 (refer to section 3.4). The brick thickness is also calculated using the HVL as well as using the thickness ratio generated for concrete. A sample of the calculation is presented in Appendix A. The calculations presented in this table are for the PET facility considering a single uptake room (refer to section 6.2). In Table 6-5, b indicates that no shielding is required for the particular point. This is because the calculated transmission factor (B) is greater than one (B> 1). The value between the brackets represents the standard deviation.

Table 6-5: Equivalent thickness for the brick walls (single uptake room).

Interest point	Weekly dose (μSv)	Transmission factor	Required concrete thickness (cm)			Required brick thickness (cm)	
			HVL	TASK Group	Thickness ratio	HVL	Estimated TASK Group
P1	50	0.4000	4.5	10 ±(0.6) (*)	2.22 (*)	4.9	11±(0.7) (*)
P2	79	0.2534	6.7	13.2± (0.71)	1.96	7.3	14 ±(1.10)
P3	37	0.5409	3.0	7.7 ± (0.57)	2.54	3.3	8.3±(0.63)
P7	19.9	4.870	b	b	-	b	b
P11	81	0.2458	7	13.4± (0.58)	1.94	8	14 ±(0.63)
P12	37.1	0.5397	3.0	7.7 ± (0.28)	2.53	3.3	8.4±(0.30)
P15	32.6	4.812	b	b	-	b	b
P16	29.6	4.543	b	b	-	b	b
P17	33.8	0.5919	2.6	6.9 ± (0.1)	2.68	2.8	7.5±(0.08)

(*) Refer to Appendix A for sample of calculations.
b indicates that no shielding is required for the particular point

Table 6-5 shows that the points near to the uptake room (P1, P2, P3, P12, and P17) represent transmission factor less than one (B<1), while the rest of the points results in transmission factor more than one (B>1). Therefore, shielding calculations will only be required for the point at which B is less than one (B<1). The calculations show that the existing brick walls in the building under consideration (the facility at the Little Company of Mary Hospital) are satisfactory. The thickness of its brick walls is 23 cm, which is far greater than the required shielding. In other words, the required brick wall thickness ranges between 7.5 and 14 cm, which is significantly less than the provided thickness (23 cm) (see Figure 6-1). The shielding for this facility could also be done using concrete walls of an approximate thickness of 13.2 cm. The economical viability might favour brick for shielding over concrete.

Table 6-6 indicates the estimated required concrete thickness for shielding the rooms above and below the PET facility. The required concrete thickness is determined in a

similar manner as in Table 6-5. This calculation is based on the assumption that the rooms above and below the PET are occupied (occupancy factor of 1). The distances used in these calculations are similar to those presented in Figure 4-1 (refer to section 4.5). In this table, b indicates that no shielding is required for the particular point. This is because the calculated transmission factor is greater than one (B> 1). The value between the brackets represents the standard deviation.

Table 6-6: Concrete thickness for shielding rooms above and below the PET (Single uptake).

Interest position	Distance from source (m)	Weekly doses		Transmission factor	Required concrete thickness(cm)
		Target(μSv)	Total(μSv)		
Above scanner (P24)	2.9	20	42.9	0.4663	8.8 ± (0.05)
Below scanner (P25)	3.47	20	30	0.6676	5.8 ± (0.04)
Above uptake (P26)	2.9	20	134	0.1488	16.7 ± (0.04)
Below Uptake (P27)	3.47	100	93.9	0.2131	b

b indicates that no shielding is required for the particular point

Table 6-6 shows that the required thickness for the roof and the floor slabs of the facility ranges between 5.8 to 17 cm. The existing concrete slab thickness in this facility is approximately 28.5 cm, which is greater than the required shielding. This indicates safe operation with regard to radiation exposure.

6.3.1.2 Considering double uptake rooms (Phase II)

Table 6-7 indicates the calculated total weekly doses for the uptake and scanner rooms considering double uptake rooms. The total weekly doses in the last column of Table 6-7 are the arithmetic sums of the weekly doses calculated for the scanner room and the first and the second uptake rooms. The calculations presented in this table are meant for future extension that may arise due to an increase in the number of patients to be scanned in the PET facility. Considering double uptake rooms for the facility under consideration may constitute the worst-case scenario for the resulting radiation dose. In Table 6-7, b indicates that no shielding is required for the particular point. This is because the calculated transmission factor is greater than one (B> 1). The value between the brackets represents the standard deviation.

Table 6-7: Total weekly doses for uptake and scanner rooms (Double uptake rooms).

Interest Point	Distance from source (m)			Weekly dose (μSv)			
	Uptake room		Scanner room	Uptake room		Scanner room	Total doses
	1	2		1	2		
P1	2.4	4.8	10.8	49	12.3	0.9	62 ± (4.6)
P2	1.9	3.2	11.3	78	27.6	0.7	110 ± (6.6)
P3	2.8	5.2	9.8	36	10.4	0.9	47 ± (2.2)
P7	4	4.7	6.4	18	12.8	2.9	33 ± (1.4)
P11	3.8	5.6	10.8	78	36	3.1	117 ± (5.4)
P12	5.7	7.9	12.6	35	18.1	2.3	55 ± (1.2)
P15	10.6	10.5	4	10	10.3	10.7	31 ± (0.39)
P16	11.1	11	4.2	9.2	9.3	12.8	31 ± (0.34)
P17	7	7.7	5.8	23	19.1	10.7	53 ± (1.2)

Table 6-8 indicates the equivalent thickness for the brick walls considering double uptake rooms. The total weekly doses are calculated as in Table 6-4. The required concrete thickness, equivalent brick wall thickness and the brick thickness using measured HVL are determined in a similar manner as in Table 6-5. In Table 6-8, b indicates that no shielding is required for the particular point. This is because the calculated transmission factor is greater than one (B> 1). The value between the brackets represents the standard deviation.

Table 6-8: Equivalent thickness for the brick walls (Double uptake rooms).

Interest Point	weekly dose (μSv)	Transmission Factor	Required Concrete thickness (cm)			Evaluated brick thickness (cm)	
			HVL	TASK Group	Thickness ratio	HVL	TASK Group
P1	62	0.3221	5.6	11.5 ± (0.52)	2.07	6.1	13 ± (0.57)
P2	110	0.1877	8.2	15.2 ± (0.41)	1.85	9	17 ± (0.44)
P3	47	0.4217	4.2	9.6 ± (0.34)	2.27	4.6	10 ± (0.37)
P7	33	3.0630	b	b	-	b	b
P11	117	0.1704	8.7	15.8 ± (0.31)	1.82	9.5	17 ± (0.33)
P12	55	0.3626	5.0	10.7 ± (0.16)	2.15	5.4	12 ± (0.17)
P15	31	3.2230	b	b	-	b	b
P16	31	3.1890	b	b	-	b	b
P17	53	0.3784	4.8	10.4 ± (0.16)	2.18	5.2	11 ± (0.17)

b indicates that no shielding is required for the particular point

The results in Table 6-4 and Table 6-7 for single uptake and double uptake rooms respectively show that the provision of double uptake rooms increases the radioactive dose at all the points in the facility due to an additional patient. This will result in an additional shielding thickness as indicated in Table 6-8. These calculations reveal that the existing brick wall thickness is also satisfactory if an additional uptake room is provided. The thickness for the concrete roof and floor slabs needs to be checked

Table 6-9 indicates the required thickness for the concrete roof and floor slabs when considering double uptake room. The calculations only included the slabs below (floor) and above (roof) the uptake area and not the rest of the facility. The value between the brackets represents the standard deviation.

Table 6-9: Required thickness for the concrete roof and floor slabs (double uptake rooms).

Interest point	Distance from source (m)		Weekly dose (μSv/week)		Total weekly dose (μSv/week)	Transmission factor	Required concrete thickness (cm)
Above uptake (P26)	2.9	3.83	33.6	19.3	53	0.3784	10.4 ± (0.34)
Below Uptake (P27)	3.47	4.3	93.9	16.1	150	0.6453	6.1 ± (0.03)

The results in Table 6-9 show that the existing thickness for the floor and the roof slabs is adequate.

6.4 MCNP simulation results

A Monte Carlo simulation model, MCNP 4C [18], was further used to calculate the shielding requirement for the PET facility of the Little Company of Mary Hospital. This was done in order to verify the shielding calculation carried out using the numerical procedures as in the previous sections.

In the simulation, the radioactive source (patient) is considered as a sphere phantom 10 cm in diameter, filled with water, containing a positron emitting radionuclide and positioned 100 cm above the floor to match the position of the patient in the uptake room. It was assumed that the radioactive radionuclides distribution was uniform within

this phantom. A TLD detector was set 30 cm from the outer surface of the wall of the uptake room at the same level of the sphere phantom. The computation time is reduced by considering a conical shape with an angle α for the photons beam. The inner and the outer dimensions of the rooms were read off the available design maps of the facility (see Figure 6-2). The chemical composition of the concrete and bricks were obtained from literature references and laboratory tests respectively (refer to Appendix C). The outputs of this simulation are indicated in Appendix D.

The simulation provides results for the energy deposition by using Tally *F8. These results were further used to estimate the weekly doses at critical points (refer to Table 6-10). The conditions used in the calculation are similar to those used in section 6.3.

Table 6-10: shows the energy deposition and the estimated weekly doses

Point	Energy deposition (MeV)	Weekly dose (μSv/week)
P1 (with 0.0cm brick)	3.6015E-09 ± (0.1844)	44.9(*)
P1 (with 23 cm brick)	4.97589E-10 ± (0.1283)	6.2
P26 (with 0.0cm concrete)	1.28628E-10 ± (0.1693)	1.6

(*) Refer to Appendix A for sample of calculations.

The results from this computer simulation are deemed to be reliable. This is because the relative error of the average energy deposition per source particle in the TLD varied from 0.18 to 0.13, which falls within the acceptable error range. The acceptable relative error is between 0.1 and 0.2 [20]. It should be noted that the error was minimised in this simulation by using a relatively larger number of photons (10^8).

6.5 Comparisons between measurements, numerical calculations and computer simulation

Table 6-11 indicates the comparison between the weekly doses from the measurements, numerical calculations and computer simulations for P1. The comparison is carried out considering a single uptake room.

Table 6-11: shows the total weekly doses at point P1 for the three methods.

		MCNP	Measurement	Numerical
Total dose	Shielded (23cm brick)	6.2	11	-
(μSv/week)	Unshielded (0.0cm brick)	44.9	-	49

In general, the proposed calculation using Monte Carlo simulation resulted in a lower weekly doses value compared to values from the numerical calculations and the measurements. Therefore, the numerical methods are deemed to be valid, as it will provide an adequate safe shielding.

CHAPTER 7

SHIELDING REQUIREMENTS FOR A NEWLY PLANNED PET FACILITY

7.1 Introduction

In this chapter, the shielding requirements for a newly planned PET facility are estimated. The PET facility of the Dr. George Mukhari Hospital will be established in the second floor of the clinical Pathology Building on the MEDUNSA campus at University of Limpopo. The experience gained from the experimental and computational work, conducted in previous chapters, is employed to determine the necessary shielding to the existing walls in order to establish the new PET facility.

7.2 Layout of the new PET facility

Figure 7-1 shows the plan view of the units forming the facility. Several points of interest were chosen within the second floor and the floors below and above. These points were selected to represent the most critical points around the radiation source and in the most occupied areas in the facility. Refer to Figure 4-1 for the elevation from the floor and distances from the walls.

7.3 Shielding calculation results

The calculation is carried out considering a single uptake room, the total weekly dose and transmission factor were calculated and thereafter the required shielding thickness is determined. The shielding thickness was determined for the rooms adjacent, above and below the scanner and uptake rooms. In the calculation, required shielding thickness for concrete was estimated using the values as recommended by the AAPM Task Group [3] presented in Table 3-1. The brick thickness is calculated using Equation 6-1 (refer to section 6-3). The conditions used in the calculation are similar to those used in section 6.3.

Table 7-1 indicates the calculated unshielded weekly doses for the uptake and scanner rooms. The weekly doses were calculated for all points of interest as shown in Figure 7-1. The distances of these points from the radioactive source, the relevant occupancy

factors and the source of interest are indicated in this table. The weekly doses for the uptake room and for the scanner room were calculated using Equations 4-4, 4-5 and 4-6 (refer to section 4.2.6).

Table 7-1**:** Weekly doses, location of the interest points, radiation sources, distances to points of interest points from the source and the occupancy factor.

Source of interest	Location of interest point P	Distance to the point P (m)	Occupancy factor	Weekly doses µSv
Scanner room	P1 control room	3.5	1	29.4
Scanner room	P2 controlled corridor	4.4	0.25	4.7
Uptake room	P2 controlled corridor	5.4	0.25	9.7
Uptake room	P3 Office 1	2	1	130
Uptake room	P4 controlled corridor	3.4	0.25	24
Scanner room	P4 controlled corridor	4.6	0.25	4.3
Uptake room	P5 Scanner room	2.5	1	180
Uptake room	P6 Office 2	2.3	1	210
Scanner room	P7 Outside	None	0	None
Scanner room	P8 Outside	None	0	None
Uptake room	P9 Outside	None	0	None
Uptake room	P10 Office 3	2.1	1	260
Uptake room	P11 room below	3.5	1	92
Uptake room	P12 room above	2.9	1	130
Scanner room	P13 room below	3.5	1	29
Scanner room	P14 room above	2.9	1	42.9

Table 7-1 indicates that the calculated weekly-unshielded doses for the points adjacent, above and below to the uptake room are higher than the points adjacent, above and below to the scanner room. This can be attributed to the three reasons discussed in section 6.3.1.1.

Table 7-2 indicates the required thickness for the brick walls around the uptake and scanner rooms (see Figure 7-1) estimated using Equation 6-1, and the required concrete thickness for floor and roof slabs of the uptake and scanner rooms using the values as recommended by the AAPM Task Group [3] presented in Table 3-1.In Table 7-2, b indicates that no shielding is required for the particular point. This is because the calculated transmission factor is greater than one (B>1). The value between the brackets represents the equivalent lead thickness in centimetres.

Table: 7-2: Equivalent thickness for the brick walls and concrete floor and roof slabs.

Walls	Required brick thickness (cm)	Floors and roofs	Required concrete thickness (cm)
A Outside	None	Uptake room floor	14.3
B Outside	None	Uptake room roof	16.7
C Outside	None	Scanner room floor	5.8
D Scanner room	20 (1.50)	Scanner room roof	8.8
E Uptake room door	3.9 (0.16)	-	-
F Office 1 and 2	24 (1.78)	-	-
G Control room	b	-	-
H Scanner room door	b	-	-
I Office 3	23 (1.72)	-	-

b indicates that no shielding is required for the particular point

Table 7-2 indicates the result of the calculation for the shielding requirements of the walls in the facility. The walls around the uptake room (D, F, and I) need additional shielding as the required brick wall thickness is between 20 and 24 cm while the existing wall thickness is 11.5 cm. The remaining walls are found to be adequate and require no additional shielding. The required thickness for the roof and the floor slabs of the facility ranges from 5.6 to 14 cm. The existing concrete slab thickness in this facility

is approximately 20 cm, which is much greater than the required shielding. This indicates safe operation with regard to radiation exposure.

Several scenarios can be set for the additional required shielding for the walls designated D, F and I. One alternative could be thickening the brick wall by attaching another wall or by using an equivalent lead thickness.

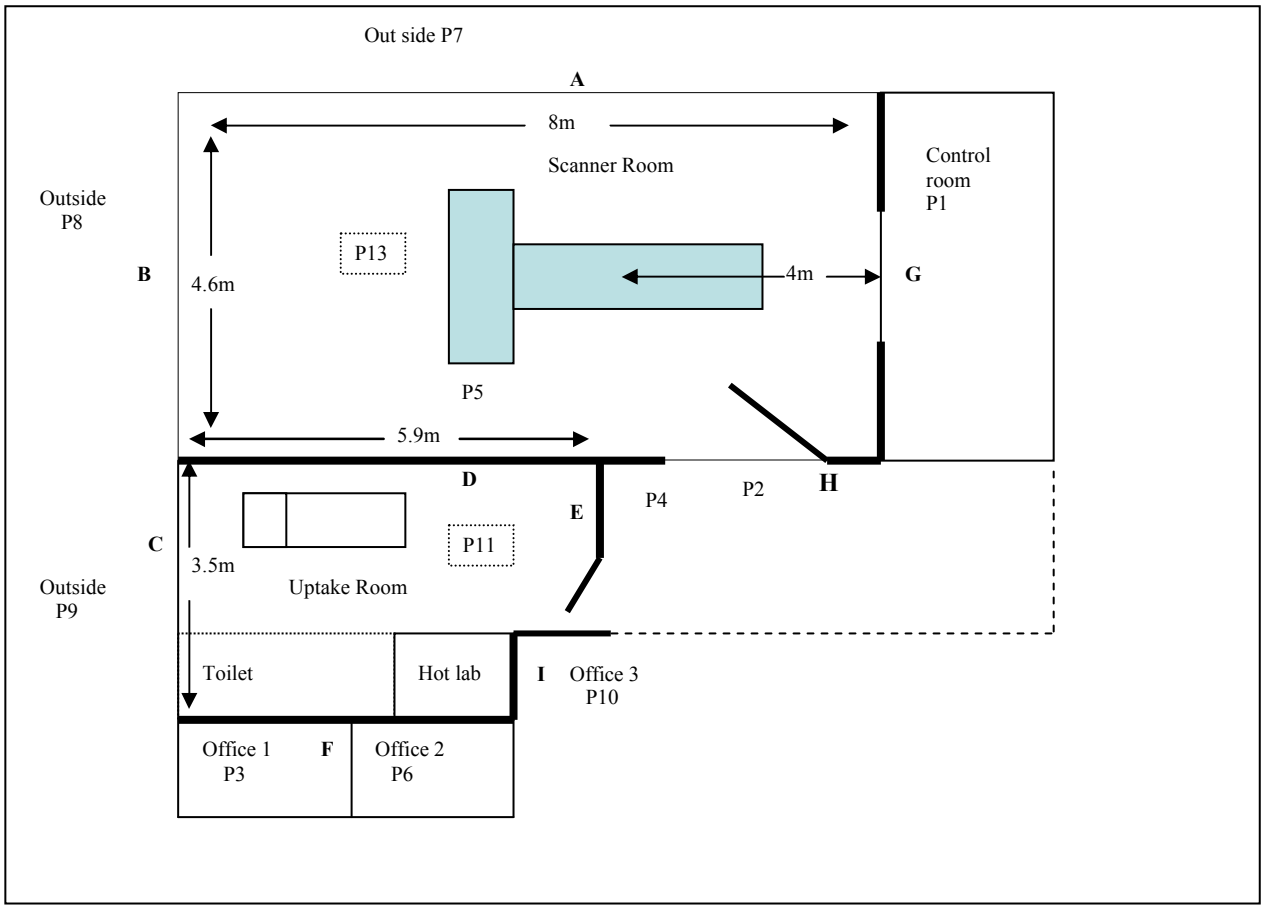

Figure 7-1: Plan view and points of interest of the PET facility and surrounding areas in the PET facility of the Dr. George Mukhari Hospital

CHAPTER 8

CONCLUSIONS AND RECOMMENDATIONS

8.1 Conclusions

The numerical approach using manual calculations was evaluated. Two different methods were used in the evaluation. In the first method, measurements were made at selected points and the results from the measurements were compared to the calculated results. In the second method, a computer simulation involving Monte Carlo Transport Code was used and the results from the simulation and the manual calculations were compared. The numerical manual approach was found valid and satisfactory. The numerical approach was then successfully used to assess the shielding adequacy of an operating PET facility and further used to calculate the shielding requirements for a newly built facility.

The existing shields at the Little Company of Mary Hospital are in general found to be adequate and satisfactory. However, the calculations found that the door of the uptake room is unsatisfactory with regard to shielding requirements. Additional shields were found to be necessary as sheets of lead that are approximately 0.5 cm have to be applied. Future extension in the facility was also found possible by adding a second uptake room by creating a brick wall dividing the existing uptake room into two units. The thickness of the brick wall is found to be approximately 25 cm.

Additional shielding was found necessary at the new PET facility at Dr. George Mukhari Hospital. The additional shield requirements were found for three of the walls of the uptake room. Two shielding scenarios were suggested either by increasing the brick walls thickness to bring the total thickness to approximately 21 cm or by adding lead sheets of a thickness 1.78 cm to the existing brick walls.

8.2 Recommendations

The Half Value layer (HVL) for the bricks was estimated on site by using a radioactive source applied on one side while measuring the radiation does on the opposite side. Further research is needed to evaluate the estimated HVL for clay bricks in proper laboratory environment. In the calculations the mortar (cement-sand paste) between the bricks and the plasters was not included and was assumed to add to the general safety factor for shielding estimation.

The manual calculation method is numerically demanding. Further work is required to create calculation sheets of small software that can be used to minimize the amount of manual calculations. The capabilities of programming software such as Mathcad can be advantageous.

For the new PET facility of Dr. George Mukhari Hospital, measurements of the radiation doses are recommended in regular bases after the facility operation. The measurements should be conducted during a period of approximately 6 months to ensure that the calculated shielding fulfills the requirements of the facility.

CHAPTER 9

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Appendix A

SAMPLE OF CALCULATIONS USING THE NUMERICAL METHODS

The sample of calculations was provided for the interest point (P1). For all calculations carried out, the following assumptions were made:

(6) No attenuation exists between the source and the point of interest.
(7) The PET facility scans eight patients per day (40 patients per week).
(8) Initial administrated activity is 370 MBq (10 mCi).
(9) The uptake time is 60 minutes.
(10) The scanning time is 30 minutes.

A.1 Dose calculation

A.1.1 Uptake room weekly dose

The total dose at a point (P1) 2.4m from the patient during the uptake time is (using equation 4-6)

$$D(t_s) = \frac{0.092\ \mu\text{Sv m}^2/\text{MBq h} \times 370\ (\text{MBq}) \times 1\text{h} \times 1 \times\ 0.83 \times 40 \times 0.25}{(2.4\ \text{m})^2} = 49\mu\text{Sv}$$

A.1.2 Scanner room weekly dose

The total dose at a point (P1) 10.8m from the patient during the scanning time is (using equation 4-6)

$$D(t_s) = \frac{0.092\ \mu\text{Sv m}^2/\text{MBq h} \times 370(\text{MBq}) \times 0.5\text{h} \times 0.91\ \times 0.85 \times 0.685 \times 40 \times 0.25}{(10.8\ \text{m})^2} = 1\ \mu\text{Sv}$$

Where 0.685 is the reduction factor (refer to section 4.4) and 0.85 is the physical decay factor.

A.1.3 Total weekly dose

The total weekly dose is the arithmetic sum of the weekly doses calculated for the uptake room and the scanner room.

$$D = 4.9 + 0.1 = \underline{50\ \mu Sv}$$

A.2 Transmission factor calculation

Transmission factor required is (using equation 4-7).

$$B = \frac{20\ \mu Sv}{50\ \mu Sv} = 0.4014$$

A.3 Required barrier thicknesses

Required barrier thicknesses were calculated for concrete and bricks.

A.3.1 Concrete

A.3.1.1 Using concrete HVL

The concrete thickness was calculated using equation 2.1 and equation 2.2 (refer to section 2.5) and the HVL value from Table 6.3

$$0.4001 = e^{-\frac{0.693}{34(cm)}X(cm)}$$

$$x = \underline{4.49\ cm}$$

A.3.1.2 Using TASK Group transmission factor values

The concrete thickness was estimated using Table 3-1 (refer to section 3.4).

$$x = 10\ cm$$

A.3.1.3 Concrete thickness ratio

The concrete thickness ratio is the ratio between the estimated thickness (using the TASK Group transmission factor value) and the calculated thickness (using the HVL).

$$\frac{\text{eqx}_{\text{concrete}}}{\text{x}_{\text{concrete}}} = \frac{10\,\text{cm}}{4.49\,\text{cm}} = 2.22$$

A.3.2 Brick

A.3.2.1 Using the measured brick HVL

Barrier thickness was calculated using equation 2.1 and equation 2.2 (refer to section 2.5) and the HVL value from Table 6.3

$$0.4001 = e^{-\frac{0.693}{37.1(\text{cm})}\text{X(cm)}}$$

$$\text{x} = \underline{4.9\,\text{cm}}$$

A.3.2.2 Estimated brick thickness

Barrier thickness was estimated using equation 6.1 (refer to section 6.3).

$$\text{eqx}_{\text{brick}} = 4.9 \times 2.22 = \underline{11\,\text{cm}}$$

A.4 calculate the total weekly dose from the energy deposition

$$\text{D(Gy)} = \frac{\text{dE(J)}}{\text{dm(Kg)}}$$

Where D = the absorbed dose

dE = the total energy deposition

dm = absorber mass (TLD)

$$\text{dE} = \text{energy deposition per source particle} \times \text{total activity}$$

$dm = \text{density} \times \text{volume}$

$$D(Gy) = \frac{3.6015E\text{-}09(MeV) \times 370E+6(Bq) \times 1.44E05(s) \times 1.602E\text{-}13(J/MeV)}{2.64E+03(Kg/cm^3) \times 6.843E-04(cm^3)} = 4.49E-05\ Gy$$

Total weekly dose = $4.49E\text{-}05 \times 10^6 = \underline{44.9\ \mu Sv}$

Appendix B

MEASUREMENTS RAW DATA

B.1 Bricks density

Table B-1: shows the raw data of measured brick density

Number of measurements	Height (cm)	Length (cm)	Width (cm)	Volume (cm^3)	Weight (g)	Density (g/cm^3)
1	7.1	22.3	10.7	1694.1	3600	2.125
2	7.1	22.3	10.7	1694.1	3700	2.184
3	7.2	22.4	10.7	1725.7	3700	2.144
4	7.1	22.2	10.6	1670.8	3600	2.155
5	7.2	22.2	10.7	1710.3	3700	2.163
Average	7.14	22.28	10.68	1699.0	3660	2.154
Standard Deviation	0.05	0.08	0.045	20.5	54.8	0.02

B.2 HVL

Calculated using survey meter and difference bricks thickness at 1m from ^{18}F Source

Table B-2: shows the row date of measured dose rate for different bricks thicknesses

Bricks thickness (cm)	Measured dose rate (μSv)
0	10.73
7.1	9.34
10.7	8.71
14.2	8.1
22.3	7.5
29.2	7.01

B.3 Annual exposure

Table B-3 shows the row data of measured annual exposure for some points in the facility.

TLD Numbers	Interest Points	TLD reading (Rem)	TLD reading[a] (Rem)	Occupational exposure per study (μSv)	the Error in TLD reading
1	P1	0.046	0.016	62.0	0.02
2	P2	0.034	0.004	14.7	0.11
3	P3	0.076	0.046	211.9	0.01
4	P4	0.038	0.008	33.4	0.04
5	P5	0.036	0.006	27.6	0.05
6	P6	0.035	0.005	22.2	0.05
7	P7	0.033	0.003	13.4	0.09
8	P8	0.023	-0.007	-29.5	0.04
9	P9	0.046	0.016	71.6	0.02
10	P10	0.033	0.003	12.4	0.10
11	P11	0.032	0.002	8.5	0.14
12	P12	0.038	0.008	36.8	0.03
13	P13	0.02	-0.01	-37.3	0.04
14	P14	0.024	-0.006	-21.4	0.07
15	P15	0.035	0.005	17.6	0.11
16	P16	0.032	0.002	7.3	0.19
17	P17	0.031	0.001	3.7	0.35
18	P18	0.01	-0.02	-73.2	0.02
19	P19	0.012	-0.018	-72.9	0.02
20	P20	0.021	-0.009	-30.3	0.07
21	P21	0.029	-0.001	-3.4	0.47
22	P22	0.027	-0.003	-10.1	0.18

[a] TLD reading corrected value (TLD reading – TLD background reading).

Appendix C

CONCRETE AND BRICK COMPOSITION

C.1 Concrete composition

Table D-1: Concrete Composition [21].

Element	Atomic density (atoms/cm^3)
Fe	0.7784
H	0.6187
C	17.52
O	41.02
Na	0.02706
Mg	3.265
Al	1.083
Si	3.448
K	0.1138
Ca	32.13

C.2 Brick composition

A brick sample was prepared as pressed powder briquettes and introduced to the ARL 9400XP+XRF spectrometer, and analysed using the Quantas software at XRF and XRD facility Geology Department, Faculty of Natural and Agriculture Science, University of Pretoria. Table D-2 indicates the brick composition.

Table D-2: Brick composition

Element	*Atomic density (atoms/cm^3)*
Si	118.8
Ti	1.8
Al	70.2
Fe	25.47
Mn	0.08
Mg	0.15
Ca	0.22
Na	0.17
K	2.47
P	0.35
Cr	0.12
Ni	0.02
V	0.15
Zr	0.06
S	3.06
Ba	0.09

XRF REPORT:

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http://www.up.ac.za/academic/science/

Faculty of Natural & Agricultural Sciences
XRD & XRF Facility
Geology Department

CLIENT: Walid Alsig

DATE: 29 November 2006

ANALYSIS: The sample was prepared as pressed powder briquettes and introduced to the ARL 9400XP+ XRF spectrometer.
Analyses were executed using the Quantas software.
The software analyse for all elements in the periodic table between Na and U, but only elements found
above the detection limits were reported.
The analytical total is low as no Loss On Ignition was determined to estimate the H2O and CO2 content.

Brick	**Brick**

SiO_2	59.40	**K_2O**	2.47
TiO_2	0.90	**P_2O_5**	0.07
Al_2O_3	23.40	**Cr_2O_3**	0.04
Fe_2O_3	8.49	**NiO**	0.02
MnO	0.08	**V_2O_5**	0.03
MgO	0.15	**ZrO_2**	0.03
CaO	0.22	**SO_3**	1.02
Na_2O	0.17	**BaO**	0.09

If you have any further queries, kindly contact the laboratory.

Analyst: 

M.Loubser

Appendix D

MONTE CARLO SIMULATION OUTPUT FILE

D.1 Output file

```
1mcnp     version 4b    ld=02/04/97                        12/15/06
15:00:19

***********************************************************************
**                 probid =   12/15/06 15:00:19
 i=PET5

    1-       Title:shielding calculation of PET facility
    2-       C cell carsd for PET facility
    3-       1  1 -1.0   -1                          u=1
imp:p=1
    4-       2  0         1                          u=1
imp:p=1
    5-       3  0         10 -11 12 -13 14 -15       u=2    fill=1
imp:p=1
    6-       4  3 -2.154   -10:11:-12:13:-14:15       u=2
imp:p=1
    7-       5  3 -2.154   20 -21 22 -23 24 -25              fill=2
imp:p=1
    8-       6  2 -2.64   30 -31 32 -33 34 -35
imp:p=1
    9-       7  0   -40  (-20:21:-22:23:-24:25) (-30:31:-32:33:-34:35)
imp:p=1
   10-       8  0    40
imp:p=0
   11-
   12-       C cylinderical phantom
   13-       1 so 10
   14-       c inner room surface
   15-       10 px -190
   16-       11 px 90
   17-       12 py -110
   18-       13 py 280
   19-       14 pz -100
   20-       15 pz 212
   21-       c outer room surface
   22-       20 px -190
   23-       21 px 113
   24-       22 py -133
   25-       23 py 303
   26-       24 pz -128
   27-       25 pz 240
   28-       c TLD surface of room below
   29-       30 pz 0
   30-       31 pz 0.9
```

```
   31-       32 py 0
   32-       33 py 0.32
   33-       34 px -243.9
   34-       35 px -243
   35-       c spher surface
   36-       40 so 600
   37-
   38-       c material cards
   39-       m1 1000 2 8000 1
   40-       m2 3000 1 9000 1
   41-       m3 14000 -59.4 22000 -0.9 13000 -46.8 26000 -16.98 25000 -
0.08 12000 -0.15
   42-             20000 -0.22 11000 -0.34 19000 -4.94 15000 -0.14
24000 -0.08 28000 -0.02
   43-             23000 -0.06 40000 -0.03 16000 -1.02 56000 -0.09 8000
-223.21
   44-       c source card
   45-       SDEf  rad=d1 erg=0.511 par=2 vec -1 0 0 dir=d2
   46-       si1  0  9.99
   47-       sp1 -21  2
   48-       si2  -1 0.9 1
   49-       sp2  0 0.95 0.05
   50-       sb2 0 0 1
   51-       c tally cards
   52-       *F8:P 6
   53-       c other cards
   54-       mode p
   55-       nps 100000000
   56-       print
   57-
1source
print table 10

 values of defaulted or explicitly defined source variables

    cel        0.0000E+00
    sur        0.0000E+00
    erg        5.1100E-01
    tme        0.0000E+00
    pos        0.0000E+00   0.0000E+00   0.0000E+00
    x          0.0000E+00
    y          0.0000E+00
    z          0.0000E+00
    ext        0.0000E+00
    axs        0.0000E+00   0.0000E+00   0.0000E+00
    vec       -1.0000E+00   0.0000E+00   0.0000E+00
    ccc        0.0000E+00
    nrm        1.0000E+00
    ara        0.0000E+00
    wgt        1.0000E+00
    eff        1.0000E-02
    par        2.0000E+00



 probability distribution   1 for source variable rad
 power law 21:     f(x)=c*abs(x)**k          k = 2.0000E+00
```

```
 probability distribution   2 for source variable dir
 biased histogram distribution

 source        source          cumulative           biased
probability          biased             weight
  entry         value          probability        cumulative
of bin           probability        multiplier

    1      -1.00000E+00       0.000000E+00       0.000000E+00
0.000000E+00       0.000000E+00       1.000000E+00
    2       9.00000E-01       0.000000E+00       0.000000E+00
0.000000E+00       0.000000E+00       1.000000E+00
    3       1.00000E+00       5.000000E-02       1.000000E+00
5.000000E-02       1.000000E+00       5.000000E-02

     the mean of source distribution   2 is  1.3878E-17


 order of sampling source variables.
 rad pos vec dir erg tme
1tally   8
print table 30
           tally type 8*   energy deposition
           tally for  photons
 cells     6
1material composition
print table 40

 the sum of the fractions of material    1 was 3.000000E+00

 the sum of the fractions of material    2 was 2.000000E+00

 the sum of the fractions of material    3 was 3.544600E+02

 material
  number     component nuclide, atom fraction

     1            1000, 6.66667E-01       8000, 3.33333E-01
     2            3000, 5.00000E-01       9000, 5.00000E-01
     3           14000, 1.15457E-01      22000, 1.02617E-03      13000,
9.46884E-02      26000, 1.65985E-02
                 25000, 7.94940E-05      12000, 3.36909E-04      20000,
2.99664E-04      11000, 8.07350E-04
                 19000, 6.89742E-03      15000, 2.46747E-04      24000,
8.39921E-05      28000, 1.86037E-05
                 23000, 6.42980E-05      40000, 1.79528E-05      16000,
1.73658E-03      56000, 3.57771E-05
                  8000, 7.61605E-01

 material
  number     component nuclide, mass fraction

     1            1000, 1.11902E-01       8000, 8.88098E-01
     2            3000, 2.67583E-01       9000, 7.32417E-01
```

```
     3           14000, 1.67579E-01      22000, 2.53907E-03      13000,
1.32032E-01      26000, 4.79039E-02
                 25000, 2.25695E-04      12000, 4.23179E-04      20000,
6.20662E-04      11000, 9.59206E-04
                 19000, 1.39367E-02      15000, 3.94967E-04      24000,
2.25695E-04      28000, 5.64239E-05
                 23000, 1.69272E-04      40000, 8.46358E-05      16000,
2.87762E-03      56000, 2.53907E-04
                  8000, 6.29718E-01

 warning.    3 materials had unnormalized fractions. print table 40.
1cell volumes and masses
print table 50

           1tally   8        nps =100000000
           tally type 8*   energy deposition
units   mev
           tally for  photons

 cell  6
                3.60154E-09 0.1844
1analysis of the results in the tally fluctuation chart bin (tfc) for
tally   8 with nps = 100000000    print table 160


 normed average tally per history  = 3.60154E-09          unnormed
average tally per history  = 3.60154E-09
 estimated tally relative error    = 0.1844               estimated
variance of the variance  = 0.0458
 relative error from zero tallies  = 0.0439               relative
error from nonzero scores  = 0.1791

 number of nonzero history tallies =         519          efficiency
for the nonzero tallies  = 0.0000
 history number of largest  tally  =     9985628          largest
unnormalized history tally = 1.69685E-02
 (largest  tally)/(average tally)  = 4.71147E+06          (largest
tally)/(avg nonzero tally)= 2.44525E+01

 (confidence interval shift)/mean  = 0.0192               shifted
confidence interval center  = 3.67054E-09


 if the largest  history score sampled so far were to occur on the next
history, the tfc bin quantities would change as follows:

      estimated quantities           value at nps           value at
nps+1           value(nps+1)/value(nps)-1.

      mean                             3.60154E-09             3.77122E-
09                    0.047115
      relative error                   1.84404E-01             1.81764E-
01                   -0.014317
      variance of the variance         4.58366E-02             4.41462E-
02                   -0.036879
      shifted center                   3.67054E-09             3.67151E-
09                    0.000264
```

```
      figure of merit                 4.09624E-02             4.21610E-
02                     0.029260

 the estimated slope of the  25 largest  tallies starting at  6.44537E-
03 appears to be decreasing at least exponentially.
 the large score tail of the empirical history score probability
density function appears to have no unsampled regions.


================================================================================
============================================================

           results of 10 statistical checks for the estimated answer
for the tally fluctuation chart (tfc) bin of tally   8

 tfc bin     --mean--      ---------relative error---------      ----
variance of the variance----      --figure of merit--     -pdf-
 behavior    behavior      value   decrease   decrease rate      value
decrease   decrease rate       value     behavior     slope

 desired      random       <0.10      yes      1/sqrt(nps)       <0.10
yes        1/nps           constant    random      >3.00
 observed     random        0.18      yes          yes            0.05
yes         yes            constant    random      10.00
 passed?        yes           no      yes          yes             yes
yes         yes               yes        yes         yes


 tally   result of statistical checks for the tfc bin (the first check
not passed is listed) and error magnitude check for all bins

     8   missed  1 of 10 tfc bin checks: the relative error exceeds the
recommended value of 0.1 for nonpoint detector tallies
         missed all bin error check:     1 tally bins had     0 bins
with zeros and     1 bins with relative errors exceeding 0.10


 the 10 statistical checks are only for the tally fluctuation chart bin
and do not apply to other tally bins.

 warning.     1 of the   1 tally fluctuation chart bins did not pass
all 10 statistical checks.
 warning.     1 of the   1 tallies had bins with relative errors
greater than recommended.
1tally fluctuation charts

                      tally    8
       nps      mean     error   vov  slope    fom
   8192000   1.6674E-09 0.8777 0.9555  0.0 2.2E-02
  16384000   5.2858E-09 0.3876 0.1977  0.0 5.7E-02
  24576000   5.0514E-09 0.3190 0.1387  0.0 5.6E-02
  32768000   4.5371E-09 0.2840 0.1144  0.0 5.3E-02
  40960000   4.3629E-09 0.2607 0.0927  0.0 5.0E-02
  49152000   4.1468E-09 0.2429 0.0816  0.0 4.8E-02
  57344000   4.2532E-09 0.2219 0.0683  0.0 4.9E-02
  65536000   4.0694E-09 0.2118 0.0630  0.0 4.7E-02
```

```
  73728000   4.2362E-09 0.1996 0.0532  0.0 4.7E-02
  81920000   3.9003E-09 0.1964 0.0520  0.0 4.4E-02
  90112000   3.7858E-09 0.1906 0.0489  0.0 4.3E-02
  98304000   3.6637E-09 0.1844 0.0458 10.0 4.2E-02
 100000000   3.6015E-09 0.1844 0.0458 10.0 4.1E-02


************************************************************************
*********************************************
 dump no.   49 on file runtpn       nps = 100000000     coll =
316525761     ctm =  717.92     nrn =     4526017044


         4 warning messages so far.


 run terminated when 100000000 particle histories were done.

 computer time =  718.13 minutes

 mcnp     version 4b    02/04/97                     12/16/06 02:59:22
probid =   12/15/06 15:00:19
```