Sudan University of Science &Technology
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Measurement of Exposure Level during Technetium Generator Elusion at Nuclear Medicine lab

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Thesis submitted for partial fulfillment ofRequirements of the M. Sc. Degree in Nuclear Medicine

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Dedication

To my lovely parent

Who

Taught me how to make my future.

To my sweetly sister and brother

Acknowledgements

First of all, I would like to thank Allah, who gives us everything and without him nothing can be done.

I would like to thank everyone who helps me to carry out this work and make it possible. Special thank to my supervisor Dr. Mohammed Ahmed Ali; for his patience, encouragement and collaboration, working staff in related hospital. And special thanks to my encourage-able parent, sisters and my brothers.

Finally deepthank s to Mohamed Omerwho support me and believe in me and always stand beside me And make the impossible be possible.

Abstract

The following experimental study has been carried out at radiation and Isotopes Center of Khartoum in Sudan, aiming to determine technetium generator time activity curve i.e. decay process, dose measurement received by the technologist hands, the exposure level relative to distance at hot lab and the cycle of waste management. With the usage of Geiger Muller survey detector the exposure level at different points, hands, and abdomen of the technologists have been measured, as well the radioactive waste management has been assessed and compared with IAEA recommendation.

The obtained results showed that: The technetium generator has a time activity curve that fits the exponential equation of the following form: $y = 886.2e^{-0.25x}$, where x refers to days and y refers to the relative activity. And the exposure level at different points distal from the generator could be calculated based on the following equation $y = 24.26e^{-0.63x}$, where x refers to the point and y refers to the accumulative dose during generator working days. And the technologist's hands received radiation dose of 34 (mCi/h) and the abdomen received a dose of 4.5 (mCi/h). And the waste management has been carried out based on the IAEA recommendation.

المستخلص

تم إجراء هذه الدراسة في مركز الخرطوم للعلاج بالأشعة والطب النووي في الفترة يوليو 2013 وحتى مارس 2014 وكان الهدف من هذه الدراسة تحديد زمن اضمحلال مولد التتكنيشيوم عن طريق منحني النشاط الإشعاعي , وتم قياس مستوي التعرض في المختبر الساخن وعلاقته بالمسافة وكيفيه إدارة النفايات المشعه .

تم قياس هذه القيم عن طريق استخدام كشاف جيجر مولر عند مختلف النقاط وعند منطقه اليدين والبطن ، وتم مقارنه هذه النتائج مع توصيات الوكالةالدوليةللطاقةالذريةوأظهرت النتائج المتحصل عليها الأتي:

إن مولد التكنيشيوم لديه زمن اضمحلال متمثل في منحني النشاط الذي يتناسب مع المعادلة الاسيةالتأليه : $y = 855.e^{-0.247x}$ حيث يشير محور xإلىالأيام ومحور yإلى النشاط النسبي . وتم قياس مستوي التعرض عند نقاط مختلفة من المولد كما هو معطي في المعادلة$y = 1.926e^{-0.261x}$ حيث يشير محور xإلى النقاط ومحور yإلى الجرعة التراكمية التي تلقتها يدي التقني و الجرعة التراكمية الواصلة للبطن .

وتمت عمليه التخلص وإدارة النفايات المشعة اعتمادا علي توصيات الوكاله الدولية للطاقه الذرية .

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Chapter One

Introduction

1-1Background

Nuclear medicine is a branch of radiology which use as diagnostic and therapy for many disease by using radio nuclide which mixed by radio pharmaceutical depend on metabolic process of organ. Unlike MRI and CT, nuclear medicine uniquely provides information about both the function and structure of organ systems within the body. While the introduction of ^{131}I for treating thyroid disease in 1946, followed a few years later by ^{131}I thyroid imaging, marks the beginning of Nuclear Medicine, it was the discovery of ^{99m}Technetium in 1937 and the subsequent development of the first commercial ^{99m}Tc generator in 1964 that lead to the tremendous growth of nuclear medicine. For nuclear imaging, Tc^{99m} has become the "universal" isotope because of its virtually ideal physical characteristics for scintigraphic applications (i.e., generator produced, 6-h half-life, 140 keV gamma radiations) and its versatile chemistry that can be manipulated to label a variety of ligands. The Mo/Tc generator can be shipped to laboratories for the production of single dose Tc^{99m} radiopharmaceuticals on site making Tc^{99m}, by far, the most utilized radioisotope in nuclear medicine. Other isotopes require cyclotron or reactor generation, which is more costly, and less available for emergency or rapid administration. Today, there are nearly 100 nuclear medicine imaging procedures available using various single photon emission isotopes and positron emission isotopes. The availability of 18F and specifically, 18F-fluorodeoxy-D glucose (FDG) has allowed for the practical application of PET. The short half-life of most PET isotopes, with the exception of the 110-minute half-life of 18F, makes them impractical for routine use because they require a cyclotron on site at the hospital. Gamma emission imaging has been successfully applied to almost every organ of the body (brain, bone heart, kidney, lung, and neuroreceptors) as well as sites of inflammation, atherosclerosis, thrombosis and cancer. The molecular nature of nuclear medicine

imaging leads to unique non-invasive pharmacokinetics modeling applications. In addition, the unique characteristics of PET allow for quantitative analysis of physiological processes, particularly cellular metabolism. The future of nuclear medicine imaging radiopharmaceuticals lies primarily in the development of new legends for Tc^{99m} (for SPECT) and 18F (for PET) to carry the radioisotope to the site of application without compromising the biological activity of the legend molecule rather than in the development or discovery of new radioisotopes (Clarke 2004)

The characteristics of an ideal radioisotope for therapy are quite different from those required for imaging; the radioisotope's energy must be deposited in the camera crystal, without significant absorption by the tissue. Whereas, the energy of a therapeutic radioisotope must be deposited in the tissue to damage the DNA chains to keep the diseased cells from replicating. No single isotope dominates radioisotope therapy as ^{99m}Tc dominates nuclear imaging (imam 2001)

Since NM has been significant however the hazard of radiation protection is very important in nuclear medicine because in many situations awareness of time factor may be used to reduce exposure work should be done as quickly as possible near radiation sources, while the levels of activity used in department of nuclear medicine have raised dramatically in the last decade. This has been due to an expansion in the use of the techniques, rather than an increase in dose to patients which if anything have been reduced a little the implication for technical and medical staff, however some different handling much larger quantities of radioactive materials each year might well be expected to increase their personal dose on the whole, however, this has not happened since betters radiation management methods, such as improvement in design and shielding generator

Anew procedure should be firstly evaluated to have familiarity with procedure before they are done near radiation sources-during elusion technologist should stay shortest time.As a result of radiation action in human cell, protection must be provided to reduce the over doses that might cause radiation reaction.

The organization introduced (ALARA concept as low achievable). This concept recommended with the minimized of radiation exposure should be justified in all fields e.g. occupational, medical and industrial (imam 2001).

To minimize or to set the radiation exposure dose there are three principles should be followed:

- Distance (increase distance result in reduce of dose following in verse square low).
- Shielding (adding shield can also reduce radiation doses by means of absorption).
- Time (Reducing the time of exposure).

For radiation safety issue, nuclear medicine facilities are divided into two parts: i) controlled area, and ii) supervised area. Hot lab is named as controlled area/room where radionuclides are always stored. The workers have to work in the hot lab. The workers spend most of their time for patient dose preparation, therefore, keeping exposure below the radiation protection issue. Hot lab is a specially designed room in a nuclear medicine hospital where the radiopharmaceuticals are delivered, stored and prepared for dispensing. Mo^{99} –Tc^{99m} -generator is the major source in the hot lab used for various medical imaging. It is important to maintain a standard for hot lab procedures to optimize the patient care and minimize radiation exposure to all nuclear medicine personnel, patients, public, as well as environment (IAEA).Vienna (1996).

1.2Effects of radiation:

Somatic effects

The somatic effect of radiation arise mainly from the depletion of the numbers of mitotic, dividing cells or from interference which are undergoing very rapid cell division; for example the cells of the lymphoid and granulocyte systems and the cells lining the gastro-intestinal tract. Because of their rate of growth young animals are more susceptible to irradiation damage than adults.The somatic effects of radiation also fall into two classes. The acute effects which are

observed within a few days or week of exposure to relatively large doses of radiation and the long-term effects which arise years after exposure to much lower doses.(IAEA).Vienna (1996).

1.2.1Hereditary effects

The hereditary effect of radiation arises from radiation induced mutations which occur in the egg cells, or ova, in the ovary of the female, or in the sperm cells of the testis in the male. Most genetic damage will probably result in embryonic or fetal death however some may be nonlethal and permit the birth of a viable child suffering from severe abnormalities of one type or another. For this reason in all medical investigations involving radiation care is taken to avoid any unnecessary irradiation of the gonads of either the patients or the investigators.(IAEA).Vienna (1996).

1.2.2Effects of Radiation on the Embryo:

Irradiation of pregnant female may cause serious damage to the developing embryo of fetus. A large number of observations on children born to women who were exposed to diagnostic or therapeutic x-rays during pregnancy have shown that irradiation in utero may lead to serious abnormality in the offspring.(IAEA).Vienna (1996).

1.3 Problem of the study:

There is a potential hazard of radioactive nuclei at hot lab of nuclear medicine department during elusion and the waste left for the technologist and the medical physicists.

1.4 The general objective:

The general objective of this study is to evaluate radiation exposure during Tc-99m elusion and the dose received by technologist at hot lab in radiation and isotopes centre in Khartoum (RICK).

1.4.1 Specific objectives:

- To assess the decay of Technetium generator versus days
- To correlate the daily exposure level versus different points at hot lab
- To determine the amount of exposure at the technologists hands
- To draw the chart for radioactive waste management.

1.5 Signifincane of the study :

In nuclear medicine department (hot lap) is use technetium generator for elution (tco4) for different procedure. This study will try to extract the value that expose to technologist hand and abdomen during elution in different point in 13 days.

1.6 General scope of the study:

The frame work of this study consists of five chapter , Chapter one: dealing with introduction the researchproblem ,objectiveand view of the study Chapter two: literature review,Chapter three: methodology of research including the theory of the research the material used in measurement Chapter four: dealing with data analysis and result. AndChapter five: dealing with discussion and recommendation.

Chapter Two

Literature Review

2-1 Hot lab Design and preparing Radiopharmaceuticals:-

The radio pharmacy would normally be designated as a controlled area; and access will be restricted. Only properly trained staff should be permitted to work in the radio pharmacy; and strict adherence to work procedures is essential.

There are three fundamental parameters that affect staff doses in the radio pharmacy:-

- The distance between the staff member and the source.
- The time spent manipulating the source and.
- The amount of shielding used to reduce the dose rate from the source.

Sometimes there is a tradeoff between these parameters as using more shielding might increase handling time. With this in mind, careful design of procedures should optimize the workflow. Skill and expertise of the staff carrying out the procedures are also important factors. Thus, it is crucial that staffs are adequately trained.(IAEA).Vienna (1996).

Work practices in the radiopharmacy should be standardized and incorporated in standard operating procedures (SOPs). These procedures should be documented and made readily available to those working in the radiopharmacy.This will ensure harmonization of practice and maintenance of standards. Accurate and comprehensive record keeping is an essential part of good work practice in a radio pharmacy. Most radiopharmaceuticals are administered by IV injection; so good pharmaceutical practice is an important consideration in their preparation. All manipulation of radioactive materials should be carried out, using aseptic techniques, within the shielded contained workstation or laminar flow cabinet (LAFC) (Figure 3.3). No food or drink, cosmetic or smoking materials, crockery or cutlery should be brought into an area where unsealed radioactive substances are used.(IAEA).Vienna (1996).

2-2 Generator:-

Generatorsare units that contain a radioactive, . It consists of a heavily shielded column with molybdenum-99 (99Mo; parent), bound to the alumina of the column. The ^{99m}Tc (daughter) is "milked" (eluted) by drawing sterile saline through the column into the vacuum vial. The parent 99Mo (small grey circles) remains on the column, but the daughter ^{99m}Tc is washed away in the saline Certain parent-daughter systems involve along-lived parent radionuclide that decays to a short-lived daughter. Since the parent and daughter nuclides are not isotopes of the same element, chemical separation is possible .The long-lived parent produces a continuous supply of relatively short-lived daughter radionuclide and there for called generator. The generator systems used in nuclear medicine are listed in the following table 3.1.

Table 2.1 Decay properties for parent and daughter radionuclide of several generators:

Generator	**Parent $T^1/_2$**	**Daughter $T^1/_2$**	**Daughter E_γ**	%
99_{Mo}_$99m_{Tc}$	2.78 d	6 hr	140 kev	90
81_{Rb} _$81m_{Kr}$	4.7 hr	13 sec	190	65
113_{Sn} _$113m_{In}$	115 d	1.7 hr	393	64
68_{Ge} _ 68_{Ga}	280 d	68 min	511	176
62_{Zn} _ 62_{Cu}	9.3 hr	9.8 min	511	196
82_{Sr} _82_{Rb}	25 d	1.3 min	511	192

There are two types of parent daughter relationships known as transient an\d secular equilibrium. In a transient equilibrium system the half-life of the parent is a factor 10-100times greater than of the daughter. Where as in secular equilibrium the half-life of the parent is 100-1000times greater than that of the daughter. The Mo^{99} –Tc^{99m} where Mo^{99} has a 67 hr half-life and Tc^{99m} has a 6 hr half-life is an example of transient equilibrium.

Example of secular equilibrium : Equations governing generator system: assuming

There is initially no daughter activity present in a generator; the daughter activity at any given time can be calculated from the following general equation:

((1) $$A_2 = \frac{\lambda_2}{\lambda_2 - \lambda_1} \times A_1^o (e^{-\lambda_1 t} - e^{-\lambda_2 t})$$

Where A_1^o is the parent activity at time zero, A_2 is the activity daughter at time and $\lambda_1 and \lambda_2$ are the decay constants for the parent and daughter, respectively. This general equation can be simplified for the special cases of transient and secular equilibrium is reached after a certain time in which the decay rates of the parent and daughter are equal. In case of transient equilibrium as the time to becomes sufficiently large, $e^{-\lambda_2 t}$ is negligible compared with $e^{-\lambda_1 t}$ and the equation can be simplified as follows:

$$A_1 = A_1^o e^{-\lambda_1 t}$$

Since where A_1 is the parent activity at time t, this equation can be rewritten as:

$$A_2 = \frac{\lambda_2}{\lambda_2 - \lambda_1} \times A_1^o e^{-\lambda_1 t}$$

$$A_2 = \frac{\lambda_2}{\lambda_2 - \lambda_1} A_1$$

Assuming the parent only decays to the daughter, at equilibrium the daughter activity will be greater than the parent activity by the factor $\lambda_2 / \lambda_2 - \lambda_1$ at equilibrium, both activities then appear to decay with the half-life of the parent. In the specific case of the Mo^{99} –Tc^{99m} generator there is only 86% decay of the Mo^{99} to Tc^{99m} This makes the factor 0.86 $\lambda_2 / \lambda_2 - \lambda_1$ which simplifies to 0.86 × 1.1 there for the actual Tc^{99m} activity present at transient equilibrium is0.946 times the Mo^{99} activity. In the case of secular equilibrium, the parent activity dose not decrease significantly during many daughter half-lives the decay constant of the parent λ_1 is much smaller than that of daughter λ_2 and the following approximation can be made:

This approximation can be used to further simplify equation to yield the following expression:

$$A_2 = A_1$$

$$\lambda_2 - \lambda_1 \approx \lambda_2$$

Thus, at equilibrium the daughter activity is equal to the parent activity.

PRINCIPLE AND DESCRIPTION OF TYPICAL GENERATOR:

Mo^{99} –Tc^{99m} Generator: This generator is commonly used in nuclear medicine because of the ideal half life (6 hr) and optimum energy (140 Kev 90%abundance) of Tc^{99m} A large number of radiopharmaceuticals are made with Mo^{99} –Tc^{99m}. Molybenum-99 $t^{\mathbf{1}}/_{\mathbf{2}}$=67 hr) the anionic species molybdate ($\boldsymbol{MO^{2-}}$)

And paramolybdate ($\boldsymbol{MO_7O_{24}^{6-}}$) in acidic medium these anions are loaded on the generator column containing positively charged alumina ($\boldsymbol{A_2O_3}$)that has previously been washed with saline0.9%NACL and Tc^{99m} is produced as pertechnetateTc^{99m} O^{-}_{4}. The two type of Mo^{99} –Tc^{99m} generator used in nuclear medicine are the wet column generators. The wet column contains a reservoir of normal saline that is connected to the alumina column. After elation of this generator saline remains on the column, leading to the formation of water radiolysis produces which are reducing agent this causes reduction of the Mo^{99} and decrease Tc^{99m} O^{-}_{4} yields since the reduced Tc^{99m} species do not elute from the column. This problem has been addressed by purging the saline reservoir with $\boldsymbol{O_2}$ previous attempts to add oxidizing agents to the column to decrease reduction of Tc^{99m} species resulted in Tc^{99m} radiopharmaceutical for mutation problems.

The dry column generator system was developed to alleviate poor elution yields of Tc^{99m} O^{-}_{4} by removing saline from the column after elution. This decrease the amount of radiolysis products formed. The dry column generator employs a 5-20ml saline charge, which applied to an exterior port of the generator. An evacuated vial draws saline through the generator to remove Tc^{99m} O^{-}_{4} followed by air to dry the column leaving. The air on the column promotes oxidation of any reduced Tc^{99m} species back to the +7 valence state of Tc^{99m} O^{-}_{4} which can then be eluted (*R, Donald etal (2oo4)vol 7).*

^{113}Sn-^{113m}In generator:

Indium-113rn can be used to prepare a number of radiopharmaceuticals for imaging lungs, liver, brain, and kidneys. Tin-113 is produced in a reactor by neutron irradiation of ^{112}In. The ^{113}Sn is then loaded onto a generator column containing hydrous zirconium oxide. Elution of ii3m1 (t½ =7 hr) is achieved with 0.05M HC1. Because of the long half-life of ^{113}Sn (115 days), the ^{113}Sn-^{113m}In generator can be used for 6 to 12 months, making it one of the most economical generators. The biggest drawback of this generator is that the photon energy of ^{113m}In (393 keV) is not ideal for use with the gamma camera. In the United States the ^{113}Sn-^{113m}In generator has largely been replaced by the ^{99}Mo/^{99}mTC generator; however, the ^{113}Sn-^{113m}In generator is still useful in developing countries and more isolated regions of the world.

^{82}Sr-^{82}Rb generator:

Rubidium-82, a positron-emitting radionuclide, is used primarily as a myocardial perfusion agent for PET imaging. The rubidium cation (Rb^+) is an analogue of potassium (K^+) and therefore gives a similar biodistribution. Strontium-82 (T½ = 25 days) is accelerator produced by bombardment of a molybdenum target wit 700 to 800 MeV protons. Strontium-Strontium-85 is also produced as a radionuclide impurity. The allowable limit for ^{82}Sr is 0.02 µCi/mCi^{82}Rb, and for ^{85}Rh is 0.2 µCi/mCi^{82}Rb. The generators regularly meet this requirement after a first elution of the generator to waste. The ^{82}Sr is loaded onto a stannic oxide column, and ^{82}Rb (t½=76sec) is eluted with normal saline (0.9% NaCI). The 76-sec hlf-1ife of ^{82}Rb allows repeat imaging studies after 10 half Jives of the ^{62}Rb, approximately 14 minutes, but poses difficulties in dose preparation for patient administration. Also the limited chemistry of this alkali metal ion severely restricts potential applications for this radionuclide in nuclear medicine. ln an effort overcome the short half-life, a calibrated continuous infusion system has been developed, allowing elution of the generator directly into an intravenous catheter. The ^{82}Rh generator is an FDA-approved radiopharmaceutical produced

by Bracco Diagnostics for myocardial perfusion imaging (*R, Donald etal (2oo4)vol 7).*

^{81}Rb-^{80}mKr generator:

Kryptod-81m, a gamma ray-emitting nuclide with a photon energy of 190keV (65% abun-dance), is used as a lung-imaging agent. Rubidiurn-81 (T½= 4.7 hr), cyclotron produced by the reaction ^{79}Br (α,2n) ^{81}Rb or ^{82}Kr (α,2n) ^{81}Rb, is loaded onto a generator column containing a strong cation-exchange resin (Bio Rad AGMP-50). The noble gas 81mKr (T½=13 see) is eluted by passing humidified oxygen over the generator column. The ^{81}m Kr and 0_2 are delivered to the patient via a nonrebreathing face mask. The major disadvantages of the ^{81}Rh/^{81}mKr generator are the high cost and the 12-hr expiration time of the generator due to the 4.5 hr half-life of the parent isotope. This limits the use of the generator to the day of delivery only. The generator is an FDA-approved radiopharmaceutical produced by Medi-Physics used for lung ventilation imaging (*R, Donald etal (2oo4)vol 7).*

^{68}Ge-^{68}Ga generator:

Gallium-68 (t½= 68 mm) emits a 2.92 MeV positron in 89% abundance, making it very useful in PET imaging. The ^{58}Ge/^{68}Ga generator is not FDA approved, and only a few ^{68}Ga radiopharmaceuticals have been investigated clinically these include ^{68}Ga macroaggregated albumili (MAA), ^{68}Ga citrate, and ^{68}Ga ethylenediaminetetraacetic acid (EDTA). Several generator systems have been developed to separate 68Ga from ^{68}Ga. The early generators consisted of ^{58}Ge adsorbed onto an alumina column. The ^{68}Ga was eluted with a 0.005 M solution of EDTA at pH 7[14] However, because ^{68}Ga forms such a strong complex with EDTA at neutral pH, it was difficult to dissociate ^{68}Ga from ^{68}Ga-EDTA to allow formation of other complexes. A solvent extraction ^{68}Ge-^{68}Ga generator was later developed that produces a ^{58}Ga-oxine chelate. Although ^{68}Ga-oxine was a weaker chelate than ^{68}Ga-EDTA, problems with the operation of the generator system prevented it from becoming clinically useful. Most recently, a generator yielding

^{68}Ga in an ionic form has been developedIn this generator ^{68}Ge is loaded onto a tin dioxide column and ^{68}Ga is eluted using 1 M l-lCl. The only problem incurred in this generator is the presence of trace metals, which sometimes reduces the amount of complexation with 68Ga. This is the only ^{68}Gc-^{68}Ga generator produced commercially (*R, Donald etal (2oo4)vol 7).*

^{62} Zn-^{62}Cu generator:

Copper-62, a positron-emitting radionuclide (98% abundance) with a 9.7 mm half-life, is an attractive radionuclide for PET imaging. ^{62}Cu-Iabeled pyru-valdehyde his (N4-mthylthiosemicarba-zone) (^{62}Cu-PTSM) has been used in clinical investigations for heart and brain blood flow measurement. The parent isotope, ^{62}Zn, is cyclotron produced via the ^{63}Cu(p,2n) ^{62}Zn reaction.

Facilities for the production of ^{62}Zn exist at a number of com--mercial facilities as well as at several clinical PET centers.

The major disadvantage is the short half-life of ^{62}Zn (9.3hr), which requires generator replacement at 1- or 2-day intervals. Two different ^{62}Zn-^{62}Cu generator systems have been developed. In one design, ^{52}Zn is loaded onto a column containing allows 1x 8 anion exchange resin, which retains Zn^{2+} and allows Cu^{2+} to be eluted using 0.2N HCI/1.8N.NaCI or 2N HCl.

The other system employs a column containing a strong cation exchange resin adsorbent (CG-120, Amberlite), and ^{62}Cu is eluted in 0.2M glycine.The eluante in the latter generator is suitable for direct intravenous injection.Clinical use of a ^{62}Cu-tracer requires a convenient method for routine, repetitive, high-yield radiopharmaceutical synthesis using the eluate of a ^{62}Zn/^{62}Cu generator. While previous work with ^{62}Cu-PTSM relied on a fairly simple remote system for radiopharmaceutical synthesis, ^{52}Cu-radtopharmaceutieal preparation has been further simplified by integration of reagent mixing operations into the 20 x 30 x 40-cm housing of a modular generator unit, available from Proportional Technologies, Inc. (Houston, Texas).This modular generator can directly deliver the ^{62}Cu-PTSM in a sterile, pyrogen-free solution suitable for intravenous

injection. The radiopharmaceutical synthesis time is the 40-second period required for generator elution. Such a modular $^{62}Zn/^{62}Cu$ generator system, nationally or regionally distributed from commercial medium-energy cyclotron facilities, may effectively support PET imaging centers as a source of short-lived radiopharmaceuticals for evaluation of tissue perfusion (*R, Donald etal (2oo4)vol 7)*

2.3 Sources of radiation waste in nuclear medicine:

2.3.1 Solid waste

The solid waste generated in nuclear medicine includes cover papers, gloves, empty vials and syringes, radionuclide generators, items used by hospitalised patients after radionuclide therapy, sealed sources used in therapy, sealed sources used for the calibration of instruments, animal carcasses and other biological waste. In the liquid waste category one can find residues of radionuclides, patient excreta, liquid scintillation solutions, and in the gaseous waste exhausted gas from patients in nuclear medicine.Depending on the final handling or disposal, the waste could be divided into different categories. One category where the waste will end up in a public waste treatment system with or without incineration, the next where it will be poured out in the public sewage system, and the third where it will be disposed of in a national plant or recycled .which after proper treatment and conditioning can be handled in the public waste treatment system. Solid waste should be immobilised by surrounding it with a matrix material in order to produce a waste form suitable for storage and transportation governed by the properties of the waste, the transport regulations and the specific waste treatment or disposal acceptance requirements. Incineration is recommended for combustible solid waste because it achieves the highest volume reduction and dilutes and disperses the radionuclide content. Total sterilisation of biological waste will be achieved. Discharge limits given by the regulatory authority should be followed (IAEA) Series No70 ,No 111)

2.3.2 Liquid waste:

Liquid waste which after proper treatment and conditioning can be handled in the public waste treatment system. For aqueous waste evaporation, chemical precipitation and ion exchange may be considered. For organic and infectious liquid waste incineration is a suitable method. The residues of the treatment process should be handled as solid waste. Treated aqueous effluents can be discharged in the environment via the sewage system if either clearance has been granted for the radioactive substance or the discharge is within the limits authorised by the regulatory authority.

Solid waste containing short-lived radionuclides capable of being stored for decay. There after it may be handled in the public waste system if the discharge is within the limits authorised by the regulatory authority ,Liquid waste containing short-lived radionuclides capable of being stored for decay. Thereafter it may be handled in the public waste system if the discharge is within the limits authorised by the regulatory authority, Sealed sources with high activities which will be disposed of in a national plant or recycled, Biological waste, which may undergo decomposition. A suitable treatment method is incineration ,Infectious waste requiring sterilisation possibly by incineration prior to disposal.Broken glass-ware, syringes etc, requiring collection in separate containers to prevent personnel being injured. Depending on other content, radionuclide type, biological, infectious etc. it should be appropriately handled, treated and stored. Radionuclide generators should after storage for decay be checked for contamination and dismounted or, preferably, returned to the producer. Bed linen and clothing from hospital wards, which are contaminated, could be stored for decay if the nuclides are short-lived. Otherwise it is treated as solid waste. Patient excreta. For diagnostic patients there is generally no need for collection of excreta. Ordinary toilets can be used. For therapy patients there are different policies in different countries. Either to use separate toilets equipped with delay tanks or an active treatment system, or allow the excreta to be released directly

into the sewage system ,Liquid scintillation solutions, which due to its flammable organic solution, could preferably be disposed of by incineration. New types of liquid scintillation solutions are less flammable and hazardous to the environment could otherwise be handled in the public waste treatment system. Containers to allow segregation of different types of radioactive waste should be available in areas where the waste is generated. The containers must be suitable for purpose (volume, shielding, leak proof, etc.) Each type of waste should be kept in separate containers properly labelled to supply information about the radionuclide, physical form, activity and external dose rate. A room for interim storage of radioactive waste should be available. The room should be locked, properly marked and if necessary ventilated. Flammable waste should be placed apart. It is essential that all waste be properly packed in order to avoid leakage during the storage. Biological waste should be refrigerated or put in a freezer. Records should be kept where the origin of the waste can be identified.(IAEA) Series No70 ,No 111)

2.4 General radiation safety in nuclear medicine and hot lab:

All personnel handling radioactive materials must document that they have received acceptable training. In addition, each department must document that all personnel have received training in specific aspects of radiation safety when hired and then annually. A component of a Radiation Safety Program is that every department must have published general rules for safe handling of radioactive materials. These should include the following: Where protective clothing, such as laboratory coats, Wear disposable gloves while handling radioactive materials, Monitor hands for contamination before leaving the area, Use syringe shields, Do not eat, drink, smoke, or apply cosmetics where radioactive material is stored or used, Do not store food or drink where radioactive material is stored or used, Wear personnel monitoring devices correctly, Dispose of radioactive waste in designated receptacles, Never pipette by mouth, Conduct surveys and wipe tests as required, Keep flood sources and other radioactive materials in shielded,

labeled containers, Assay each patient dose before administration and Use a cart or wheelchair to move large sources.

Other aspects of initial and annual training should include a review of the license conditions, rights of radiation workers as given in Form NRC-3 Notice to Employees, emergency procedures, personnel monitoring reports, and radiation safety procedures specific to the institution. A written record of training, including the date of the training and a legible list of attendees, must be maintained (IAEA) Series No70 ,No 111)

2.5 Control of Radioactive Materials:

2.5.1Ordering and receiving:

All radioactive materials that are ordered must be listed on the radioactive materials license and must not exceed any limits given on the license. To meet this requirement the Radiation Service Officer (RSO) must review radioactive materials orders. Generally, standing orders for general use must be reviewed annually, whereas “special” orders should be reviewed on a case-by-case basis. In addition, manufacturers or commercial radio-pharmacies must keep a copy of the license on file and should check that materials ordered are approved on the license. Procedures for receiving radioactive materials should detail how the delivery will reach the nuclear medicine pharmacy. If packages are received at a loading dock, personnel working in that area must be trained on safe handling technique including the need for visual checks of packages and procedures to follow if a package is damaged. Packages received after hours must be delivered to a designated location. Typically that location will have a locked, possibly shielded storage space, or a protocol will be in place for security personnel to deliver the package to the radiopharmacy Again, it is necessary for security personnel to be trained in radiation safety techniques. The most straightforward situation exists where a commercial radiopharmacy delivers radioactive material directly to the nuclear medicine department. Often this eliminates transport by security or distribution personnel (IAEA) Series No70 ,No 111)

2.5.2 Opening package:

Radioactive materials received in the nuclear medicine department must be checked in within 3 hours during normal working hours and, if any are received at other times, as soon as the department reopens. This check should consist of donning disposable gloves to conduct a visual inspection of the package, if there is any evidence of damage or wetness, indicating leakage. The package should be immediately secured and the RSO should he notified. Next, all packages requiring White I, Yellow II, or Yellow III Department of transportation labels must be wipe tested when received. The wipe should cover 300 cm^2 of the package surface area. The amount of radioactivity measured should not exceed 22 (dpm/cm^2) for beta and gamma radiations and 2.2 (dpm/cm^2) for alpha particles. The package should then be opened, and a second visual check should be made to ensure that the contents are what was indicated on the packing slip and that there is no damage. Indications if damage would be breakage, less liquid than expected, or discoloration of the packing material. If indications of Jam age are observed, the package should be secured and the RSO should be contacted. Finally, the contents should be removed and the empty shipping package should be surveyed for contamination with a low-range GM meter. Before disposal, all radioactive symbols must be defaced. During all these steps, good radiation safety practices shout be used, such as wearing disposable gloves and using tongs to move the source. Records of receipt of the martial and the findings of the wipe test must be maintained (IAEA) Series No70 ,No 111)

2.6 Maximum permissible Dose in Nuclear Medicine:

The recognition of the potentially harmful effects of radiation founded in 1928, which is now known as the International Commission on Radiological Protection (ICRP) is responsible of watching brief on all aspects of protection against ionizing radiations and makes recommendations concerning basic principles and radiation dose limits. The recommendation of the ICRP has been extensively reviewed recently and the latest recommendations published in 1917 defined a

system of dose limitation designed to ensure that no source of radiation exposure is unjustified in relation to its benefits that any necessary exposures arc kept as low as is reasonably achievable and that the dose equivalents received by individual do not exceed certain specified limits. The IRCP makesrecommendations only and it is the responsibility of the appropriate authorities in individual countries to implement those recommendations which they consider necessary to meet their own particular circumstances (imam s.k (2001)

In setting dose equivalent limits the Commission believe that these should be based on the total risk of all tissues irradiated, In consequence a single dose equivalent limit for the uniform irradiation of the whole body has been set and various weighting factors have been defined to permit assessment of the total risk from Irradiation of only part of the body. The recommended annual dose equivalent for uniform irradiation of the hole body is 50 mSv (5 rem) for radiation workers with the further constraint that no tissue should receive marc than 0.5 Sv, (50 rem) in a year. The only exception to this is the lens of the eye for which an annual limit of 0.3 Sv is proposed, For individual members of the public the annual dose equivalent limits are reduced by a factor of 10 to 5mSv" for the whole body and 50 mSv for tissues. These new dose equivalent limits recommended by the ICRP differ in concept from the previous recommendations which have been in effect for more than 20 years, These latter have been incorporated into national legislation, or codes of practice, in a number of countries, The principal difference to be noted is that, with the exception of the lens of the eye, specific dose limits are no longer recommended for particular tissues. However the basic limit on whole body irradiation is unchanged at 5 rems (50 mSv) per year.

In Nuclear medicine and other fields were radioactive solutions or other unsealed radionuclide source are used radioactive material may enter the human body accidentally by swallowing inhalation of gases aerosols or dusts; or through the

skin or a wound.The ultimate fate of radionuclides entering the body depends on their chemical properties and on their interactions with the natural components of cells and tissues. Some materials may be widely distributed throughout the body while others may localize in specific organs or tissues, for example the radionuclide of iodine concentrate in the thyroid gland while those of calcium and radium deposit predominately in the Skeleton. The fate of radionuclide in the body may be markedly influenced by the chemical form in which it is administered; for instance if iodine ^{131}I is administered as sodium iodide about 30percent enters the thyroid gland from which it is released with an average biological half time of about 70 days in contrast if the same nuclide is administered in the form of iodine ^{131}I labeled Hippuran essentially all the radioactivity will be illuminated though the kidneys within a few hours. The retention of radionuclide in the body may vary widely from complete elimination within a few hours to retention times of many years, depending on the chemical and physical properties of the deposited material l(Donald2004) .

2.7 Previous study :

In the study carried out by Reza et al, (2009), among a total of 3,442,111 imaging procedures associated with radiation exposure that were performed in 655,613 subjects (68.8%) during the 3-year study period, with a mean of 1.2±1.8 procedures per person per year and a median of 0.7 procedures per person per year, they found that: the mean effective dose was 2.4±6.0 mSv per person per year, and the median effective dose was 0.1 mSv per person per year.

Also Thea et al (2008) found that the average whole-body dose at the collar site was 1.7 times the waist readings. And the dose for the technologist performing injections approximately 2 Sv/h. Doses received while scanning ranged from 0.2 to 2 Sv/h. The average dose for a scan depended not only on the administered activity and isotope but also on the amount of patient contact required. Even for high activities, such as patients who had already received therapy, the dose to the technologist was low for patients requiring little assistance.

Clarke E. A et al, (1992) studied Radiation Doses to Staff in a Nuclear Medicine Department and the aim of their study was: to measure external radiation doses and estimate internal radiation doses due to the process of radionuclide injection to staff members working in a nuclear medicine department at Queen Mary Hospital, Hong Kong over a 1-year period; to assess the possible radiation doses to staff members in order to determine whether classification of radiation workers is necessary.

The radiation doses to 4 nuclear medicine physicians, 8 radiographers, and 2 laboratory attendants were measured by digital pocket dosimeters. And their results showed that: the mean annual radiation dose to the physicians was 0.29±0.21 mSv. This was lower than the mean annual radiation dose of 2.07 ±0.97 mSv ($p = 0.017$) to the radiographers and 1.97 ±0.05 mSv ($p = 0.064$) to the laboratory attendants, respectively. The mean radiation dose to the radiographers performing data acquisition and radionuclide injection (1.82 ±1.08

mSv) was not different from that of the radiographers performing data acquisition only (2.53±0.47 mSv) [p = 0.439]. The annual internal radiation dose to individual staff members performing radionuclide injection was estimated to be 0.01 mSv, which can be considered negligible in an estimation of total effective dose.

Pratt et al,(1997) carried out an assessment of radiation dose in nuclear medicine hot lab in Centre for Nuclear Medicine & Ultrasound, Rajshahi Medical College Hospital Bangladesh and the aim of the study was to measure the package surface doses and generator surface doses. The radiation doses in the hot lab were measured by GM and NaI detectors for about 12 months. An increase in the counted rate above background was considered for the study. A constant distance was made in every step. They found that: at the receipt date of the 99Mo/99mTc generator; the surface dose (450±150 μGy/hr) found to be nearly six times higher than the package surface dose (80±20 μGy/hr). The dose rate at the outer surface of the fume-hood glass found to be 80±15 μGy/hr in the 1st day of generator placement, whereas at the 2nd day it was 70±12 μGy/hr; showing a gradual decline in dose rate during 3rd (50±10 μGy/hr), 4th (40±9 μGy/hr), 5th day (30±6 μGy/hr) and 6th day (25±4 μGy/hr).

Taha et al,(2008)studied the Hand Dose in Nuclear Medicine Staff Members in National Cancer Institute, Cairo University and the aim of their study was to measure the hand dose during preparation and injection of radiopharmaceuticals which is useful in the assessment of the extremity doses received by nuclear medicine personnel. Hand radiation doses to the occupational workers that handling ^{99m}Tc-labeled compounds, ^{131}I for diagnostic in nuclear medicine were measured by thermo luminescence dosimetry. A convenient method is to use a TLD ring dosimeter for measuring doses of the diagnostic units of different nuclear medicine facilities. Their doses were reported in millisieverts that accumulated in 4 weeks. The radiation doses to the hands of nuclear medicine staff at the hospitals under study were measured. The maximum expected annual

dose to the extremities appeared to be less than the annual limit (500 mSv/y) because all of these workers are on rotation and do not constantly handle radioactivity throughout the year.

Wy et al, (2002) studied Radiation Doses to Staff in a Nuclear Medicine Department and the aim of their study was: to measure external radiation doses and estimate internal radiation doses due to the process of radionuclide injection to staff members working in a nuclear medicine department at Queen Mary Hospital, Hong Kong over a 1-year period; to assess the possible radiation doses to staff members in order to determine whether classification of radiation workers is necessary.

Methods: Radiation doses to 4 nuclear medicine physicians, 8 radiographers, and 2 laboratory attendants were measured by digital pocket dosimeters. And the Results were: After correction for background natural radiation dose, the mean annual radiation dose to the physicians was 0.29±0.21 mSv. This was lower than the mean annual radiation dose of 2.07 ±0.97 mSv ($p = 0.017$) to the radiographers and 1.97 ±0.05 mSv ($p = 0.064$) to the laboratory attendants, respectively. The mean radiation dose to the radiographers performing data acquisition and radionuclide injection (1.82 ±1.08 mSv) was not different from that of the radiographers performing data acquisition only (2.53±0.47 mSv) [$p = 0.439$]. An empirical formula was applied to compute the possible risk of receiving an internal dose in the process of radionuclide injection. The annual internal radiation dose to individual staff members performing radionuclide injection was estimated to be 0.01 mSv, which can be considered negligible in an estimation of total effective dose. Conclusions: This 1-year study showed that effective radiation doses to nuclear medicine department staff members were within permissible levels, and that the classification of radiation workers is unlikely to be necessary NM in a typical nuclear medicine department in Hong Kong

Ahasan (2004) carried out an assessment of radiation dose in nuclear medicine hot lab in Centre for Nuclear Medicine & Ultrasound, Rajshahi Medical College Hospital Bangladesh and the aim of the study was :to measure the package surface doses and generator surface doses Materials and Methods: The radiation doses in the hot lab were measured by GM and NaI detectors for about 12 months. Were also measured. An increase in the counted rate above background was considered for the study. A constant distance was made in every step. Results: At the receipt date, the 99Mo/99mTc generator surface dose (450±150 μGy/hr) found to be nearly six times higher than the package surface dose (80±20 μGy/hr). The dose rate at the outer surface of the fume-hood glass found to be 80±15 μGy/hr in the 1st day of generator placement, whereas at the 2nd day it was 70±12 μGy/hr; showing a gradual decline in dose rate during 3rd (50±10 μGy/hr), 4th (40±9 μGy/hr), 5th day (30±6 μGy/hr) and 6th day (25±4 μGy/hr). Conclusion: In the 1st day of a generator storing in the hot lab, the dose rate found to be 3-4 times higher than the 6th days. The dose rate at various places indicated poor performance of the fume-hood glass. The study emphasizes on the need of growing awareness among all the radiation workers and encouraging the safe working practices in nuclear medicine. Iran. J. Radiat. Res., 2004; 2 (2): 75-7
The doses for staff handling ^{99m}Tc-labeled compounds and ^{131}I for diagnosis cases has been shown in Table (2.1) which implies the radiation doses to the hands of the diagnostic workers. As expected, the radiation doses to the hands of the nurses involved in intravenous administration were observed to be higher than those for the radio-pharmacy staff whom are prepared the isotopes.

In their study, relatively higher finger doses were observed for technician whom prepared Radiopharmaceuticals in hot laboratory, that gives indicator that the fume hood may be need to Clean daily and check its negative pressure. The equivalent doses during injection were lower than those doses during preparation due to using lead syringe during injection. The equivalent doses to extremities are lower than of the extremity dose limit during the measured periods, (The dose

limit is 500 mSv/year) i.e. the maximum dose received is lower than the dose limit. Their practice revealed that: exposure is not likely to exceed the annual limit of 500 mSv for any one of the staff members under normal prevailing work circumstances (IAEA).Vienna (1996).(see appendix (1)).

Chapter Three

Methods and materials

3.1 Material and Method :

3.1 Materials:

- **GM survey meters:-** The GM counter used in this study has the ability to accurately measure the exposure at different energies of radiation with an enhancement of designed electronics that made it suitable for survey meters Fig (3.1)

Figure 3.1GM Survey Meter

- **Generator:-**
  - Generatorsare units that contain a radioactive, The used generator in this study was the technetium-99m (^{99m}Tc) generator with :
    an activity of 15 GBq , code: MTC-4 , calibration date ,
    batch NO 81/15 ,Expiration date : 2015,Generator No 81/13 shown in Fig 3.2. It consists of a heavily shielded column with molybdenum-99 (99Mo; parent), bound to the alumina of the column. The ^{99m}Tc (daughter) is “milked” (eluted) by drawing sterile saline through the column into the vacuum vial. The parent 99Mo (small grey circles) remains on the column, but the daughter ^{99m}Tc is washed away in the saline .

Figure 3.2 Generators Tc-99 m

3.2 Method:

3.2.1 Study duration:

This study was from June 2013 to October 2015

3.2.2 The study place:

This study conduct in Radiation & Isotope center of Khartoum (RICK)

3.2.3 The Hot Lab :

The hot lab of NM department at RICK has been divided into sets of points, where the technologists and physicists may presence based on the Figure (3.4). The measurement of exposure level has been carried out in those points as daily following the generator decay using GM survey detector.

the points from 2 – 8 represented the places where the staff wandering, and the exposure laid out in mSv/h.

point 1 it place 20cm from the generator and take the measurer by the GM ,and point 2 it about 50 cm from generator , point 3,4 it far 30cm from point 3 ,point 5 far 50cm from point 3, point 6 far 50 from point 5 and point 6,7 about 30 cm from point 6.

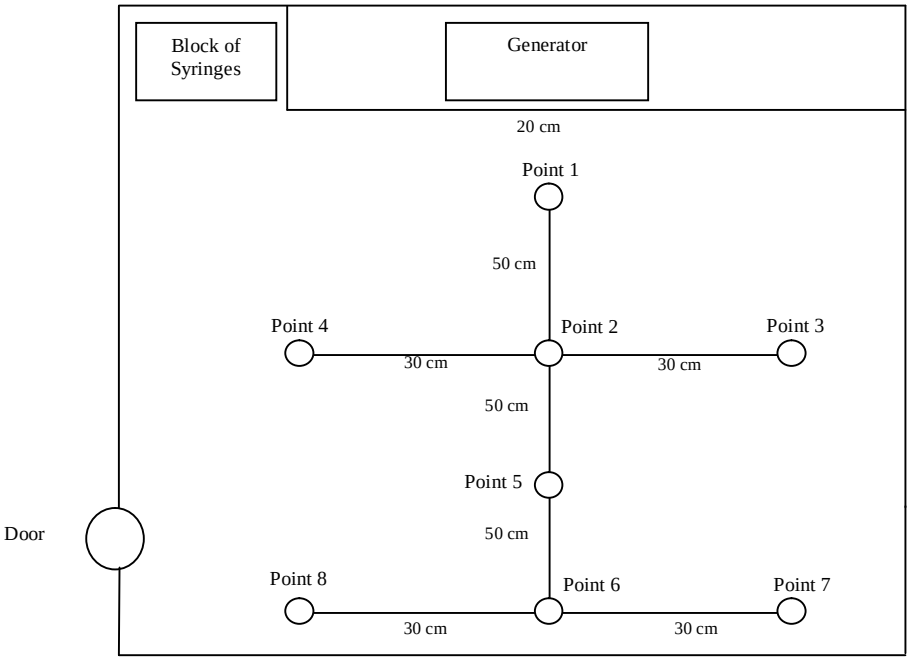

Figure 3.3 shows the area of hot lab and the points of measurements in meter.

3-3Analysis of data:

All dose parameters will registered from the display monitor in survey meter then used as input to the Microsoft excel and SPSS software for analysis.

Chapter Four

4.1 Results

The following chapter is dealing with the result of radiation exposure doses measurements from different 8 point in versus distance from generator in nuclear medicine department (hot lap) .

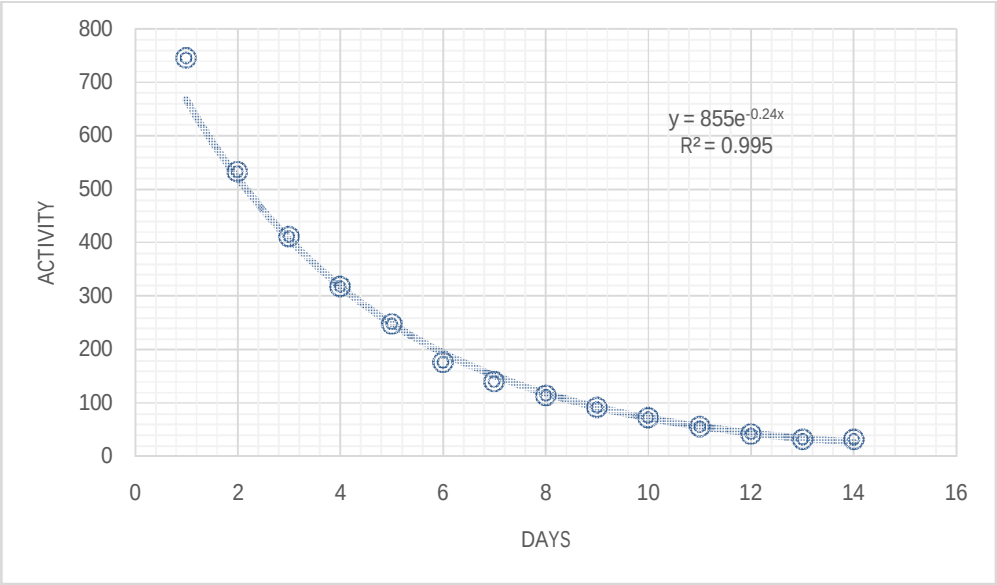

Figure 4.1 shows the daily Technetium generator activity (time activity curve)

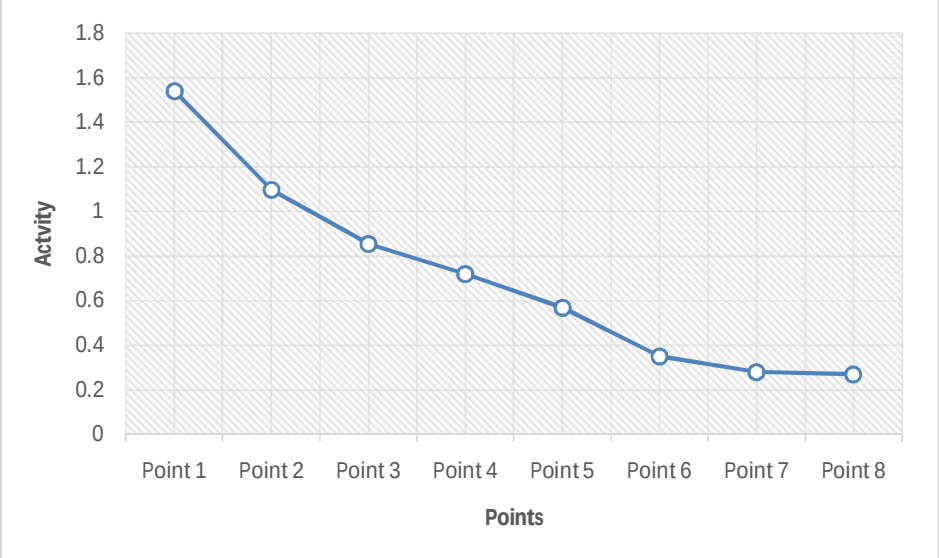

Figure 4.2 shows the decay of the activity points

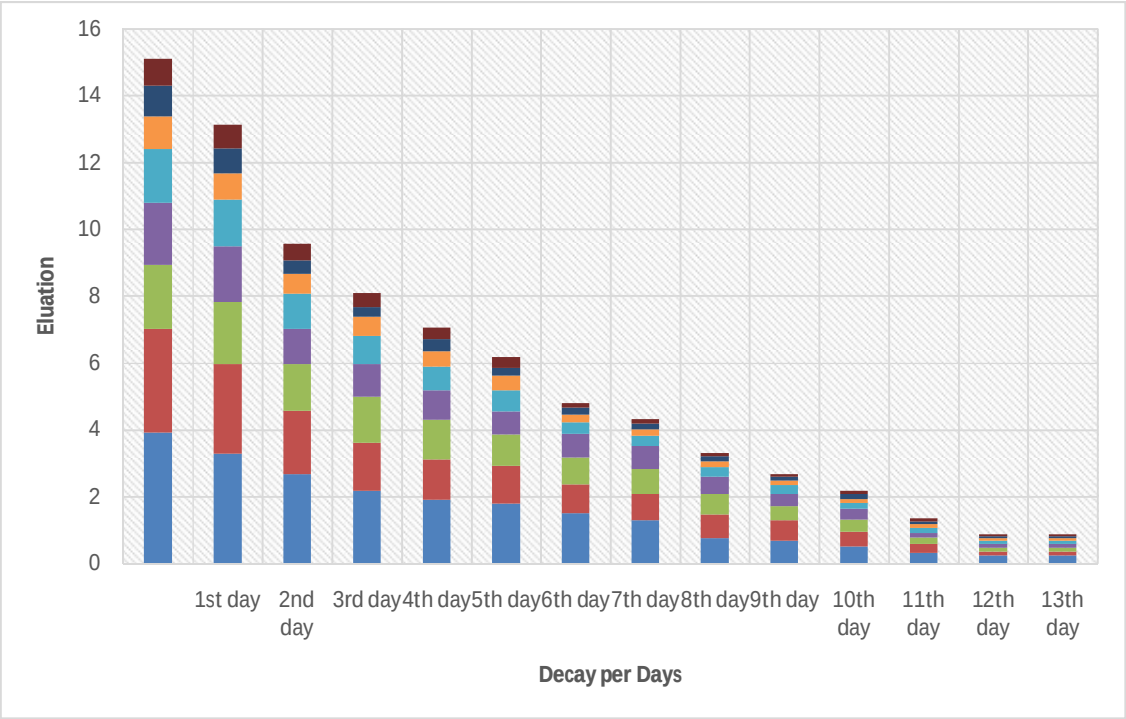

Figure 4.3 show reading of Activity per points

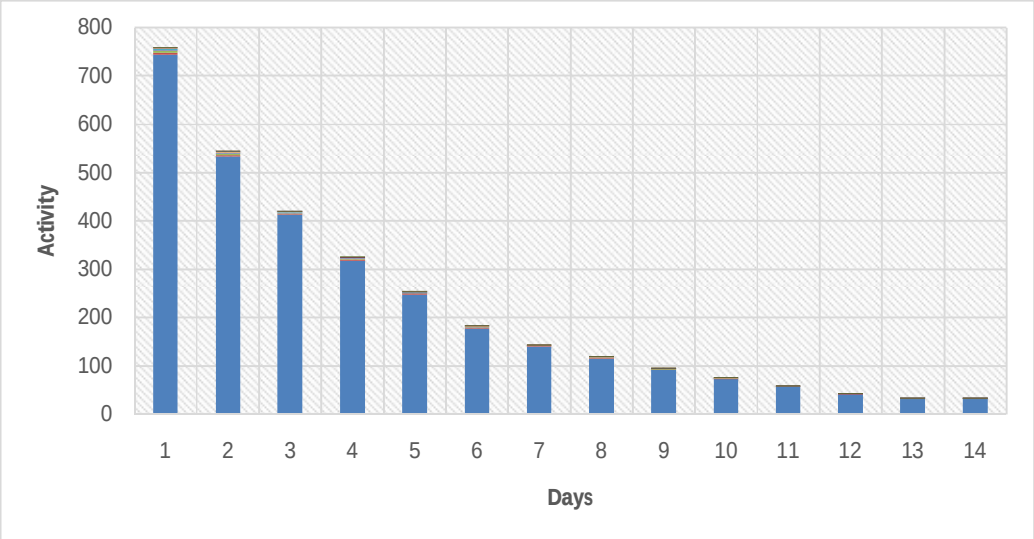

Figure 4.4 show decays of Activity per days

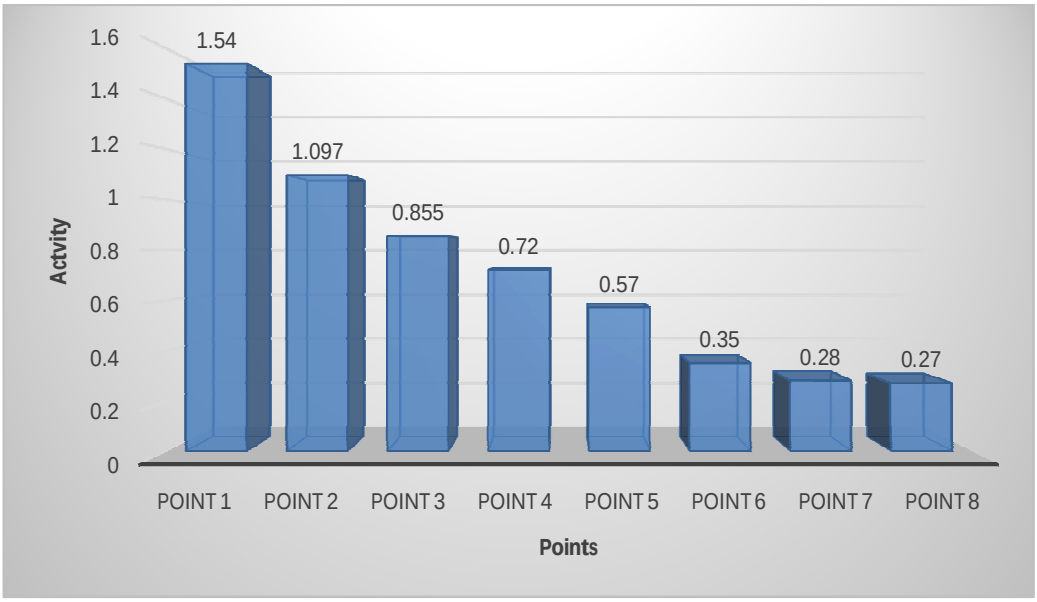

Figure 4.5 show decay of eluation per days

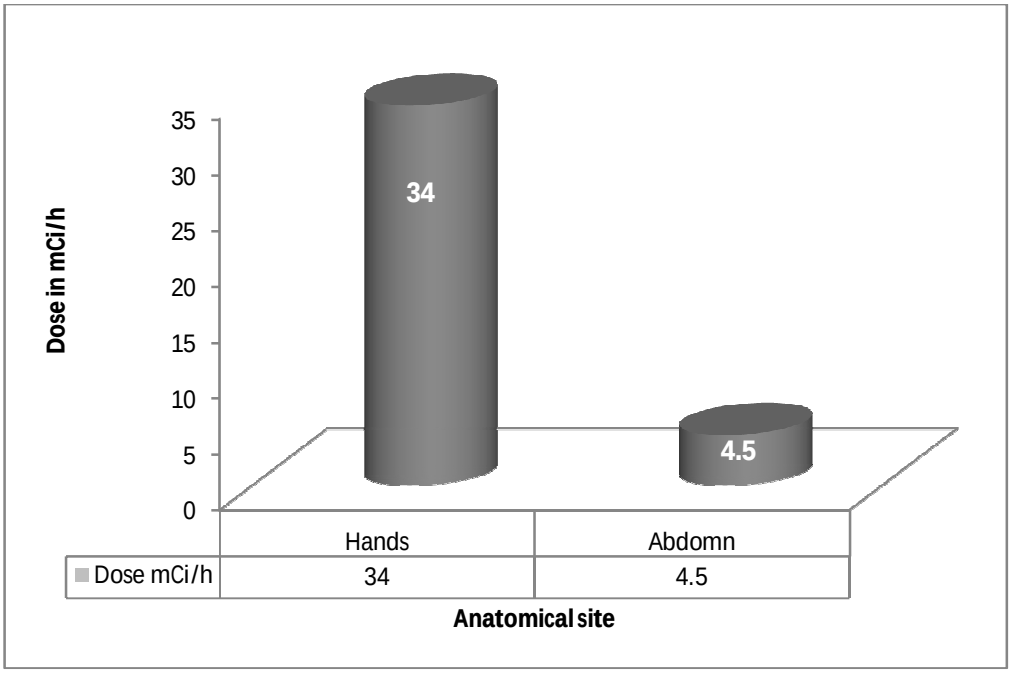

Figure 4.6 shows the dose received by the abdomen and the hands during working days of technetium generator

Chapter Five

Discussion , Conclusion and Recommendation

5.1 Discussion :

the daily Technetium generator activity (time activity curve). It reveals that the Tc-generator has an activity decaying relative to days in a form of exponential inversely relationship that fits in equation: $y = 855e^{-0.25x}$, which is significant as $R^2 = 0.995$.

the accumulated dose in mCi/h per each point during the working days of Technetium generator. It reveals that: the dose at points of measurement have been decreased as the point of measurement set far from the generator and the relationship between the dose and the distance fitted in an exponential equation of the following form: $y = 1.926e^{-0.261x}$, the correlation so significant as $R^2 = 0.979$, such result deduced that: the dose just behind the hot zone of generator was 24.3 mCi/h, and the dose at the vicinity of generator by 1 meter was 12.97 mCi/h. same result have been obtained by Jankowski et al (2003). And based on the measured dose; the expected dose received by the technologists at the most nearest point to the zone of hot lab will not exceed the maximum permissible dose due to presence of shield.

the dose received by the abdomen and the hands during working days of technetium generator. It reveals that the technologist's hands received radiation dose of 34 mCi/h and the abdomen received a dose of 4.5 mCi/h.

this study is agreed with the previous study carried Majowska et al (2006) which showed that: the hands received a high dose of 108.2% higher than the extremities permissible dose. So that in this study the expose during elution at nuclear medicine department in (RICK) showed that the technologist can be expose to radiation dose even behind the lead shield surrounding the technetium generator far to tow meter

Measurement of exposure level received by technologist hand and abdomen relative to distance at hot lap and cycle of waste management in different point during thirteen days .

Hence in the nuclear medicine department it needs to be certified for protection . the exposure is receive from hot lap during technetium generator elution by using Geiger muller survey to detected the radiation received technologist in 8 points.

This study is provide the hazard and exposure was received to technologist . while the hazard is decline relative to distance of generator and source of elution from technologist .whole body can expose while we foucse to the hand abdomen we found the hand receive radiation dose of 34 mCi/ and abdomen received a dose of 4.5 mCi/h.

5.2 Conclusion:

In conclusion of this study the successful achievement of the objectives related to technetium generator time activity curve i.e. decay process, dose measurement received by the technologist hands, the exposure level relative to distance at hot lab and the cycle of waste management, the study conclusion notify that:

The technetium generator has a time activity curve that fits the exponential equation of the following form: y = 855.e-0.247x, where x refers to days and y refers to the relative activity ,The exposure level at different points distal from the generator could be calculated based on the following equation y = 1.93e-0.261x, where x refers to the point and y refers to the accumulative dose during generator working days and The technologist's hands received radiation dose of 34 (mCi/h) and the abdomen received a dose of 4.5 (mCi/h).

5.3 Recommendation:

The worth points to be recommended in relation to the scored objectives are:

- Determination of the dose received by hands during elusion and the estimation of relative risks as radiation sickening.
- Carry out same study at other radiation centers in Sudan like Aljazeera center and Shandi center.
- Considering the high radioactive waste used in medical field.
- Such survey study could be carried out for environment and some habitual cities.

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Appendixes

Appendix (1)

Table: Hand Doses for Group diagnostic workers Involved in handling ^{131}I and ^{99m}Tc-Labeled Compounds in two of the nuclear medicine facilities.

Worker	Activity handled/month (GBq)	Dose (mSv) accumulated in 1 month
Radiopharmacy laboratory		
1	29.6	2.40
2	37	0.99
3	29.6	1.20
4	44.4	1.48
Isotopes injection		
5	29.6	0.5
6	22.2	0.12
Imaginig technicians		
7		0.96
8		0.60
9		0.44
10		1.28

Appendix (2)

Table: Basic table measurement of TC^{99m} generator elusion from first day to thirteenth day

Day of ellusion	Activity	Volume (ml)	Time of elusion (pm)	Doses at different points (msv/hr)							
				Point 1	Point 2	Point 3	Point 4	Point 5	Point 6	Point 7	Point 8
1st day	745.4	9	8: 30	3.94	3.10	1.91	1.86	1.6	0.98	0.91	0.82
2nd day	533	9	8: 50	3.3	2.7	1.84	1.66	1.41	0.78	0.74	0.71
3rd day	412	9	9: 00	2.7	1.89	1.38	1.07	1.05	0.58	0.41	0.51
4th day	317.8	9	9: 00	2.2	1.43	1.37	0.99	0.83	0.57	0.30	0.42
5th day	248	9	9: 20	1.93	1.21	1.18	0.88	0.71	0.46	0.36	0.34
6th day	177	8	9: 00	1.81	1.13	0.93	0.7	0.63	0.43	0.23	0.33
7th day	140	9	8: 30	1.53	0.85	0.81	0.71	0.34	0.23	0.21	0.15
8th day	115	9	9: 00	1.32	0.79	0.73	0.68	0.31	0.21	0.17	0.13
9th day	92.2	5	9: 15	0.77	0.70	0.63	0.51	0.28	0.18	0.15	0.11
10th day	73	10	9: 00	0.71	0.61	0.41	0.38	0.24	0.16	0.10	0.08
11th day	56.3	5	8: 30	0.53	0.44	0.37	0.31	0.19	0.11	0.13	0.11
12th day	42.59	5	8: 30	0.35	0.27	0.17	0.14	0.15	0.11	0.08	0.10
13th day	32.6	5	9: 00	0.25	0.12	0.12	0.11	0.10	0.07	0.06	0.05

Appendix (3)