Does Measurements for Internal Medical Treatments

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ABSTRACT

Internal dosimetry is a major component of nuclear safety for the the approximately 100, 000 radiation workers at DOE radiological or nuclear facilities. Workers who handle nuclear materials or who are involved in nuclear waste management are potentially at risk of inadvertent intakes of radioactive material . DOE policy and associated radiological control programs for limiting internal effective doses are based on containment of radioactive material to ensure (to the extent reasonably achievable) that radionuclides from work at radiological lr nuclear facilities are not taken into the body. Most significant occupations intake of radionuclides, i.e ., individuals receiving doses approaching or exceeding a limit, occur as the result of contamination incidents associated with either the inadvertent release of radioactive material in the workplace or the unplanned loss of containment .

Keywords: Internal , Medical Treatments, radioactive

Introduction

1-1:Internal dosimetry

Internal dosimetry is the science and art of internal ionizing radiation dose assessment due to radionuclides incorporated inside the human body[1]. Radionuclides deposited within a body will irradiate tissues and organs and give rise to committed dose until they are excreted from the body or the radionuclide is completely decayed. The internal doses for workers or members of the public

exposed to the intake of radioactive particulates can be estimated using bioassay data such as lung and body counter measurements, urine or faecal radioisotope concentration, etc. The International Commission on Radiological Protection (ICRP) biokinetic models are applied to establish a relationship between the individual intake and the bioassay measurements, and then to infer the internal dose.

Internal dosimetry is the method used to convert the amount of ionizing radiation deposited in tissue to its effect in tissue, which is influenced by the "damage potential" of the radiation type (e.g., energy, size, charge, half-life, etc.), the administered dose, and the dose rate [2]. The quantities obtained from dosimetry calculations are fundamental to estimating radiation protection, risk assessment, diagnostic dose estimates, and treatment planning [3].

Internal dosimetry in drug development is primarily used from two perspectives [4]. For early-phase I or phase I clinical trials, dosimetry is performed after radiopharmaceutical administration to provide standard diagnostic procedural dose estimates and define dose-limiting organs in a limited number of healthy volunteers. A second type of dosimetry is used to guide treatment and thus performed prior to the therapeutic drug administration for all patients undergoing treatment. Organ-specific (target and source organs) and total effective dose equivalent dosimetry estimates are calculated in diagnostic studies, whereas in treatment planning, dosimetry estimates are focused regionally that correspond to the treatment area [5]. Since radiation dose calculations include many factors and are lengthy, tedious, and error-prone when performed manually, the FDA strongly recommends using dosimetry code software when submitting dosimetry estimates for new radiopharmaceuticals that are or will be administered to humans for experimental or clinical use [6]. Dosimetry software such as Medical Internal Radiation Dose (MIR Dose) [75] or Organ Level Internal Dose Assessment Exponential Modeling . successor to MIR Dose, [7] offer users a variety of phantoms that permit calculating radiation doses for individuals at various ages and sizes, and for women at different stages of pregnancy.

1-2- Radiation sources:

occurs when energy is emitted by a source, then travels through a medium, such as air, until it is absorbed by matter.

Radiation can be described as being one of two basic types :-

non ionizing and ionizing.

There are many types of ionizing radiation. The following are some of the relevant ones :-

Alpha radiation :-

Alpha radiation consists of two protons and two neutrons; since they have no electrons, they carry a positive charge. Due to their size and charge, alpha particles are barely able to penetrate skin and can be stopped completely by a sheet of paper.

Beta radiation :-

Beta radiation consists of fast-moving electrons ejected from the nucleus of an atom. Beta radiation has a negative charge and is about 1/7000th the size of an alpha particle, so it is more penetrating. However, it can still be stopped by a small amount of shielding, such as a sheet of plastic.

Gamma radiation :-

Gamma radiation is a very penetrating type of radiation. It is usually emitted immediately after the ejection of an alpha or beta particle from the nucleus of an atom. Because it has no mass or charge, it can pass through the human body, but it is absorbed by denser materials, such as concrete or lead.

X-rays :-

X-rays are a form of radiation similar to gamma radiation, but they are produced mainly by artificial means rather than from radioactive substances

.Neutron radiation :-

Neutron radiation occurs when neutrons are ejected from the nucleus by nuclear fission and other processes. The nuclear chain reaction is an example of nuclear fission, where a neutron being ejected from one fissioned atom causes another atom to fission, ejecting more neutrons. Unlike other radiations, neutron radiation is absorbed by materials with lots of hydrogen atoms, like paraffin wax and plastics.

Artificial sources of radiation :-

Atmospheric testing :-

The atmospheric testing of atomic weapons from the end of the Second World War until as late as 1980 released radioactive material, called fallout, into the air. As the fallout settled to the ground, it was incorporated into the environment. Much of the fallout had short half-lives and no longer exists, but some continues to decay to this day. People and the environment receive smaller and smaller doses from the fallout every year.

Medical sources :-

Radiation has many uses in medicine. The most well known use is in X-ray machines, which use radiation to find broken bones and diagnose disease. X-ray machines are regulated by Health Canada and provincial authorities. Another example is nuclear medicine, which uses radioactive isotopes to diagnose and treat diseases such as cancer. These applications of nuclear medicine, as well as the related equipment, are regulated by the CNSC. The CNSC also licenses those reactors and particle accelerators that produce isotopes destined for medical and industrial applications.

This image shows examples of medical sources of radiation including an x-ray, CT scan, nuclear medicine, and a particle accelerator that produce isotopes.

Industrial sources :-

Radiation has a variety of industrial uses that ranges from nuclear gauges used to build roads to density gauges that measure the flow of material through pipes in factories. It is also used in smoke detectors and some glow-in-the dark exit signs, and to estimate reserves in oil fields. Radiation is also used for sterilization in which large, heavily shielded irradiators are used. All of these uses are licensed by the CNSC.

This image shows examples of industrial sources of radiation including nuclear gauges, a smoke detector, and glow in the dark exit sign.

Nuclear fuel cycle :-

Nuclear power plants (NPPs) use uranium to drive a chain reaction that produces steam, which in turn drives turbines to produce electricity. As part of their normal activities, NPPs release regulated levels of radioactive material which can expose people to low doses of radiation.

Similarly, uranium mines, fuel fabrication plants and radioactive waste facilities release some radioactivity that contributes to the dose of the public.

Sources of ionizing radiation :-

People are constantly exposed to small amounts of ionizing radiation from the environment as they carry out their normal daily activities; this is known as background radiation. We are also exposed through some medical treatments and through activities involving radioactive material.

Natural background radiation :-

Radiation has always been present and is all around us. Life has evolved in a world containing significant levels of ionizing radiation. Our bodies are adapted to it.

The following section outlines sources of natural background radiation. For information on dose levels from these sources, visit the Radiation Doses page and fact sheet on natural background radiation.

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) identifies four major sources of public exposure to natural radiation :-

cosmic radiation

terrestrial radiation

inhalation

ingestion

Radiation Dose Units:-

Radiation absorbed dose and effective dose in the international system of units (SI system) for radiation measurement uses "gray" (G y) and "sievert" (Sv), respectively.

In the United States, radiation absorbed dose, effective dose, and exposure are sometimes measured and stated in units called rad, rem, or roentgen (R).

For practical purposes with gamma and x rays, these units of measure for exposure or dose are considered equal.

This exposure can be from an external source irradiating the whole body, an extremity, or other organ or tissue resulting in an external radiation dose. Alternately, internally deposited radioactive material may cause an internal radiation dose to the whole body, an organ, or a tissue.

Smaller fractions of these measured quantities often have a prefix, such as milli (m) that means 1/1,000. For example, 1 sievert = 1,000 mSv. Micro (μ) means 1/1,000,000. So, 1,000,000 μSv = 1 Sv, or 10 μSv = 0.000010 Sv.

Conversions from the SI units to older units are as follows:

1 Gy = 100 rad

1 mGy = 100 mrad

1 Sv = 100 rem

1 mSv = 100 mrem

With radiation counting systems, radioactive transformation events can be measured in units of "disintegrations per second" (dps) and, because instruments are not 100 percent efficient, "counts per second" (cps) .

Radiation Quantities and Units :-

The unique radiation exposure conditions that exist in computed tomography (CT), during which thin slices of the patient are irradiated by a narrow, fan-shaped beam of x rays emitted from the x-ray tube during its rotation around the patient, have required the use of special dosimetry techniques to characterize the radiation doses to patients and to monitor CT system performance. This section describes the basic dosimetry quantities used to indicate patient doses during CT.

Absorbed dose - The fundamental quantity for describing the effects of radiation in a tissue or organ is the absorbed dose. Absorbed dose is the energy deposited in a small volume of matter (tissue) by the radiation beam passing through the matter divided by the mass of the matter. Absorbed dose is thus measured in terms of energy deposited per unit mass of material. Absorbed dose is measured in joules/kilogram, and a quantity of 1 joule/kilogram has the special unit of gray (Gy) in the International System of quantities and units. (In terms of the older system of radiation quantities and units previously used, 1 Gy equals 100 rad, or 1mGy equals 0.1 rad.)

Equivalent dose - The biological effects of an absorbed dose of a given magnitude are dependent on the type of radiation delivering the energy (i.e., whether the radiation is from x rays, gamma rays, electrons (beta rays), alpha particles, neutrons, or other particulate radiation) and the amount of radiation absorbed. This variation in effect is due to the differences in the manner in which the different types of radiation interact with tissue.

The variation in the magnitude of the biological effects due to different types of radiation is described by the "radiation weighting factor" for the specific radiation type. The radiation weighting factor is a dimensionless constant, the value of which depends on the type of radiation. Thus the absorbed dose (in Gy) averaged over an entire organ and multiplied by a dimensionless factor, the radiation weighting factor, gives the equivalent dose. The unit for the quantity equivalent dose is the sievert (Sv). Thus, the relation is :

equivalent dose (in Sv) = absorbed dose (in Gy) x radiation weighting factor

In the older system of units, equivalent dose was described by the unit rem and 1 Sv equals 100 rem or 1 mSv equals 0.1 rem.

For x rays of the energy encountered in CT, the radiation weighting factor is equal to 1.0. Thus, for CT, the absorbed dose in a tissue, in Gy, is equal to the equivalent dose in Sv.

Effective dose - The risk of cancer induction from an equivalent dose depends on the organ receiving the dose. A method is required to permit comparison of the risks when different organs are irradiated. The quantity "effective dose" is used for this purpose. The effective dose is calculated by determining the equivalent dose to each organ irradiated and then multiplying this equivalent dose by a tissue-specific weighting factor for each organ or tissue type. This tissue- or organ-specific weighting factor accounts for the variations in the risk of cancer induction or other adverse effects for the specific organ. These products of equivalent dose and tissue

weighting factor are then summed over all the irradiated organs to calculate the "effective dose." (Note that effective dose is a calculated, not measured quantity) The effective dose is, by definition, an estimate of the uniform, whole-body equivalent dose that would produce the same level of risk for adverse effects that results from the non-uniform partial body irradiation. The unit for the effective dose is also the sievert (Sv).

Quantities specific to CT - A number of special dose quantities have been developed to characterize the doses associated with CT. It is beyond the scope of this discussion to describe these unique dose descriptors. They include the Computed Tomography Dose Index, referred to as the CTDI, the "weighted" CTDI (CTDIW), the "volume" CTDI (CTDIVOL), the "multiple scan average dose" (MSAD), and the "dose-length product" (DLP).

Aim of the project :-

To become familiar with the physical principle upon which internal dosimetry calculations are based , and to gain experience with the classical (ICRP) and MIRD systems of internal dose calculation .

2- Theory :-

All methods of internal dosimetry have as their aim the estimation of the absorbed dose to an individual who has had an intake of radioactive material. To arrive at the dose estimate, the physicist must work through three generalized steps.

First , an estimate of the total activity and its distribution in body organs must be made. . A whole body counter which externally records the emissions of internally deposited radionuclides , is another common technique for obtaining the body burden .

The second step in arriving at the dose estimate is development of a model which describes behavior of the deposited radioisotopes in the body as a function of time .

The third and final step is to calculate the doses to the various body organs as a consequence of the deposited activity. This requires a knowledge of the body burden (step 1) , the time behavior (step 2) and the decay scheme of the radionuclide . In addition , a model must be generated which relates the energy emitted by the radionuclide to the energy actually deposited in various body organs . Step 2 and 3 will now be discussed in greater detail .

The behavior with time of deposited radioisotopes in the body depends on a number of factors . Some of the most important include :-

A) chemical elements involved

B) chemical form

C) physical form

D) excretion rate

E) physical half - life

The body usually doesn't distinguish a radioactive isotope from a stable istope of the same element. Thus radioiodine will tend to concentrate in the thyroid gland and radio strontium in the bone. The body does distinguish chemical form , however. Compounds which are tagged with a radionuclide will be metabolized as the compound , not as free atoms or ions of the tagging radionuclide. Occasionally, the physical form determines de- position and excretion. Large tagged macroaggregates in injected intravenously will become trapped in small capillaries (eg :- in the lung) due to their diameter exceeding the size of the capillary construction.

The last two factors affecting the time behavior of deposited radionuclide have to do with the removal of activity. The excretion rate is the rate that the body removes the material through various normal or induced physiological processes. In case of large uptake of radionuclides , drugs are sometimes administered to speed the clearance of the radionuclide from the body . In addition to the biological removal mechanisms, the radio- isotopes are also undergoing physical decay. The most common model employed assume that both biological removal and physical decay are exponential in time . The net effect of both of these independent processes is that the activity decreases as the product of the two exponentials . Hence , the body burden , A , as a function of time is expressed in terms of the original deposited activity A_O as

A(t) = A_O $(e^{-\lambda bt})$ $(e^{-\lambda pt})$ (1)

Where :-

λb = biological decay constant

λp = physical decay constant .

If we define an effective decay constant , λe , as the sum of the physical and biological decay constants , then equation 12 .1 becomes:-

A(t) =A_O $e^{-\lambda et}$ (2)

The effective decay constant corresponds to an effective half life ,

$$T_{eff} = \frac{T_P \cdot T_b}{T_P + T_b} \quad (3)$$

Equation 12 .3 can be used to calculate effective half- lives if physical and biological half - lives are known . The biological half - life is frequently difficult to determine since it depends on so many variables. Values for many radionuclides will be given below in connection with the two dosimetry systems discussed .

In order to complete the dose calculation , physical data on the decay sheme is needed and a model is required for calculating the actual absorbed doses to organs relative to emitted energy. The energy released as particulate radiation is assumed to be totally deposited in the organ in which it is released . For alpha emitter , the

decay energy is given to a single particle . The average energy per disintegration is thus :-

E= $E_\alpha f$ (4)

Where :-

Eα = alpha energy in MeV

f= fraction of decays leading to alpha emission .

In beta decay , however , the energy is shared between the electron and a neutrino . The average energy per disintegration , E , is approximately 1/3 of the maximum beta energy.

E=$\frac{1}{3} E_{max} . f$ (5)

Two somewhat different approaches to the problem of photon dosimetry have evolved over the years . The classical method employs energy absorption coefficients . A consistent set of dosimetry data using this approach has been published by the international com- mission on Radiological Protection as ICRP Publication 2 . The MIRD Committee (Medical Internal Radiation Dose) has developed a system using the absored fraction concept for photon dosimetry . It was originally designed for the nuclear medicine community, but recent expansion of the data tables to cover a wide variety of radionuclides has made it useful in health physics applications as well .

Classical Method :-

For photon radiation, the average energy deposited over the volume of the source organ is given by

E = $E_O f(1 - e^{-\mu x})$ (6)

Where :-

E_O is the gamma ray energy

f is the fraction of the decays leading to y emission

μ is the total absorption coefficient minus the compton scattering contribution

And x is the effective radius of the organ .

By applying the conversion factors relating Mev , ergs and rads it can be shown that the dose to an organ due to the total decay (i.e , the dose commitment) of A○ microcuries is initially deposited of a radionuclide is :-

D= $73.8\frac{A_O}{m}\ E_{eff}T_{eff}(rems)$ (7)

Where :- m = organ mass in g

E_{eff}= effective energy released per disintegration in M_{ev} and T_{eff}= effective half - life in days

In the ICRP tables , the effective energy is tabulated as :-

E_{eff}= { EF (RBE)n (8)

In this system , E is the average energy per disintegration and is computed for each different emission from equation (4), (5)and (6) . F is a correction factor which increases the effective energy if the corresponding radiation emission leads to a radio-active daughter atom , RBE has the same value as the quality factor and n has a value of 5 for bone seeking radionuclides or 1 for all other cases .

In order to use equation (7) for practical calculations, the effective half - life of the nuclide in various body organs must be known . Average values are available for many nuclide in ICRP Publication 2 . A few values are reproduced the organ masses of the ICRP Standered Man . Finally , the activity deposited in a particular organ must be known . In many accident situations , the total intake in the whole body can be calculated or measured. ICRP has defined a parameter f2 as the ratio of the activity of a radionuclide in a particular organ to the total body burden . Tabu-lated values of f2 . The value for A_O for a particular organ is then f2× Total Activity Deposited . Finally a few ICRP calculated values for E_{eff} a

Organ doses are then simply calculated as a product of tabulated factors. Unfortunately the tabulated values of the metabolic parameters T_{eff} and f2 apply only to a highly ideal- ized standard man model Variations due to different chemical or physical forms of the radionuclide or departures from standard man physiology are not taken into account. If actual measurements of Teff and f2 can be made in a particular case, they should certainly be used in place of tabulated values for dose calculations

The kidney weight is 300 (gm)

Sample problems :-

A technician was working in a glove box in a chemical operation involving separation of fission products. A chemical explosion resulted in which the employee was exposed for a short time to Cs-137 in soluble form. Later bioassays and whole body

counting determined the person had ingested about 200 μCi of the isotope. Calculate the dose commitment to the total body and the kidney.

Equation :-

Ao =200

m= 300

$Eeff$ = = 0.664

$Teff$ = 20

D = 73.8 $\frac{Ao}{m}$ $Eeff$ $Teff$

D = 73.8 ($\frac{200}{300}$) (0.664) (20)

D= 653.376 ram

Discussions :-

Internal dosimetry is " ... the Scientific methodology used to measure, calculate, estimate, assay , predict, and otherwise quantify the irradiative energy absorbed by the ionization and excitation of atoms in human tissues as a result of the mission of energetic radiation by internally deposited radionuclided . Radiation protection requirements for U.S. Department of Energy (DOE) and DOE contractor employees are given in DOE's Occupational Radiation Protection, Title 10 , Code of Federal Regulations, Part 835(DOE 2007a) . On June 8 , 2007 , DOE published an amendment to 10 CFR 835 which, in part , updated DOE's requirements for assessing and recording internal dosimetry results. The update was made to make DOE's system consistent with more recent national and international consensus standards. In this Technical Standard this regulation will be referred to as "10 CFR 835 ".

Further , the Radiological Control Standard ("Radcon Standard ;" DOE 2008a) contains provisions that apply to many contractors by virtue of being included in their contract . DOE's 10 CFR 835 and Radcon Standard require monitoring of the workplace , and monitoring of radiation workers who , under typical conditions, are likely to receive 0.1 rems (0.001 Sv) or more committed effective dose , and/or 5 rems (0.05 Sv) committed equivalent dose to any organ or tissue, from all Occupational radionuclide intakes in a year . The regulation 10 CFR 835 also requires that measurements of internal radionuclides and the assessments of committed effective dose resulting from intakes of radionuclides be recorded , reported , and archived .

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