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GOVERNMENT OF INDIA

AERB SAFETY GUIDE

Monitoring and Assessment of Occupational Exposure Due to Intake of Radionuclides

ATOMIC ENERGY REGULATORY BOARD

AERB SAFETY GUIDE NO. AERB/NRF/SG/RP-1

Monitoring and Assessment of Occupational Exposure Due to Intake of Radionuclides

Atomic Energy Regulatory Board

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FOREWORD

Activities concerning establishment and utilisation of nuclear facilities and use of radioactive sources are to be carried out in India in accordance with the provisions of the Atomic Energy Act 1962. In pursuance of the objective of ensuring safety of members of the public and occupational workers, as well as protection of environment, the Atomic Energy Regulatory Board (AERB) has been entrusted with the responsibility of laying down safety standards and enforcing rules and regulations for such activities under its purview. The Board, therefore, has undertaken a programme of developing safety codes, safety standards and related guides and manuals for the purpose. While some of these documents cover aspects such as siting, design, construction, operation, quality assurance and decommissioning of nuclear and radiation facilities, other documents cover regulatory aspects of these facilities.

Safety codes and safety standards are formulated on the basis of nationally and internationally accepted safety criteria for design, construction and operation of specific equipment, structures, systems and components of nuclear and radiation facilities. Safety codes establish the safety objectives and set requirements that shall be fulfilled to provide adequate assurance for safety. Safety guides elaborate various requirements and furnish approaches for their implementation. Safety manuals deal with specific topics and contain detailed scientific and technical information on the subject. These documents are prepared by experts in the relevant fields and are extensively reviewed by advisory committees of the Board before they are published. The documents are revised when necessary, in the light of experience and feedback from users as well as new developments in the field.

The radiation dose to workers due to internal uptake of radionuclides arising from the operation of nuclear fuel cycle and radiation facilities including nuclear power plants must be assessed, controlled and documented. This safety guide provides guidance for the dose assessment of radiation workers due to internal uptake of radionuclides. In drafting of this Guide, information contained in relevant documents published by the International Atomic Energy Agency (IAEA) under the Basic Safety Standards, Recommendations of the International Commission on Radiological Protection (ICRP) and other international publications have been extensively used. Annexure and references are included to provide information that might be helpful to the user.

For aspects not covered in this guide, applicable national and international standards, codes and guides acceptable to AERB should be followed. External dose assessment is not considered. These aspects are covered in the relevant safety guides.

This Guide has been prepared by specialists in the field drawn from the Atomic Energy Regulatory Board, Bhabha Atomic Research Centre, Indira Gandhi Centre for Atomic Research, Nuclear Power Corporation of India Ltd. and other consultants. It has been reviewed by the relevant AERB Advisory Committees on Codes and Guides and the Advisory Committee on Nuclear and Radiation Safety (ACNRS).

AERB wishes to thank all individuals and organizations who have prepared and reviewed the draft and helped in its finalization. The list of persons, who have participated in this task, along with their affiliations, is included for information.

(G. Nageswara Rao)
Chairman, AERB

DEFINITIONS

Absorbed Dose

The fundamental dosimetric quantity D is defined as:

$$D = dE / dm$$

where, 'dE' is the mean energy imparted by ionising radiation to the matter in a volume element and 'dm' is the mass of matter in the volume element. The energy can be averaged over any defined volume, the average dose being equal to the total energy imparted in the volume divided by the mass in the volume. The SI unit of absorbed dose is joule/kg ($J.kg^{-1}$), termed the gray (Gy).

Activity

The quantity 'A' for an amount of radionuclide in a given energy state at a given time is defined as:

$$A(t) = dN/dt$$

where 'dN' is the expectation value of the number of spontaneous nuclear transformations from the given energy state in a time interval 'dt'. The SI unit of activity is the reciprocal of second (s^{-1}), termed the Becquerel (Bq).

Activity Median Aerodynamic Diameter (AMAD)

The value of aerodynamic diameter such that 50% of the airborne activity in a specified aerosol is associated with particles smaller than the AMAD, and 50% of the activity is associated with particles larger than the AMAD.

Assessment

Systematic evaluation of the arrangements, processes, activities and related results for their adequacy and effectiveness in comparison with set criteria.

Atomic Energy Regulatory Board (AERB)

A national authority designated by the Government of India having the legal authority for issuing regulatory consent for various activities related to the nuclear and radiation facility and to perform safety and regulatory functions, including their enforcement for the protection of site personnel, the public and the environment against undue radiation hazards.

Commissioning

The process during which structures, systems, components and equipment of a nuclear or radiation facility, on being constructed, are made functional and verified in accordance with design specifications and found to have met the performance criteria.

Committed Effective Dose, E (τ)

The time integral of the whole body effective dose rate following an intake of a radionuclide. The quantity 'E (τ)' is defined as

$$E(\tau) = \sum_{T} w_T . H_T(\tau)$$

Where $H_T(\tau)$ is the committed equivalent dose to tissue or organ T over the integration time τ elapsed after an intake of radioactive substances and w_T is the tissue weighting factor for tissue or organ T. Where τ is not specified, it will be taken to be 50 years for adults and the time to the age of 70 years for intakes by children.

Committed Equivalent Dose, $H_T(\tau)$

The time integral of the equivalent dose rate in an organ or tissue following an intake of a radionuclide. The quantity '$H_T(\tau)$' is defined as

$$H_T(\tau) = \int_{t_0}^{t_0+\tau} \dot{H}_t(t)dt$$

where 't_0' is the time of intake, '$H_T(t)$ dt' is the equivalent dose rate at time 't' in an organ or tissue 'T' and 'τ' is the integration time elapsed after an intake of radioactive substances. Where τ is not specified, it is taken to be 50 years for adults and the time to the age of 70 years for intakes by children.

Competent Authority

Any official or authority appointed, approved or recognised by the Government of India for the purpose of the Rules promulgated under the Atomic Energy Act, 1962.

Consentee

A person to whom consent is granted by the competent authority under the relevant rules.

Contamination

The presence of radioactive substances in or on a material/the human body or other places in excess of quantities specified by the competent authority.

Controlled Area

A delineated area to which access is controlled and in which specific protection measures and safety provisions are, or could be, required for

(a) controlling normal exposures or preventing the spread of contamination during normal working conditions; and

(b) preventing potential exposures or limiting their extent should they occur.

Derived Air Concentration

That activity concentration of the radionuclide in air (Bq/m^3) which, if breathed by reference man for a working year of 2000 h under conditions of light physical activity (breathing rate of 1.2 m^3/h), would result in one annual limit on intake, or the concentration, which for 2000 h of air immersion, would lead to irradiation of any organ or tissue to the appropriate annual dose limit.

Derived Limits

A limit on a measurable quantity set, on the basis of a model, such that compliance with the derived limit may be assumed to ensure compliance with a primary limit.

Deterministic Effect

A radiation induced health effect for which generally a threshold level of dose exists above which the severity of the effect is greater for a higher dose. Deterministic effects are also referred to as (harmful) tissue reactions..

Document

Recorded or pictorial information describing, defining, specifying, reporting or certifying activities, requirements, procedures or results.

Dose

A measure of the energy deposited by radiation in a target. The quantities termed absorbed dose, equivalent dose, effective dose, committed equivalent dose, or committed effective dose are used, depending on the context.

Dose coefficient

Used as a synonym for dose per unit intake of a radioactive substance, but sometimes also used to describe other coefficients linking quantities or concentrations of activity to doses or dose rates, such as the external dose rate at a specified distance above a surface with a deposit of a specified activity per unit area of a specified radionuclide.

Dose Constraint

A prospective and source-related restriction on the individual dose delivered by the source, which serves as a bound in the optimisation of protection and safety of the source. For occupational exposures, dose constraint is a source-related value of individual dose used to limit the range of options considered in the process of optimisation. For public exposure, the dose constraint is an upper bound on the annual dose that a member of the public should receive from the planned operation of any controlled source. The exposure to which the dose constraint applies is the annual dose to any critical group summed over all exposure pathways, arising from the predicted operation of the controlled source. For medical exposure the dose constraint level should be interpreted as a guidance level, except when used in optimising the protection of persons, other than workers, who assist in the care, support or comfort of exposed patients.

Dose Limit

The value of the effective dose or the equivalent dose to individuals from controlled practices that shall not be exceeded.

Effective Dose

The quantity ‘E’ defined as a summation of the tissue equivalent doses, each multiplied by the appropriate tissue weighting factor:

$$E = \sum_{T} w_T \, . H_T$$

where ‘H_T’ is the equivalent dose in tissue or organ ‘T’ and ‘w_T’ is the tissue weighting factor for tissue or organ ‘T’.

Employer

Any person with recognised responsibility, commitment and duties towards a worker in his or her employment by virtue of a mutually agreed relationship. (A self-employed person is regarded as being both a worker and employer).

Equivalent Dose ($H_{T,R}$)

The quantity '$H_{T,R}$,' is defined as

$$H_{T,R} = w_R . D_{T,R}$$

where '$D_{T,R}$' is the absorbed dose delivered by radiation type 'R' averaged over a tissue or organ 'T' and 'w_R' is the radiation weighing factor for radiation type 'R'. When the radiation field is composed of different radiation types with different values of 'w_R' the equivalent dose is

$$H_T = \sum_R w_R . D_{T,R}$$

Exposure

The act or condition of being subject to irradiation. Exposure may be either external (irradiation by sources outside the body) or internal (irradiation by sources inside the body). Exposure can be classified as either normal exposure or potential exposure; occupational, medical or public exposure; and in intervention situations, either emergency exposure or chronic exposure. The term 'exposure' is also used in radiation dosimetry to express the amount of ions produced in air by ionising radiation.

Investigation Level

The value of a quantity such as effective dose, intake, or contamination per unit area or volume, at or above which an investigation should be conducted.

Intake

The process of taking radionuclide into the body by inhalation or ingestion, or through the skin, and the amount of given radionuclide taken in during a given period.

Monitoring

The continuous or periodic measurement of parameters for reasons related to the determination, assessment in respect of structure, system or component in a facility or control of radiation.

Nuclear Fuel Cycle

All operations associated with the production of nuclear energy, including mining, milling, processing and enrichment of uranium or processing of thorium, manufacture of nuclear fuel, operation of nuclear reactors, reprocessing of irradiated nuclear fuel, decommissioning, and any activity for radioactive waste management and research or development activity related to any of the foregoing.

Occupational Exposure

All exposures of personnel incurred in the course of their work.

Occupational Worker

Any person, working full time or part time in a nuclear or radiation facility, who may be employed directly by the "consentee" or through a contractor.

Operation

All activities following and prior to commissioning performed to achieve, in a safe manner, the purpose for which a nuclear/radiation facility is constructed. For nuclear power plants, this includes maintenance, refuelling, in-service inspection and other associated activities.

Quality Assurance (QA)

Planned and systematic actions necessary to provide the confidence that an item, process or service will satisfy given requirements for quality.

Radiation Surveillance

Measures that may be specified by the competent authority to provide adequate radiological protection either generally or in any individual case.

Radiation Worker

Any person who is occupationally exposed to radiation, and who in the opinion of the regulatory body, should be subjected to radiation surveillance.

Radiological Safety Officer (RSO)

Any person who is so designated by the employer and who, in the opinion of the competent authority is qualified to discharge the functions outlined in the Radiation Protection Rules, 2004.

Records

Documents which furnish objective evidence of the quality of items and activities affecting quality. It also includes logging of events and other measurements.

Reference Level

Action level, intervention level, investigation level or recording level established for any of the quantities determined in the practice of radiation protection.

Regulatory Constraints

Restrictions on radiation protection parameters as specified by the regulatory body.

Source

Anything that causes radiation exposure, either by emitting ionising radiation or releasing radioactive substances or materials.

Source Region

An anatomical region within the reference phantom body which contains the radionuclide following its intake. The region may be an organ, a tissue, the contents of the gastrointestinal tract or urinary bladder, or the surfaces of tissues as in the skeleton, the alimentary tract, and the respiratory tract.

Stochastic Effects (Radiation)

Radiation effects generally occurring without a threshold level of dose whose probability is proportional to the dose and whose severity is independent of the dose.

Supervised Area

Any area not designated as a controlled area but for which occupational exposure conditions are kept under review even though specific protective measures and safety provisions are not normally needed.

Surveillance

All planned activities, viz. monitoring, verifying, checking including in-service inspection, functional testing, calibration and performance testing carried out to ensure compliance with specifications established in a facility.

Target Tissue/Organ

The tissue or organ to which radiation is directed.

Working Level (WL)

A unit of potential alpha energy concentration (i.e. the potential alpha energy per unit volume of air) resulting from the presence of radon progeny or thoron progeny, equal to 1.3×10^5 MeV per litre. In SI units, a working level is 2.1×10^{-5} J/m^3.

Special Definitions

Reference Biokinetic Model (ICRP 130, 2015)

A reference biokinetic model describes the intake, uptake, distribution, and retention of a radionuclide in various organs or tissues of the body, and the subsequent excretion from the body by various pathways.

Clearance (IAEA Glossary)

The net effect of the biological processes by which radionuclides are removed from a tissue, organ or area of the body.

Uptake (ICRP 130, 2015)

Activity that enters blood from the respiratory or alimentary tract through the skin or wound.

Bioassay (IAEA Glossary)

Any procedure used to determine the nature, activity, location or retention of radionuclides in the body by direct (in vivo) measurement or by in vitro analysis of material excreted or otherwise removed from the body

LIST OF ABBREVIATION

AERB	:	Atomic Energy Regulatory Board
AMAD	:	Activity Median Aerodynamic Diameter
ASV	:	Anodic / Adsorptive Stripping Voltametry
BOMAB	:	Bottle Mannequin Absorber
Ca-DTPA	:	Calcium diethylene triamine penta acetic acid
CED	:	Committed Effective Dose
DAC	:	Derived Air Concentration
DRL	:	Derived Reference Level
DNAA	:	Delayed Neutron Activation Analysis
FTA	:	Fission Track Analysis
GSR	:	General Safety Requirements
HEP	:	High Energy Photon
IAEA	:	International Atomic Energy Agency
ICP-MS	:	Inductively Coupled Plasma Mass Spectrometry
ICRP	:	International Commission on Radiological Protection
JAERI	:	Japan Atomic Energy Research Institute
IL	:	Investigation Level
LEP	:	Low Energy Photon
LLNL	:	Lawrence Livermore National Laboratory
LLRDS	:	Low Level Radon Detection System
MDA	:	Minimum Detectable Activity
MEQ-CWT	:	Muscle Equivalent Chest Wall Thickness
MRI	:	Magnetic Resonance Imaging
MRL	:	Medical Referral Level
NAA	:	Neutron Activation Analysis
NCRP	:	National Council on Radiation Protection and Measurements
NODRS	:	National Occupational Dose Registry System
OBT	:	Organically Bound Tritium
PAS	:	Personal Air Sampler
PIPS	:	Passivated Implanted Silicon
RL	:	Recording Level
RSO	:	Radiological Safety Officer
SAS	:	Static Air Samplers
SEE	:	Specific Effective Energy
SSNTD	:	Solid State Nuclear Track Detector
TLD	:	Thermo Luminescent Dosimeter
WBC	:	Whole Body Counter
WBM	:	Whole Body Monitor

CONTENTS

1. INTRODUCTION

1.1 General

Radiation exposure, both external and internal, can occur to radiation workers during the operation of various nuclear fuel cycle facilities and radiation facilities. In view of the potential health effects of radiation exposures, it is necessary that all activities involving radiation exposures and the facilities handling radioactive materials are regulated. Therefore, radiological protection programs are usually planned to prevent, control / limit and minimize the exposures,so that deterministic effects are prevented and stochastic effects are reduced to the extent reasonably achievable. The basic requirements for such radiation protection programme are given in the AERB Safety Code on "Radiation Protection for Nuclear Fuel Cycle Facilities" [1]and AERB Safety Code on "Radiation sources, equipment and installations', AERB/RF/SC"[1].

The assessment of radiation doses to workers routinely or potentially exposed to radiation through intake of radionuclide is an integral part of the radiation protection programme.Approved procedures are required to be used for assessment of both external and internal exposures. Unlike external exposure, internal exposure cannot be measured directly. Its evaluation is based on the calculation of the intake of radionuclide either from direct measurements (e.g. external monitoring of whole body or ofspecific organs and tissues) or indirect measurements (e.g. radioactivity in urine, faeces, breath or samples from the working environment)[2][3].

1.2 Objective

The objective of this Safety Guide is to provide criteria and guidance for evaluating the radiation dose that may be received following intakes of radioactive materials due to occupational exposure. This safety guide provides the methodologies and procedures for assessment of radiation dose due to intake of various radionuclides. This Safety Guide also provides guidance for monitoring of internal exposures in both routine and off-normal situations, using direct and indirect methods.

1.3 Scope

This guide is applicable for the internal dosimetry of personnel working in nuclear fuel cycle and radiation facilities having potential for internal exposure such as:

a) Mining and milling of uranium and thorium ores;

b) Fuel enrichment/fabrication facilities;

c) Nuclear power plants;

d) Research/experimental reactors;

e) Fuel reprocessing plants;

f) Radioactive waste management plants;

[1] The Safety Code (AERB/RF/SC) is under preparation.

g) Radiation facilities producing or handling large quantities of radionuclides for medical, industrial, research purpose etc.; and

h) Any other facility as determined by AERB

This Safety Guide does not cover the medical exposure of patients or exposure of members of the public.

2. ROLES AND RESPONSIBILITIES

2.1 General

Consentee / licensee has the overall responsibility for the radiological safety of the radiation workers. The consentee / licensee shall ensure that the radiation protection programme put in place has an appropriate internal dosimetry components commensurate with the radiological hazards associated with the operations of the facility [4]. Roles and responsibilities of consentee/ licensee, Radiological Safety Officer (RSO), the radiation worker and the in-charge of internal dosimetry laboratory are stated in this section.

2.2 Consentee / Licensee

The consentee /licenseeis responsible for the dose assessmentof radiation workers through an appropriate individual monitoring programmeandshall maintain the dose record as specified by AERB[1]. The consentee / licenseeshallensure establishment of an internal dosimetry programme. The consentee / licenseeshould use only laboratories recognized by AERB for internal dose assessment.The internal dosimetry programme should include the following features:

a) Defined criteria for identifying workers who need to be covered in the individual monitoring programme;

b) Appropriate bioassay measurement methods and the measurement frequencies;

c) Methods for collection, control, accountability, and safe handling of samples;

d) Scheme for timely analysis of bioassay samples and measurements;

e) Defined criteria and actions for identifying individuals with suspected intakes, based on workplace monitoring and bioassay measurements;

f) Establish appropriate reference levels;

g) Formulation of action plan for medical management of internally contaminated workers; and

h) Defined program to report internal doses to AERB and workers.

The consentee / licensee should investigate case(s) of exposure, if any, in excess of regulatory constraints / limits received by individual workers in consultation with the Radiological Safety Officer (RSO) and maintain records of such investigations. The consentee / licenseeshall informAERB promptly of the occurrence, investigation, follow-up and corrective actions in cases of exposures in excess of regulatory constraints / limits. The consentee / licensee should also examine the overall trend of internal dose contribution over the years.

2.3 Radiological Safety Officer (RSO)

The Radiological Safety Officer (RSO) is responsible for the implementation, coordination, and day-to-day overseeing of the radiation protection programme. As

part of this responsibility, withrespect to internalexposure monitoring programme, RSO should:

a) Ensure that the radiation workers are covered by an appropriate internal dosimetry programme;

b) Assist the consentee / licensee in coordinating with the internal dosimetry laboratory for implementation of the bioassay programme;

c) Instruct the workers on the importance of bioassay programme;

d) In case of off – normal situation, recommend the workers for in vivo as well as in vitro monitoring;

e) Ensure timely entry of internal dose data into the individual dose record;

f) Assist the consentee / licensee in carrying out investigation in cases exceeding reference levels;

g) Furnish reports to AERB on cases exceeding investigationlevels;

h) Coordinate with the hospital staff in cases requiring medical attention;

i) Obtain physicochemical parameters, such as isotopic composition, solubility and particle size data wherever required.

2.4 Radiation Worker

Every radiation worker with respect to internalexposure controlshould;

a) Adhere to the recommendationsof the RSOincluding use ofpersonal protective equipments (PPEs);

b) Provide samples of urine / faecesetc. for bioassay purposes and / or report for whole body counting / organ monitoring as instructed by the RSO;

c) Assist the consentee / licensee and the RSO in investigations when the reference levels are exceeded.

2.5 In charge of the Internal Dosimetry Laboratory

Internal dosimetry laboratory shall have the accreditation from relevant agency recognised by AERB. The In charge of the internal dosimetry laboratory should;

a) Co-ordinate with the RSO / consentee / licensee for implementation of bioassayprogramme;

b) Carry out necessary follow-up measurements in case of significant intake, if required;

c) Collect and assess the monitoring results obtained by all the individual monitoring techniques before computation of internal dose based on information provided by the consentee / licensee and the RSO;

d) Report the results to the Consentee / Licensee.

3. METHODS FOR INTERNAL DOSIMETRY

3.1 General

Internal dose is the radiation exposure that results from the intake of radioactive materials into the body by inhalation, ingestion, absorption through the skin or via wounds. Assessment of radiation doses arising from the intake of radioactive material by the workers is termed as internal exposure assessment. It is primarily needed for demonstrating compliance with the dose constraint /limits and facilitating safe operations in nuclear or radiological facilities. In case of off-normal exposures, it can provide useful information for taking appropriate intervention measures. Evaluation of internal exposure of an individual is mostly based on bioassay measurements. Another method of internal dose assessment is based on the measurement of airborne radionuclides in the working areas of the facility and the worker's occupancy in those areas.

3.2 Criteria for Individual Internal Monitoring

Individual monitoring measures are necessary if there is a likelihood of workers receiving internal dose of 1 mSv or more from all occupational intakes of radionuclides in a year[5]. The consentee / licensee, in consultation with the RSO should identify such workers for monitoring purposes.

The probability of workers exceeding an internal dose of 1 mSv per year should be assessed on the basis of the following considerations;

a) The amount of radioactive material handled by the worker;

b) The type of radionuclides involved;

c) The physical and chemical form of the radionuclides;

d) The type of containment used;

e) The nature of the operations performed;

f) The air activity in the operating area; and

g) Workers to whom TLD / work permit issued and / or in the opinion of RSO considering exposure potential.

In addition to these, before introducing any new operations or starting a new facility, individual monitoring is to be implemented for all radiation workers and reviewed over the years based on the experience accumulated.

Individual monitoring for internal exposures should be carried out for

a) Generation of baseline bioassay data for new workers;

b) Routine bioassay of radiation workers;

c) Investigation of any incident at workplace;

d) Workers with potential for inhalation exposures from radon/thoron daughters; and

e) Demonstration of regulatory compliance of the dose limit.

3.3 Types of Monitoring

Following types of internal monitoring programmes are generally needed [2]:

a) Routine monitoring

b) Special or task related monitoring

c) Confirmatory monitoring.

3.3.1 Routine Monitoring

Routine bioassay monitoring is the usual and the most frequent type of bioassay measurements which indicates internal exposures that might have been incurred by the workers during the course of their work either during normal operations or due to minor events. It also includes the baseline bioassay monitoring which is carried out to establish a pre-employment condition of internal deposition, which may exist in any individual before undertaking work, involving handling of radioisotopes. These measurements form the personal baseline and need to be subtracted from results obtained during subsequent routine monitoring programmes in the facility to identify potential intakes, if any, before assignment of internal dose.

The clearance rate and/or effective half-life of the internally deposited radioactive materials and the minimum detectable activity (MDA) of the measurement technique govern the frequency of routine internal monitoring. It is recommended that monitoring periods should generally be selected such that assumption of intake occurring at the midpoint of two successive monitoring intervals, would not lead to an underestimation of intake by a factor of more than three[2][6].

Annual monitoring frequency is commonly adopted for radionuclides like ^{90}Sr, ^{137}Cs, ^{60}Co,^{232}Th, U(nat), ^{233}U, $^{239+240}$Pu, ^{241}Am, etc. which have long effective retention in the body whereas for radionuclides like ^{3}H and ^{131}I, weekly/fortnightly monitoring is recommended.

3.3.2 Special / Task Related Monitoring

Special monitoring is performed in response to a particular circumstance, such as a known or suspected intake of radioactive material due to an abnormal incident in the workplace and on completion of planned operations. Monitoring of temporary workers may also fall in this category. The reason for the special monitoring and interpretation of the results should be clearly identified and recorded. In all such cases, internal monitoring should begin as early as possible and dose assignment as well as refinement, if required, should be based on follow-up bioassay measurements.

In case of special monitoring there is fair knowledge on time and type of intake. When the time of intake is not known (or cannot easily be determined), it should be assumed that the intake occurred at the mid-point of monitoring interval or of the planned operation.

Special internal monitoring is recommended in following situations:

a) Nasal smears, nose blows showing activity in excess of the prescribed action levels;

b) Injury resulting in wound in controlled areas;

c) Surface contamination and air activity levels in excess of the derived levels (100 DAC-h = 1 mSv committed effective dose);

d) Contamination of the face or mouth area due to incidents like spillage;

e) Unexpected air activity levels and workers found in the area without adequate respiratory protection;

f) Detection of contamination inside the respiratory protection equipment worn during any special work permit operation;

g) High level of activity in urinary / faecal excretion, whole body/ organ.

3.3.3 Confirmatory Monitoring

The supervised area generally has very low potential for internal contamination because no radioactive material is handled openly in this area and radioactive contamination is not likely to occur. Hence, there is no need to establish a routine internal monitoring program for the workers in this area. However, provision should be made for confirmatory monitoring as per the frequency specified by RSO or once in five years, whichever is earlier,for few workers working in supervised area to confirm that working conditions at the facility are satisfactory.

A confirmatory bioassay program is intended to verify that

a) Assumptions about the radiological exposure conditions in the workplace are valid;

b) Protection measures are effective during any off-normal event

c) Routine bioassay is not required.

3.4 Assessment of Internal Contamination

The assessment of intake of radionuclide can be carried out, in most cases by whole body / organ activity measurements, excreta monitoring, air sampling with personal/static air samplers or a combination of these techniques.

The choice of measurement technique will be determined by several factors like

a) Characteristics of radiation emitted by the radionuclide

b) Biokinetic behaviour of the contaminant

c) Likely level of dose.

d) Required frequency of measurements and

e) Sensitivity; availability, and convenience of the appropriate measurement techniques /systems.

Methods of internal contamination monitoring can be classified in two categories:

(i) Direct method or in vivo method which refers to direct measurement of the amount of radioactive material deposited in organs, tissues, or the whole body. Common methods are thyroid monitoring, lung/organ monitoring and whole body monitoring.

(ii) Indirect method or in vitro method which refers to measuring the amount of radioactive material in biological samples collected from the workers. The most common method is urine analysis. Other methods are faecal, nasal swabs, blood analysis and breath analysis especially for radon / thoron inhalation intakes. It also includes analysis of filter papers collected from static and / or personal air samplers.

Routine monitoring programme generally involves only one type of monitoring if adequate sensitivity can be achieved. For some radionuclides only one measurement technique is feasible for example urine monitoring for pure beta emitters like ^{3}H. For radionuclides like plutonium isotopes,due to limitations of measurement as well as interpretation of the measured data, a combination of techniques may have to be used to assess the dose.

3.5 Assessment of Internal Dose

For intakes of radionuclides in the workplace, a number of routes are possible including inhalation, ingestion and entry through intact skin and wounds. In inhalation, the radioactive contaminant enters the body through the nasal passage and gets deposited in the respiratory tract. A fraction of the material deposited in the respiratory system will be absorbed in the blood and some part of the deposited material transferred to the throat and swallowed, resulting in possible absorption of radionuclides from the gastrointestinal tract. In occupational exposure, the main route of intake is by inhalation. Intake by direct ingestion may occur by accidental swallowing of radioactivity, eating with contaminated hands or eating contaminated foods etc. Some radionuclides like tritium, radioiodines may get absorbed through the intact skin. Damage to the skin by cuts or other wounds can also result in intakes of radionuclides. Following the intake of radionuclide, the dose is delivered as the material gets metabolized and deposit in various organs. For radiation workers the commitment period over which the effective dose is assessed is taken as 50 years following the intake [7].

3.6 Biokinetic Models

Biokinetic and dosimetric models published by the ICRP and the National Council on Radiation Protection & Measurements (NCRP) are recommended for computation of dose[8][9][10][11][12][13] [14].Annexure I gives the description of the ICRP models for selected important radionuclides.

3.6.1. Respiratory Tract and Gastro Intestinal Tract Models

The deposition and transport behaviour of inhaled radionuclides in the lungs depends on the particle size of the aerosols and their solubility characteristics in lung fluids[10].The knowledge of particle size distribution in the work environment is necessary for the interpretation of results in terms of intake and Committed Effective Dose (CED). In the absence of such information, default value of 5μm AMADas recommended by ICRP [10] is used for assessment of the intake. Default parameters for material deposited in the respiratory system are assigned Types F, M and S for material cleared with Fast, Moderate and Slow kinetics, respectively. Soluble radionuclides readily enter the blood stream from the lungs and deposit in body organs

and their elimination from the body is mainly through urinary excretion. For insoluble radionuclides, the lung is usually the target organ because it retains them for a long time. In the event of incidents involving significant intakes, efforts should be made to obtain the physicochemical characteristics of the aerosols like their particle size distribution and solubility of the inhaled particles.

The retention fractions in various regions of the gastrointestinal tract and ingestion dose coefficients [15] [16] presently being used are based on the gastro intestinal tract model [8].It has four compartments representing namely the stomach, the small intestine, the upper large intestine and the lower large intestine with mean residence times of 1, 4, 13, 24 h respectively. It is assumed that the uptake / absorption of radionuclides to the blood takes place only from small intestine and is specified by fractional uptake (f_1) values.

3.6.2 Wound Model

The National Council on Radiation Protection and Measurements (NCRP) has published a biokinetic model for radionuclide-contaminated wounds[14]. The model comprises seven categories for absorption from wound site to blood. This categorization depends on the chemical characteristics of the radionuclides which is injected. These categories are weakly, moderately, strongly and avidly retained in case of solutions & colloid, and particle and fragments in case of solids. The wound model consists of following compartments: Fragment (FRG); Particles, Aggregates and Bound States (PABS); Trapped Particles and Aggregates (TPA); Colloid and Intermediate State (CIS); Soluble (SOL); Lymph nodes (LN); and Blood (or transfer compartment). Once injected and absorbed into the blood, the distribution of the radionuclide is similar to that material entering the blood from lungs or alimentary tract. This model can be adopted for computation of internal dose using Integrated Module for Bioassay Assessment (IMBA) software or by using the wound retention, excretion fractions and the dose coefficients published by Toohey et al [17].

3.6.3 Systemic Models

A systemic model describes the time-dependent distribution and retention of a radionuclide in the body after it reaches the systemic circulation, and its excretion from the body. Use of biokinetic models is important for the estimation of organ doses resulting from internal deposition of radionuclides in humans. The generic systemic model structures in ICRP's reports [9] [11] [12] [13] are physiologically more realistic with regard to the dynamics of organ retention and excretion so that they are applicable to the interpretation of bioassay data as well as the calculation of dose coefficients.

3.7 Date of Intake

For assessment of intake and dose, the actual intake time or period should be used when that time is known. When the actual intake time or period is not known, it is necessary to identify the probable time of intake. This may be done by considering available data from operational log books and radiological monitoring reports, such as air monitoring results, contamination surveys, unusual occurrences report, operating periods, specific tasks performed and previous bioassay measurement results. After the

intake time is narrowed to a probable time period, it is assumed that an acute intake occurred at the midpoint of that period. If the evidence suggests that a chronic intake is more reasonable, it is assumed that the intake occurred uniformly throughout the probable exposure period. In addition, follow-up bioassay measurements data could also be used to correlate the time or period of intake in special circumstances.

3.8 Normalizing Bioassay Data

In-vitro bioassay data should be normalized based on the sampling protocol. Generally, urine data are automatically normalized to a total 24-hour excretion by use of the standard "approximate 24-hour" sampling protocol of collecting overnight urine sample. The information collected from worker should be used to ascertain the representativeness of 24 hrs sample. However, in case of individual monitoring programme for tritium, urine samples need not be collected over a whole day. It is assumed that the activity concentration of tritiated water in urine is equal to the activity concentration in body water [2]. Thus, analysis of spot urine sample can provide information regarding the activity concentration of tritium in body water at the time that sample was collected.

3.9 Assignment of Internal Doses

Internal dose assessments are mostly based on estimated intake. The intake is estimated using available data, preferably bioassay measurements. The 50-year committed effective dose, E(50), is calculated based on the intake multiplied by the appropriate dose coefficient[15]. The 50-year committed doses, assigned to the year of intake, should be used as the basis for compliance monitoring.

3.10 Reference Levels and Derived Reference Levels

A reference level is a predetermined value of a quantity that triggers a specified course of action when exceeded or expected to be exceeded. Reference levels may be dose-based or intake-based.

There can be many reference levels like recording level, investigation level and medical referral levels for any of the quantities determined in the internal dosimetry practices. Derived reference levels are the measurement values for particular bioassay or air sampling results that correspond to a reference level under specifically defined operational circumstances. The reference levels recommended in this guide are as follows:

3.10.1 Recording level (RL)

Recording level (RL) is a defined value of committed effective dose of 1mSv or corresponding intake at annual monitoring frequency above which a result from a monitoring programme is of sufficient significance to require inclusion in a dose record.

$$RL = \frac{0.001}{Ne(g)}$$

Where, N is number of monitoring periods per year and e(g) is the appropriate dose coefficient in Sv/Bqdepending on the route of intake[15].

3.10.2 Investigation Level (IL)

Investigation level (IL) is a defined value of committed effective dose of 2 mSv or corresponding intake at annual monitoring frequency above which investigations are carried out tofind the circumstances and, to the extent reasonable, to determine actual conditions and parameters for dose evaluation, rather than use default assumptions. An investigation may involve special measurements, work history review, determination of material form, and modification of biokinetic parameters, and may culminate in a dose assessment.

$$IL = \frac{0.002}{Ne(g)}$$

3.10.3 Medical Referral Level (MRL)

Medical Referral Level(MRL)[18]is a defined value of committed effective dose of 20 mSv or corresponding intake at annual monitoring frequency above which the medical staff should be notified. The notification should be made as promptly as possible, but does not necessarily constitute an identified need for intervention.MRLhas the value of annual dose limit and if the sum of internal and external exposures exceeds the limit, the worker is to be removed from further radiation work and to be investigated thoroughly. The actions are similar to that of IL except in this case the radiation worker is laid off from radioactive jobs till the dose is assigned / investigation completed. If the assigned dose is more than 20 mSv the case is referred to MedicalOfficer.

$$MRL = \frac{0.020}{N\,e(g)}$$

3.10.4 Derived Reference Levels (DL)

Derived Reference Levels (DLs)are measured values of radioactivity in the whole body / tissue or excreta concentrations or excretion rates that indicate an intake resulting in a dose exceeding one reference level. e.g. For routine monitoring Derived Reference Level (DL) is determined from a reference dose D as

$$DL = \frac{D}{Ne(g)} \times m(T/2)$$

Where, m(T/2) = ICRP predicted fraction of the intake retained in the body or having been excreted from the body at monitoring interval of T days following an intake of 1 Bq

And for special monitoring,

$$DL = \frac{D}{Ne(g)} \times m(t)$$

Where, m(t) = ICRP predicted fraction of the intake retained in the body or having been excreted from the body at time t days following an intake of 1 Bq.

Derived Recording Level (DRL),Derived Investigation Level (DIL) and Derived Medical Referral Level (DMRL) will correspond to the reference doses of 0.001 Sv ,0.002 Sv and 0.020 Sv respectively.

Table 1 gives the Derived Reference Levels and Recommended Actions appropriate to these levels.

Table 1: Derived Reference Levels and Recommended Actions

Bioassay Result	Recommended Action
< DRL	Record the bioassay measurement result as < DRL
DRL ≤ Result < DIL	Confirm and record the bioassay measurement result. If the result is confirmed, assess the CED and include in individual dose record.
DIL ≤ Result <DMRL	Confirm and record the bioassay measurement result. If the result is confirmed, assess the CED and include in individual dose record. Radiation worker laid off from radioactive job to facilitate investigation
≥ DMRL	Confirm and record the bioassay measurement result. If the result is confirmed, assess the CED and include in individual dose record. Radiation worker laid off from radioactive job to facilitate investigation. Case is referred to Medical Officerfor consideration of measures to decorporate radionuclides from the body.

3.11 Requirements of Internal Dosimetry Laboratories

Adequate laboratory and office space should be available to accommodate the necessary equipment and personnel with adequate space for maintaining dose records. Preferably in vivo and in vitro monitoring facilities should be located away from any nuclear and radiation facility in order to minimize background radiation levels and the possibility of contamination. In particular, the in vivo monitoring laboratory should be located at an appropriate distance from areas where radioactive materials are processed, stored or transported or where ionizing radiation is generated. These laboratories shall be designed with sufficient ventilation; filtration and shielding in order to avoid interfering background fluctuations, such as those due to radon. Access

control to all facilities is necessary, both to protect sensitive equipment and to maintain confidentiality of records. Change rooms and showers should be provided at facilities used for direct measurements.

4. DIRECT MEASUREMENT METHODS

4.1 General

Direct measurement of radioactivity content of body or organ provides a quick and convenient estimate of activity in the body [19]. Such a measurement is possible for all those radionuclides which emit radiations that can be detected from outside the body. The technique can be used for radionuclides that emit: χ-ray or gamma radiation; positrons, since they can be detected by measurement of annihilation radiation; energetic beta particles that can be detected by measurement of bremsstrahlung and certain alpha emitters that can be detected by measurement of characteristic χ-rays that follow the alpha decay. This covers all relevant fission / activation products and other important radionuclides like Pu/Am, U, Th etc.

4.2 Direct Measurement Systems

Monitoring systems for the measurement of radionuclides in the whole body or in regions of the body consist of one or more number of high efficiency detectors housed in well-shielded, low-background environments along with the necessary electronics. The geometrical configuration of the detectors is arranged to suit the purpose of the measurement, e.g. the determination of whole-body activity or of activity in a region of the body such as the thorax or the thyroid or the liver. The skull or knees may be used as a suitable site for measurement of bone seeking radionuclides.

Commonly encountered fission and activation products, such as ^{131}I, ^{137}Cs and ^{60}Co, can be detected with comparatively simple equipment at levels that are adequate for radiation protection purposes. Such simple equipment may consist of a single detector, viewing the whole body or a portion of the body/ wound, or, for iodine isotopes, a small detector placed close to the thyroid. In contrast, highly sensitive techniques are needed for monitoring a few radionuclides at the levels that are required for protection purposes. Examples are ^{210}Pb, ^{241}Am and plutonium isotopes.Table 2 gives types of in-vivo methods that should be used for measurement of various radionuclides.

Table 2: In Vivo Methods for Important Radionuclides

In Vivo Method	Radionuclide
Whole Body Monitoring	^{40}K, ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{57}Co, ^{58}Co, ^{60}Co, $^{95}Zr/^{95}Nb$, ^{106}Ru, ^{110m}Ag, ^{124}Sb, ^{125}Sb, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{203}Hg, ^{226}Ra, ^{228}Ra, etc.
Lung/organ Monitoring	^{60}Co, ^{210}Pb, ^{232}Th, ^{235}U, ^{238}U, ^{239}Pu ^{240}Pu, ^{241}Am, etc.
Thyroid Monitoring	^{125}I, ^{131}I, etc.

4.3 Measurement Geometry

For whole body counting of High Energy Photon (HEP) emitters which are distributed uniformly throughout the body, the linear scanning geometry provides the best

sensitivity. The scanning is accomplished by either moving the detector or the stretcher at a fixed speed in a linear axis. It can be used to generate a profile curve of radioactivity in the body that can be further used to assess the radioactive contents in different organs or regions of the body. For radionuclides that are highly localized in particular organs or tissues, after incorporation e.g. iodine in thyroid, monitoring of that specific site is recommended. Localized monitoring is also recommended when intake is through a wound, using conventional β-γ detectors. However, in case of wound contamination with α-emitters, detection is more difficult as the low energy χ-rays that follow the α-decay are severely attenuated in tissue. For Low Energy Photon (LEP) emitters, the detectors need to be positioned as close to the organ as possible e.g. in case of lung counting of actinides anterior chest offers the maximum advantage.

4.4 Detection System

For HEP emitters, mainly fission and activation products, NaI(Tl) and HPGe detectors are used with partially or totally shielded arrangements. NaI(Tl) are most popular. The partially shielded systems such as shadow shield, shielded chair or standing type are most popular and design of their shield is such that no gamma ray from the environment can reach the detector directly. Standing type Whole Body Counter (WBC) typically uses minimum two NaI(Tl) detectors positioned in a linear array. The system is designed to achieve sensitivity comparable with conventional Whole Body Monitors (WBM) for a 1 - 2 minutes counting time. The chair and quick scan type of WBM are useful during radiation emergencies when rapid measurements are required. For thyroid monitoring system, detectors like HPGe and NaI(Tl) can be mounted in simple chair type of geometry to measure ^{125}I, ^{131}I and ^{133}I.

For LEP emitters, phoswich detector is used. Detection sensitivity is improved by reduction of background in the primary thin NaI(Tl) crystal in combination with thick CsI (Tl) crystal working in anti-coincidence with the NaI(Tl) crystal. Arrays of three to six HPGe detectors are also widely being used for the monitoring of actinide (LEPs) contamination in specific organs such as lungs and liver. The superior energy resolving power of HPGe detectors simplifies the interpretation of spectra obtained from complex mixtures of radionuclides as compared to phoswich detectors. Both these detectors need to be placed inside a well shielded steel room to reduce the natural background radiation. Totally shielded steel room is constructed with typically 20 cm thick mild steel walls all around which are lined inside with [Pb (3cm) +Cd (2 cm) +Cu (0.5 cm)] to reduce the background radiation.

Miniature cadmium telluride (CdTe) offer high sensitivity for detection of low energy photons and are ideal for localized wound monitoring. These detectors can be operated at room temperatures, require no shielded enclosure, thus are more useful in quick assessment and enable timely surgical excision procedure.

4.5 Calibration Procedures

The purpose of calibration is to derive factors, which will facilitate the conversion of observed net count-rates from a whole body counter to contents of radionuclides in the body. Calibration of all in-vivo monitoring systems used for measurement of

radionuclides should be carried out at least annually using standard phantoms. However, the energy resolution and efficiency of detection system should be checked weekly with test sources and the record of the same should be maintained. The standard method of calibration is based on the use of anthropomorphic phantoms. These phantoms are constructed from the tissue equivalent materials (in terms of density and effective atomic number) to closely approximate the human body and its various organs in their shapes and sizes.

For HEP emitters, Masonite phantoms or Bottle Mannequin Absorber (BOMAB) phantom should be used for calibration. Calibration can be done with either point source or uniformly distributed sources in the entire body or its different organs. In case of in vivo systems used for radioiodines measurements, phantom representing the shape and size of the thyroid should be used for the calibration of thyroid monitor.

However, when calibrating in-vivo measurement systems for LEPs, especially isotopes of plutonium, ^{241}Am and uranium appropriate phantoms should be used. The most commonly used phantoms are those developed by (a) Lawrence Livermore National Laboratory (LLNL), or (b) Japan Atomic Energy Research Institute (JAERI) phantom. The efficiency and MDA for estimating activity in the lung are strongly affected by the Muscle Equivalent Chest Wall Thickness (MEQ-CWT) of the subject. The CWT can be determined using ultra sound technique or a correlation should be made between physical measurements of the subject's weight, height; chest circumference and chest wall thickness. This should be used to estimate chest wall thickness of subject being monitored. The estimated MEQ-CWT is used to derive subject specific calibration factors of the actinide lung monitoring system [20].

The calibration of in-vivo measurement systems can also be performed using computational voxel phantoms [21]. The voxel phantom is constructed using data from Computed Tomography (CT) or Magnetic Resonance Imaging (MRI). Monte Carlo simulations are used to model photon transport from the phantom and the detection of photons by a simulated detector [22]. Using this numerical calibration technique, calibration factors with respect to body size, body shape and radionuclide distribution can be derived.

4.6 Whole Body / Organ Counting Procedure

All the radiation workers undergoing whole body counting /organ monitoring should take a shower to remove any external loose deposition of radioactivity and change over from personal clothing to pre-monitored contamination free clothes provided by the monitoring laboratory. Accessories such as jewellery, watches and spectacles should be removed to eliminate any interference in the counting. The monitoring laboratory should have the following facilities.

a) Personnel decontamination facilities should be provided in proximity to the counting facility and away from contaminated area. Decontamination measures, such as showering, followed by recounting may be used as a means for removing the external contamination.

b) The measurement room should be provided with anti-claustrophobial features such as a fail-safe door opening service that can be operated by the individual being monitored, a two way communication system, CCTV, musicetc.

c) The measurement area should have adequate ventilation.

Wherever liquid nitrogen is used appropriate personal safety measures should be provided.

4.7 Background Measurement

Background counts arising in the detector are normally attributed to four sources:

a) Ambient background radiation from natural sources, such as cosmic rays or radon and its decay products;

b) Background radiation from the shielding, nearby nuclear/radiological facilities and other equipment;

c) Radiation from natural radioactivity in the subject;

d) Radiation scattered into the detector by interactions of the subject with the nearby materials.

For counting systems, background counts should be determined using an appropriate phantom, as similar as possible to the subject to be counted and placed in the defined counting position. For whole body counting, background counts determined using uncontaminated subjects matched with respect to gender, height and weight will improve the results. Measurements of background in the counter should be made as close as possible in time to the measurement of the subject, just before or just after the measurement.

4.8 Minimum Detectable Activity (MDA):

The MDA is primarily dependent on the background count rate and the efficiency factor for the detector system. The standard deviation of background (σ_B) is obtained from the average counts of background spectra of non-radiation worker subjects of varying body builds. The number of background subjects should be at least 20 and the spectra region (energy band) considered should be equal to 3*FWHM (keV) region. MDA at 95% confidence level is evaluated from:

$$MDA(Bq) = \frac{4.65\sigma_B + 2.7}{E \times I \times T}$$

Where, σ_B is square root of background counts (integral counts of the selected region), T: counting time, E: efficiency factor, I: gamma abundance fraction (can be unity for nuclide specific efficiency). The typical MDAs of various system used for in vivo measurement are given in Table 3.The CEDs are evaluated using ICRP 78 [2] default parameters for inhalation intake.

Table 3: Typical MDAs of in-vivo monitoring systems and corresponding CEDs.

Isotope	Measured organ	MDA (Bq)	Lung absorption Type	Monitoring Frequency	Corresponding CED (mSv)
^{60}Co	Whole Body	100	S	Annual	0.05
^{137}Cs	Whole Body	200	F	Annual	0.01
^{125}I	Thyroid	25	F	Monthly	0.002
^{131}I	Thyroid	100	F	Monthly	0.03
^{241}Am**	Lungs	6	M	Annual	13.5
^{239}Pu	Lungs	2000	S	Annual	518.0
^{239}Pu(Pu:^{241}Am = 3 : 1)	Lungs	18	S	Annual	4.8
^{235}U	Lungs	3	S	Annual	0.6
^{238}U	Lungs	40	S	Annual	7.1
^{228}Th*	Lungs (^{208}Tl)	10	S	Annual	27.8
	Lungs (^{212}Pb)	6	S	Annual	6.0
^{232}Th*	Lungs (^{228}Ac)	10	S	Annual	3.8

*Assuming in equilibrium with its daughter products.

**^{241}Am, if present, can be used as a tracer for ^{239}Pu provided ^{239}Pu: ^{241}Am ratio is known. The ^{239}Pu:^{241}Am ratio depends on the burn-up of nuclear fuel and time elapsed after purification of Pu.

5. INDIRECT MEASUREMENT METHODS

5.1 General

Indirect monitoring programmes usually involve analysis of biological sample collected from the exposed individuals. These include urine, faeces, breath, sputum, blood or filter papers from static /personal air samples, surface swipes etc. In-vitro bioassay monitoring is most suitable for radionuclides which have no gamma ray emissions or which have only low energy photon emissions, particularly the alpha and beta emitters [23**].** For such radionuclides, excreta monitoring may be the only measurement technique for ascertaining internal contamination and hence intake. Bioassay monitoring programmes usually involve analysis of urine due to ease of collection, transportation and analysis. However, faecal analysis is carried out in addition to urine analysis, if an element preferentially excretes via faeces (Type S) or during special incidental exposure.

Breath is a significant route of excretion only for those few materials which are gaseous and are exhaled directly (radon, thoron), or metabolized to gases (CO_2). Analysis of the radon or thoron in breath can provide a sensitive method for estimating the body content of ^{226}Ra (from ^{238}U series) or ^{228}Ra (from ^{232}Th series) respectively. Breath analysis requires the worker to breathe into a breath collecting apparatus for a period up to 30 min depending upon the volume required. Other samples like nose blows or nasal smears are analyzed in case of suspected high level contamination or to confirm that inhalation exposure has taken place [24]. They also provide information regarding the radionuclide content and isotopic composition in the inhaled activity.

Injection of radioactive material into the body or absorption through intact skin is also an additional route of intake of radionuclides. Tissues samples excised from the contaminated wound by medical doctors can be subjected to direct monitoring using HPGe detectors [25] and then radiochemically analyzed to know the injected/ ingested radionuclides and their relative concentrations. Bioassay measurements also need to be carried out before and after every wound excision.

5.2 Sample Collection Methodology

Due to the ease of collection and interpretation of the results, urine samples are preferred. Urine/ faecal samples need to be collected away from workplace or at home after taking bath, to prevent the external contamination of the sample. As diurnal variations are observed in the individuals, collection of 24h bioassay samples is preferable. In case of overnight bioassay samples, the results should be extrapolated to 24 h. It is advisable that the bioassay samples are taken-up immediately for analysis without delay, or when necessary, stabilized until analysis by refrigeration or freezing especially for tritium, or by addition of acid (1% by volume hydrochloric acid or nitric acid) to prevent the growth of pathogens or degradation of the samples. The workers should be provided with an appropriate protocol and the sample collection kit to take home.

Faecal monitoring is more often used in special investigations, particularly following a known or suspected intake by inhalation of Type M or S compounds. In case of off-normal exposure or exposure to insoluble radioactive materials, an additional kit may be given to the workers to collect the faecal samples. Faecal samples undergo biodegradation; therefore they should be analyzed promptly, ashed or preserved by deep freezing.

In case of administration of chelating agents like Ca- DTPA, the monitoring frequency recommended is repeat urine analysis at an interval of 15 days for 12 weeks. After 12 weeks, 3 days pooled urine sample is analyzed for estimation of intake and CED. However, changes in the above frequency of monitoring may be needed based on the observed levels of radionuclides in urine.

5.3 Radiochemical Separation

The radiochemical process adopted depends on the radionuclide that needs to be analysed. Most of these procedures include wet digestion, preconcentration, purification, source preparation and activity quantification. For the analysis known quantities of an appropriate radioactive chemical yield monitor is added to the sample as early in the process as is feasible e.g. during sample preparation. A determination of how much of the yield monitor was lost by the end of the analysis indicates how much analyte loss occurred during the analysis. The analytical results for the analyte are then corrected accordingly [26]. In addition to the actual samples, blank sample analysis is also carried out to study the background interferences and estimate the detection limits.

5.4 Analytical Techniques

The techniques involved in the measurement of the contents of the radionuclides depend on the types of radiation emitted by them. The techniques selected for bioassay monitoring need to be specific, sensitive, precise, rapid, economical and applicable to a wide variety of matrices. Some common analytical techniques used are

a) Alpha Spectrometry
b) Spectrophotometry
c) Laser Fluorimetry (LF)
d) Neutron Activation Analysis (NAA)
e) Delayed Neutron Activation Analysis (DNAA)
f) Kinetic Phosphorescence Analysis
g) Solid State Nuclear Track Detector (SSNTD)
h) Anodic / Adsorptive Stripping Voltametry (ASV)
i) Fission Track Analysis (FTA)
j) Inductively Coupled Plasma – Mass Spectrometry (ICP – MS)

Alpha spectrometry technique using Passivated Ion Implanted Silicon (PIPS) detector is most commonly used for qualification and quantification of alpha emitters in

electrodeposited samples. This technique provides information regarding the isotopic composition of radionuclides in the sample but requires long counting times to achieve the required sensitivity for routine monitoring of most of the actinides. NAA is sufficiently sensitive for estimationof ^{232}Th and ^{238}U at few nanogram levels; however it requires a nuclear reactor for irradiation of the samples and also no information is provided on the isotopic composition of Th and U in the samples. Kinetic Phosphorescence and Laser Fluorimetry detect uranium in samples by 2 to 3 orders of magnitude lower than that compared to alpha spectrometry. High Resolution ICP-MS and FTA [27] are sufficiently sensitive to measure Pu in urine at level well below the DIL. The advantage of High Resolution ICP-MS technique over alpha spectrometry and FTA is its sensitivity and capability to measure ^{239}Pu and ^{240}Pu separately. High Resolution ICP-MS is superior to most of these techniques in terms of capability to measure extremely low concentrations of actinides and also provide information on various isotopic ratios.

For low energy pure beta emitters like ^{3}H, ^{14}C, ^{35}S, ^{241}Pu, liquid scintillation spectrometry is the only method of detection. Other beta emitters mainly fission products present in the sample are estimated using gas flow Geiger-Müller counters or plastic scintillator based detectors.

The gamma ray spectrometric system is required to measure the activity in neutron activated (^{232}Th and ^{238}U) samples [28] [29] and direct estimation of ^{241}Am in ashedfaecal samples.

5.5 Minimum Detectable Activity (MDA)

The MDA [30]of alpha spectrometer is calculated by using the following formula

$$MDA(mBq/day) = \frac{4.65\sigma_B + 2.7}{aDERT}$$

Where, σ_B is square root of background counts in the corresponding spectral region in the recorded alpha spectrum for blank sample for counting time T(mins), a is the appropriate conversion factor (60), D is the fraction of urine sample collected in 24h, E is the absolute efficiency of the alpha spectrometer, R is the recovery fraction, T is the background counting time (mins).Table 4 gives the typical MDAs for in vitro (urine analysis) monitoring and corresponding CEDs in case of inhalation intake using ICRP 78 default parameters [2].

Table 4: Typical MDAs and corresponding CEDs

Isotope	Method	MDA (mBq d^{-1})	Lung absorption Type	Monitoring Frequency	Corresponding CED (mSv)

^{90}Sr	RCA and gas flow GM counter	15	F	Annual	0.008
^{137}Cs	RCA and gas flow GM counter	15	F	Annual	0.0001
^{232}Th	RCA and NAA	0.0008	M	Annual	0.003
		0.0008	S	Annual	0.03
U(nat.)	RCA and LF	0.005	M	Annual	0.0001
		0.005	S	Annual	0.01
U(nat.)	RCA and α-spectrometry	0.5	M	Annual	0.01
		0.5	S	Annual	1.0
^{239}Pu	RCA and α-spectrometry	0.5	M	Annual	3.0
		0.5	S	Annual	26.0
^{239}Pu	RCA and FTA	0.012	M	Annual	0.07
		0.012	S	Annual	0.6
^{241}Am	RCA and α-spectrometry	0.5	M	Annual	1.2

RCA –radiochemical analysis

5.6 Internal Dosimetry for Tritium

5.6.1 General

Tritium is the radionuclide giving rise to internal dose to the radiation workers in a Heavy Water Reactor (HWR) and tritium handling facilities. The pathway for exposure of tritium is mostly through inhalation and skin absorption. Radiation dose from tritium intake cannot be measured directly and is usually estimated by measuring the tritium in bioassay samples (such as urine).

Tritium can exist in several chemical forms including tritiated water (HTO), tritiated gas (HT) and organically bound tritium (OBT).

The most common form of tritium is tritiated water (HTO), where a tritium atom replaces a hydrogen atom in water (H_2O) to form HTO. Being equivalent to water,

tritiated water (HTO) is absorbed quickly and completely in all soft tissues of the body. Following an intake of tritium, a fraction of the tritium can become incorporated into organic molecules such as carbohydrates, fats, or proteins and is termed organically bound tritium (OBT).

The dose of tritium is dependent upon the initial intake and effective removal from the body. Within few hours tritium will equilibrate with the fluid compartments of the body and deliver the dose to the whole body.

5.6.2 Method of dose calculation due to chronic exposure

Tritium rapidly attains equilibrium with body water and since bioassay (urine) sample is representative of body water, concentration in the samples can be used to estimate effective dose due to chronic exposure. Biological half-life in human body varies from 4-10 days and depends on various individual and environmental parameters. In order to control the dose occupational radiation workers are advised to submit urine samples at regular intervals.

5.6.2.1 Dose calculation from urine samples submitted at regular interval

The effective dose E (mSv) due to an average tritium activity of Q (MBq/l) in urine for a period of t (days) is given by

$$\underset{\text{(mSv)}}{E} = \underset{\text{(Bq/l)}}{Q} \times 10^{6} \times \underset{\text{(MeV/dis)}}{0.0057} \times \underset{\text{(J/MeV)}}{1.6 \times 10^{-13}} \times \underset{\text{(sec/day)}}{86400} \times \underset{\text{(Note 1\&2)}}{1.11} \times \underset{\text{(days)}}{(2/3)} \times t \times \underset{\text{(mSv/Sv)}}{1000}$$

$$E\ (mSv) = 0.0583 \times Q \times t$$

Note 1:

ICRP publication 78 [2] states that 97% of tritium activity intake in the form of Tritiated water equilibrates in a few hours with body water and follows biological half-life of 10 days for Reference Person. While the 3% activity, which forms the organically bound tritium (OBT) follows biological half-life of 40 days. Thus the total dose due to both components is 11% higher than the dose calculated by body water activity calculations.

Note 2:

The factor for calculating dose from Tritium in equilibrium with body water to soft tissue is used. 90% of body weight is soft tissue and 60% of body weight is water. Therefore, to arrive at the soft tissue dose the multiplication factor is 2/3.

Total effective dose due to samples submitted at fairly regular interval less than or equal to seven days is calculated by summing over a period as follows.

$$E\ (mSv) = 0.0583 \times \Sigma_i\ (Q_i \times t_i)$$

Where Q_i and t_i are the urine sample average tritium concentrations and corresponding period of intervals.

Committed Effective Dose is calculated by adding contribution from exponential decay of residual activity at the end of monitoring period as follows.

$$\mathbf{E(mSv) = 0.0583\times(\Sigma\ Q_i\times t_i + Q_n\times T_e/0.693)}$$

Where Q_n is last urine sample tritium concentration and T_e is effective half life.

5.6.2.2 Dose calculation from urine samples submitted at interval more than specified period

If the interval between two successive samples is > 7 days and Q_i and Q_{i+1} are first and second urine sample concentrations at start and end of the interval, then the interval is divided into 2 parts. First part is the duration of seven or multiple of seven days and the second part is the remaining duration which is less than seven days. For the first part, the dose is calculated by exponential decay of Tritium activity concentration in the body. Effective dose, $E_{i,1}$ for first part of the interval is calculated as follows.

$$\mathbf{E_{i,1}(mSv)= 0.0583\times(Q_i \int_t e^{-0.693\times t\ /\ Te} dt)}$$

Where, T_e is the effective half-life and integration over t is the duration from 0 to m×7 days where m is an integer.

For calculation of dose in the second part of the interval duration, average concentration method as described in section 5.6.2.1 is used. Average concentration of residual activity concentration obtained by exponential decay of first urine sample concentration at the end of duration of first part and urine sample concentration at the end of the interval are used to calculate dose. Thus

$$\mathbf{Q_{av}= 0.5\times(Q_i\times e^{-0.0693\times t\ /\ Te} + Q_{i+1})}$$

Thus effective dose, $E_{i,2,}$ for second part of the interval is calculated as follows.

$$\mathbf{E_{i,2}\ (mSv) = 0.0583\times Q_{av}\times(t_i - t)}$$

Where t_i is the period of the interval for sample Q_i and Q_{i+1}.

Total effective dose for the interval is given by

$$\mathbf{E_i(mSv)= E_{i,1}+ E_{i,2}}$$

5.6.3 Method of dose calculations for acute exposure

The method of average is very appropriate for chronic uptake cases, but care should be taken while calculating doses for cases where initially tritium concentration in urine was low and later due to involvement in high active jobs, high tritium concentration in urine is observed. The average method will result in underestimation. Under such cases, as the time of intake is known, committed effective dose should be calculated from the date of incident only.

$$\mathbf{E(mSv) = 0.0583\times Q_0\times T_e/\ 0.693}$$

Where Q_0 is extrapolated concentration (uptake) due to acute exposure at the time of intake. If Q is the sample concentration t days after intake then $\mathbf{Q = Q_0\ e^{-0.693\times t/Te}}$.

Or $$\mathbf{Q_0 = Q\times e^{0.693\times t/Te}}$$

For High uptake cases (> 1 IL) the sampling frequency has to be increased to more than 2 samples per week or as specified by RSO commensurate with the potential of intake.

6. WORK PLACE MONITORING METHODS

6.1 General

In certain situations and for certain radionuclides the internal dose has to be assessed based on the monitoring of the working environment as the direct and indirect methods described in the earlier sections cannot be used. For example the workplace monitoring is used for assessment of internal exposures in the case of mining workers exposed to radon and its daughter products. Workplace monitoring data can be used to confirm the requirement for individual monitoring by in-vivo and in-vitro methods.

6.2 Monitoring of Radon and its Daughter products in Mines

Internal exposure of workers in uranium mines mainly arises from inhalation of radon with its progeny. A small fraction of internal dose is also from the inhalation of uranium ore dust. The airborne ore dust is mainly composed of free silica (SiO_2) particles and the alpha activity due to uranium and long-lived daughter products is often very small. All the components of internal exposures should be identified and appropriate measurements should be carried out to assess their contribution to the internal dose. In mill areas and tailings sites, the exposure of the workers is mainly from the long-lived uranium bearing dust or airborne uranium particles.

Internal exposure of workers is therefore evaluated by monitoring radon at the workplace (such as mine or mill) and calculated occupancy period for a set of workers, and using a measured or a default equilibrium factor (F) between radon and its short-lived decay product. Personal radon dosimeters based on SSNTD are used for internal dose assessment. As for the exposure from long-lived uranium radionuclides, routine bioassay techniques can be used.

6.2.1 Radon monitoring

Radon in mines is monitored by drawing a sample of the air through filter in pre-evacuated scintillation cells (grab sample) [31] which are counted for alpha activity using an appropriate counting system in the laboratory. In areas with lower concentration of radon like the mill, tailings site and the environment, a low level radon detection system (LLRDS) or a solid state nuclear track detector (SSNTD) based radon dosimeter is used to measure radon. Techniques developed for measurement of radon, thoron and their progeny are given in the Annexure II.

6.2.2 Radon progeny monitoring

For measurement of radon daughters the air sample is drawn through a filter paper at a known flow rate for a specified time period and counted for alpha activity after a delay time varying from 40 to 90 minutes according to the most widely used Kusnetz's technique also given in the Annexure II.

Radon and the potential alpha energy concentration of its progeny are measured simultaneously to evaluate the equilibrium factor, F, ($F = WL \times 3700 / Rn_{conc}$, in $Bq.m^{-3}$; i.e. 3700 $Bq.m^{-3}$ of Rn in equilibrium with its progeny = 1Working Level). When

the F is not precisely known, the ICRP recommends a default value of 0.4 in work areas.

6.3 Dose Evaluation for Mine Workers

For dose evaluation, radon progeny concentration (WL) is computed from the radon monitoring data and the average equilibrium factor 'F' for different locations. From the progeny concentrations in unit of working level (WL), occupancy period and gamma dose rate obtained from the TLD, the annual effective dose (E) is given as

$$E(mSv) = \frac{K.A.\sum_{i}^{n} WL_i T_i}{170} + \sum_{i}^{n} G_i T_i$$

Where, E =effective dose, mSv

K =conversion factor

WL =radon daughter conc. at i^{th} location (WL)

T_i =time spent at i^{th} location (h)

N =no. of locations

A =annual attendance (d/y)

G_i =gamma dose rate at i^{th} location

6.4 Personal and Static Air Samplers

A Personal Air Sampler (PAS) is portable device specifically designed for the estimation of intake by an individual worker from a measurement of time integrated concentration of activity in air in the breathing zone of the worker. A sampling head containing a filter is worn on the upper torso close to the breathing zone. Air is drawn through the filter by a calibrated air pump carried by the worker. Activity on the filter may be measured at the end of sampling period to give an indication of any abnormally high exposure. The filters can then be retained, bulked over a longer period, and activity determined by radiochemical separation and high sensitivity measurement techniques. An estimate of intake during the sampling period can be made by multiplying the measured average air concentration by the volume breathed by the worker during the period of intake.

PAS monitoring is most often used for radionuclides such as plutonium, for which a very small number of particles may contain activities that would correspond to a significant intake. The statistics of sampling small numbers of events then becomes the critical factor in determining sampling accuracy.

Static Air Samplers (SAS) are also commonly used to monitor workplace conditions, but can underestimate concentration in air in the breathing zone of a worker, typically a factor of up to about 10. Nevertheless, if SAS device are sited appropriately, a comparison of PAS and SAS measurements can be used to define a PAS : SAS air

concentration ratio which can be used in the interpretation of SAS measurements for dose assessment purposes. It should however be recognized that the use of SAS is a relatively indirect method for assessing doses and use of the results to estimate individual dose requires a careful assessment of exposure conditions and working practices. SAS devices are used for collection of aerosols not only for the measurement of concentration but also for characterisation of the aerosols, in particular their size distribution and solubility types F, M & S.

7. INTERPRETATION OF BIOASSAY MEASUREMENTS

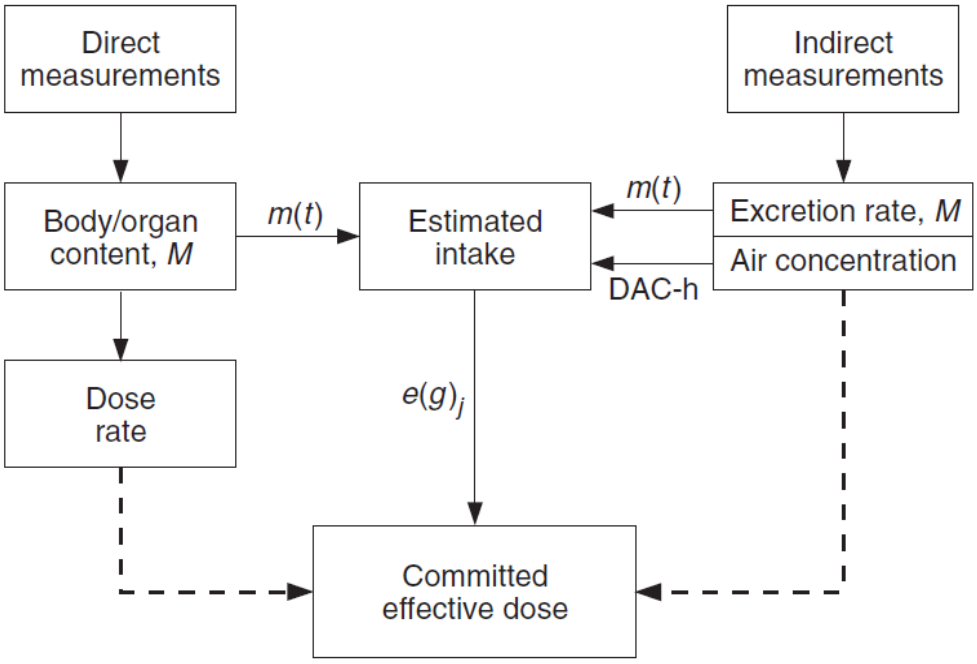
7.1 General

Measurement data obtained by using in vivo and in vitro monitoring techniques are first interpreted using the biokinetic models given by ICRP in terms of intake of radionuclide by the worker. Once the intake is estimated, the committed effective dose is then computed from the product of the intake and the appropriate dose coefficient. Alternatively, in some cases measurements of activity in the body can be used to estimate dose rates directly, but the calculation of CED still requires a biokinetic model in case sufficient numbers of measurements are not available to determine individual specific retention functions.

In order to compute the estimated intake *I*, the measured body content, body region content or excretion rate, *M*, is divided by the fraction of a unit intake, *m*(*t*), which is the fraction retained in the whole body or in body tissues / organs (for direct in vivo measurements) or excreted in a time period, usually per day, from the body (for excreta measurements) at time *t* (usually in days) after intake (Fig. 1).

When only a single bioassay measurement is available, a point estimate of the intake is made. If multiple measurements are available, a best estimate of intake may be obtained by applying a statistical method. When significant intakes may have occurred, more refined calculations based on material data and wherever possible individual specific parameters need to be made.

Fig. 1: General scheme for the interpretation of the results of monitoring measurements (alternative approaches for calculation are indicated as dashed lines)



7.2 Intake Estimate from Special Monitoring

For special monitoring, intake is computed using following equation:

$$I(Bq) = \frac{M}{m(t)}$$

Where, m(t) is the predicted value of the measured quantity at time t days after intake of 1Bq. The *m*(*t*) for selected radionuclides in urine, faeces, the whole body and selected tissues for intakes by inhalation, ingestion and injection are published by the ICRP[2] [17]. A few softwares like Mondes-Mondal [32] and IMBA [33] which are based on the standard ICRP biokinetic models can also be used to obtain the m(t) values.

For whole body or organ retention the m(t) value is the fractional retention in the region considered at time t (in days). Whole body content is the sum of all systemic material, the contents of the urinary bladder, the GI tract and all regions of the respiratory tract. The contents of the lungs are taken to be the sum of the bronchi,bronchioles and alveolar interstial regions together with the thoracic lymph nodes. The contents of the skeleton is taken to be the contents of the bone compartment in the simple models and the sum of all compartments of cortical and trabecular bone and the bone marrow in the more complex models, for example those for plutonium and uranium.

For excretion, in general m(t) is the fraction of the intake excreted during the sampling period of 24 h preceding time t, taking into account radioactive decay. The m(t) values are given for time since intake in days, on an expanding scale (i.e. t = 1, 2, 3,…, 10, 20, 30…, etc.). To obtain a value for a time not listed, a logarithmic interpolation between adjacent values is needed. If the values are not changing rapidly, a linear interpolation may be sufficiently accurate.

7.3 Intake Estimate from Routine Monitoring

For routine monitoring, intake is computed using following equation:

$$I(Bq) = \frac{M}{m(\frac{T}{2})}$$

where, m(T/2) is the predicted value of the measured quantity per Bq intake at the mid-point of monitoring interval. When the time of intake is not known (or cannot easily be determined), it should be assumed that the intake occurred at the mid-point of two successive monitoring data. Computed intake when multiplied by the appropriate dose coefficient (e_{50} Sv Bq^{-1})[15], GSR [16**]** gives the CED receivable by the individual over a period of 50 y following the intake.

$$CED(Sv) = I(Bq) \times e(50) \left(\frac{Sv}{Bq}\right)$$

An intake in a preceding monitoring interval may influence the actual measurement result obtained. For a series of measurements in a routine monitoring programme, the following procedure may be followed:

a) Determine the magnitude of the intake in the first monitoring interval.

b) Estimate the contribution to each of the subsequent measurements from this intake.

c) Subtract the corresponding contributions from all subsequent data.

d) Repeat above for the next monitoring interval.

If a measured value in a routine monitoring programme exceeds a predetermined investigation level (or dose level), special monitoring needs to be initiated for dose refinement.

7.4 Intake Estimate from Multiple Bioassay Data

Usually, the bioassay data for an intake estimate will consist of results for different samples collected at different times, and even from different monitoring techniques, for example urine data and faecal data, and perhaps also from direct measurements. If the initial result from a routine sample indicates a potentially significant intake, special monitoring is started to characterize the intake more accurately. Numerous statistical methods for data fitting are available; the two methods that are most widely applicable are the maximum likelihood method and the Bayesian approach.

7.5 Intake Estimates from Measurements of Related Nuclides (Tracer)

Some radionuclides cannot be measured directly, but their body content can be assessed by the measurement of a daughter nuclide; examples are the in vivo measurement of: ^{228}Ac for the assessment of ^{232}Th in the body; ^{214}Bi for ^{226}Ra; and ^{234}Th for ^{238}U. These assessments rely on assumptions about the activity ratios or equilibrium of the radionuclides or on a well-established degree of non-equilibrium. Values for m(t) for ^{228}Ac in the body following ^{232}Th intakes are given in ICRP [2].

7.6 Workplace Information

The workplace information is required to understand the exposure situations, *e.g.* radionuclides that may have been incorporated (including equilibrium assumptions for the natural radionuclide series), chemical form, particle size, likely time, pattern and pathway of any intake.

7.7 Uncertainties in Internal Dose Assessment

The uncertainty in an internal dose assessment based on bioassay data depends on the uncertainties associated with measurements used to determine the activity of a radionuclide *in vivo* or in a biological sample, uncertainties in the exposure scenario used to interpret the bioassay results, and uncertainties in the biokinetic and dosimetric models used to interpret the bioassay results. The major sources of uncertainties on the estimate of intakes and doses are as follows:

a. Individual monitoring measurements

Uncertainties associated with measurements include representativeness of the samples in terms of volume, time of collection, duration of collection (24h, 78h), counting statistics in case the observed body content or urinary activity excretion level is close to the detection limit of the counting system, correction of the current measurement

result for residual activity from previous intakes can contribute significantly in overestimating the internal dose especially in case of radionuclide with long effective retention in the body.

b. Assessment of intake from the measurements

A principal source of uncertainty in the interpretation of bioassay data is the determination of the time of intake in case of routine monitoring. If the exposure takes place before the mid-point of monitoring interval the intake will be underestimated. The monitoring intervals therefore need to be selected in such a way that any underestimation introduced by the unknown time of intake is no more than a factor of three and can be achieved if the monitoring (T) is less than 3.2 times the effective half – life of the radionuclide in the body ($T \leq 3.2\ T_{eff}$)[6]. Similarly, overestimation in computed intake occurs if the exposure happens just before the monitoring or after mid - point of monitoring interval. This results in significantly higher fraction of the radionuclide in urine. In such situations, the calculated body content is significantly high and is therefore, investigated before making entry in dose records, in order to avoid raising of a false alarm from reporting of such an exposure as an over exposure. Lastly use of default parameters like particle size and chemical nature recommended by ICRP may be sufficient when exposures are well within the prescribed dose limits. However, for exposures approaching or exceeding annual dose limits, more specific information may be needed for realistic dose assessment.

c. Assessment of dose from the intake

Use of standard dosimetric and biokinetic models published by ICRP for Caucasians also add to the uncertainty while computing dose for Indian population. For intakes of radionuclides approaching the annual dose limit, ICRP recommends an individual-specific analysis based on the biokinetic parameter values for that individual. However, use of individual specific model is not justified for small intakes and doses. Whenever medical intervention is carried out to enhance the elimination of the radionuclide from the body the standard models recommended by ICRP cannot be used. In such cases internal doses should be based on the bioassay data collected as part of the detailed investigations.

8. QUALITY ASSURANCE

8.1 General

It is important to establish a quality assurance (QA) programme for the internal dosimetry laboratory. The programme should cover the overall management of the laboratory, equipment, materials, staff, procedures, computer software, documentation etc. The internal dosimetry laboratory should have an in-house QA manual.

8.2 Quality Assurance Plan

The internal dosimetry laboratory should have a written QA plan to ensure conformance to policies, procedures, and instructions. The plan should include the following:

a) Organization structure, management and operational responsibilities;

b) Instructions and procedures, including procedure on software validation;

c) Qualification and training of laboratory personnel;

d) Document control;

e) Documentation and chain of custody of samples;

f) Inspection and testing of material and equipment;

g) Control and maintenance of calibration standards;

h) Preventive and corrective actions;

i) Review of procedures, specifications and operating logs;

j) Observation of operations and evaluation of quality control data;

k) Periodic audit;

l) Quality assurance records;

m) Documentation of detection limit, relative bias, repeatability and methods of calculating results for periodic quality control determinations;and

n) Non Conformance

8.3 Quality Assuranc eProcedures

Performance checks should be conducted to ensure the conformance of analytical processes, measurement equipment and the facilities to predetermine operational requirements. The laboratory should have written quality control procedures to verify that the quality of measurements or radioactivity determinations complies with the accuracy requirements specified in the QA manual. The quality control procedures should include the following:

a) Use of traceable radionuclide reference standards;

b) Performance checks of measurement systems;

c) Instrument calibration;

d) Intra-laboratory analyses (e.g., known quantities, replicates and blanks);

e) Participation in available interlaboratory intercomparison programs;

f) Computational checks;

g) Review of procedures, specifications and operating logs;

h) Observation of operations and evaluation of quality control data;

8.4 Performance Checks of Instrumentation for In-vivo and In-vitro Radiobioassay

Performance of the measurement equipment should be checked and evaluated at regular intervals while the equipment is in use. These checks should be sufficient to demonstrate that the measurement equipment is properly calibrated and that all components are functioning properly. Measurements should include instrument background and response checks. In the case of in-vivo, measurement system response stability should be established by means of a check source and a "tolerance chart". The response should not vary by more than 5% from the established mean.

8.5 Performance Checks on In Vitro Bioassay Procedures

Reagent blank and biological samples of each type known to contain only natural or baseline levels of radioactivity should be analysed periodically to determine the values of appropriate blanks. Samples containing known quantities of each radionuclide of interest should be analysed annually to determine bias and repeatability of the analytical procedures.

8.6 Use of Reference Radioactive Materials for Equipment Calibrations

Radionuclide standards used for equipment calibrations and to test the accuracy of analytical procedures and/or measurement equipment should either be those designated as Certified Reference Material (CRM) or standards traceable to accredited national / international testing laboratories.

8.7 Computer Software Quality Assurance

Computer software is an important tool in internal dosimetry. The software may include commercial dosimetry codes, in-house developed dosimetry codes, calculational algorithms incorporated into commercial application codes (e.g., spreadsheets), and database application software for management, analysis and reporting of data. Quality assurance activities involve configuration management, code testing, error correction, and security.

Dosimetry codes should be subject to configuration management, including records of the version of the code, the user's manual, instructions for running the code, limitations of the code, hardware requirements, acceptance testing records, and a copy of the code itself.

Computer codes should undergo a two-step verification and validation process as acceptance testing before their routine use for dosimetry. Part of the testing should include running selected "benchmark" cases for comparison against an independent solution process (e.g., hand calculations, published tabulations of reference man dose,

results from other verified code, etc). Results of this testing should be maintained with the internal dosimetry laboratory records. This testing should be successfully completed before the code or algorithm is used for dosimetry calculations of workers.

9. DOCUMENTATION AND RECORDS

9.1 General

Record keeping is an essential part of the individual and work place monitoring programme. Dose records should be kept up-to-date and procedures should be established to ensure that assessments of dose from any monitoring period reach the individual's dose record.

The individual occupational exposure record should be uniquely linked to the worker and should enable the appropriate summation of external and internal doses. For each year, the record should comprise:

a) Unique identification of the individual (TLD and / or Employee No.);

b) Exposure for the year to date and, where necessary, for the appropriate five year period;

c) Method used for assessment of internal dose;

d) Information about the material and radionuclides involved in suspected intakes;

e) Lifetime dose to date.

9.2 Record Keeping of Internal Dosimetry Laboratories

The results of in-vivo measurements are expressed interms of retained activity in whole body / organ in Bq with standard uncertainty. Similarly the results of in-vitro samples are expressed as mBq/day with standard uncertainty. Counting time of sample should be chosen in such a way that the degree of uncertainty is not more than 30% for samples near the detection limit and not be more than 10% for samples having activity concentration of 5 times the detection limit. The records to be kept at the monitoring laboratory should include the results of bioassay data (whole body counting, organ / tissue monitoring and excreta analysis) and of PAS sample analyses (if employed). It is essential that type of sample, time of collection, techniques used, measurement results, and details of internal dose estimates are recorded systematically and the records maintained for future reference. These should be maintained in a log book as well as in a computer. The minimum information maintained in the records should include:

a) Worker details (age, weight, TLD No., date of joining the institution etc.) and nature of job;

b) Type of sample submitted and its mass (faeces, tissues etc.) and / or volume (urine, blood etc.) or in vivo monitoring technique used (organ / whole body counting);

c) Documentation and the custody of the sample from collection to analysis and disposal of the sample;

d) Dates and times of in vivo monitoring, dates and times of biological sample collection and analysis, sample mass (faeces, tissues etc.) and / or volume (urine, blood etc.);

e) Analytical procedure(s) employed, analytical instrument / detecting systems used, its corresponding MDA(s) and radiochemical yield of the procedure;

f) Calibration information, gross and background counts in regions of interest;

g) Measured activity of each radionuclide analysed, together with its estimated uncertainty. The uncertainty should specify whether it is due to counting statistics only or to a total propagated error, and in terms of standard deviations (or errors) it represents. All measurement results should be recorded, even if these are below the detection limits;

h) Calculation of final results such as activity content of the body or daily excretion rates and their statistical analyses;

i) Information regarding the solubility type and particle size pertaining to the radioactive contaminant as received from the Consentee / Licensee / RSO;

j) Calculated estimates of intake and the biokinetic models (software model and version) from which they were derived;

k) Estimated doses and the dose coefficient used;

l) Remark on whether or not the result exceeds a specified derived recording or investigation level;

m) Name the radiochemical analyst(s) / scientist and / or the person responsible for the whole body / lung counting;

n) Copies of reports issued to Consentee / Licensee and RSO

The Internal Dosimetry Laboratory should maintain periodic QA audit records.

9.3 Record Keeping of Individual Monitoring and Work place Monitoring by RSO

RSO of the facility should keep the following radiological safety records

a) Radionuclide handled

b) Details of personnel contamination

c) Decontamination procedure followed

d) Levels of air activity concentrations

e) Evaluation of the radiological conditions in all workplaces,

f) Particle size analysis and solubility type

g) Surface contamination level/ nasal swab level etc.

h) Samples submitted for in vivo / in vitro monitoring

9.4 Record Keeping by Consentee / Licensee

The Consentee / Licensee should:

a) Provide any worker information of his own dose records on request;

b) Make the dose data available to

(i) Approved physician whenever a medical investigation/ review is required

(ii) AERB and

(iii) National Occupational Dose Registry System (NODRS).

c) If a worker changes employment, his dose record shall be transferred to his new employer and inform the same to the NODRS.

d) The dose and health records of all the radiation workers shall be maintained by the Consentee / Licensee and the NODRS; and

e) Give due care and attention to the maintenance of appropriate confidentiality of records.

9.5 Retention Periods for Radiological Record

Type of record	Retention period
Workplace monitoring, calibration of survey instrument, calibration of personal monitoring equipment, Individual monitoring labs	5 years
Occupational exposure of worker	During the worker's working life and afterwards at least until the worker attains or would have attained the age of 75 years, and for not less than 30 years after cessation of work involving occupational exposure

9.6 Reporting Information

9.6.1 Reporting by Internal Dosimetry Laboratories:

The required dose reports are sent to concerned authority for the purpose of implementation of radiation protection programmes and maintaining of dose records. Copy of the results should be sent to Consentee / Licensee and RSO of the facility.

The report should contain the following information.

a) Identification of the worker and his TLD No.

b) Techniques used for monitoring

c) Radionuclides analyzed

d) Final result and it's uncertainty

e) If measured activity content in the worker or bioassay sample is less than MDA of the system then routine and special monitoring result should be reported as below detection level (BDL) with writing MDA equivalent activity in the footnote

f) Whenever an assessment of internal exposure is made that is found to be below the recording level, indication to that effect shall be entered in the dose records with the mark, 'below recording level'

g) Quote appropriate references from the Consentee / Licensee regarding the input parameters used for dose computation

h) The dose report should specify the methodology, the biokinetic and dosimetric model or the software used for internal dose computation

i) Internal dose due to individual radionuclides as well as sum of total CED receivable

j) Recommendations on follow-up individual monitoring, if necessary

k) Identification of the individual providing the report

9.6.2 Reporting by RSO

RSO shall furnish reports to AERB on cases exceeding investigation levels with details of

a) Incident information,

b) Details of personnel contamination

c) Medical intervention

d) Workplace restrictions, if any

9.6.3 Reporting by the Consentee / Licensee

All installations should maintain appropriate documentation on the various aspects described in this guide. They should be updated and reviewed periodically. Appropriate information (as mentioned in Section 9.4) should be communicated to the AERB and the NODRS.Documents should be easily identifiable, retrievable and should include time periods for which the relevant information is recorded.

9.7 Training of personnel

9.7.1 Training of workers

It is the consentee's / licensee's responsibility to ensure that workers occupationally exposed to radiation and persons with assigned responsibilities in the Radiation Protection Programme receive general radiation protection information and training. Such training programme should also include internal exposure control and assessment procedures.

9.7.2 Training of dosimetry service personnel

Training should include on the following topics:

a) Basic philosophy and strategy of internal dose assessment;

b) Principles and details of the methods and procedures used, and their limitations;

c) Technical details of the processes;

d) Importance of their work;

e) Reporting method; and

f) Quality assurance programme and its objectives,

g) Decorporation treatment and follow-up monitoring.

Annexure I

IMPORTANT INTERNAL DOSIMETRY INFORMATION FOR SELECTED RADIONUCLIDES

ICRP Metabolic and Biokinetic Models

The currently recommended systemic models for the radioelements are of two different types:

1. Non-recycling compartments ("metabolic" models) that were introduced in ICRP Publication 30 [8].
2. Recycling compartments ("biokinetic" models) that were introduced in ICRP Publication 56 [9] onwards.

A few of these models for important radionuclides are described in the following sections. In addition to these, ICRP Publication 78 [2] used the biokinetic model for the urinary bladder used earlier in Publication 67 [11] and Publication 68 [34] to evaluate urinary excretion, assuming number of urine voids per day as 6 and the rate of elimination from the urinary bladder (content) to be 12 d^{-1}.

1. Hydrogen (Tritium)

ICRP currently recommends three metabolic models for tritium (T): (1) for tritiated water (HTO), (2) for dosimetry of organically bound tritium (OBT), adopted in ICRP Publication 67[11], and (3) for bioassay of OBT (adopted in ICRP Publication 78, [2]. The models are shown in Figures 1.1 to 1.3, respectively.

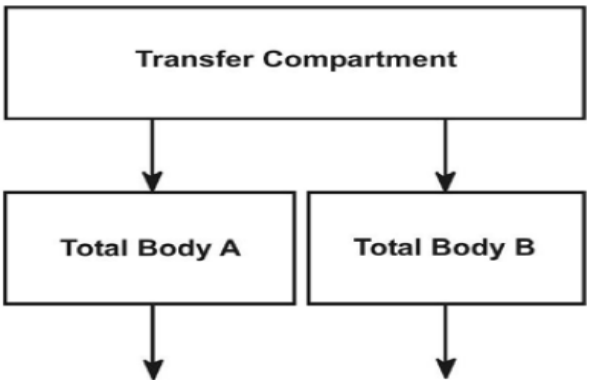


Figure 1.1 ICRP *Publication* 67 [11] metabolic model for tritiated water (HTO).

It is assumed that tritiated water (HTO) equilibrates rapidly with all body water. Thus its concentration is uniform throughout the body. ICRP Publication 67 [11] recommended the values of fractional uptake and transfer rates shown in Table 1.1, for substitution in the tritiated water (HTO) metabolic model (Figure 1.1).

Table 1.1 Retention half-times and fractional uptake for tritiated water (HTO).

Retention Compartment	Fractional Uptake	Half-time (d)
Transfer compartment (Blood)	-	0.25
Total Body A	97%	10
Total Body B	3%	40

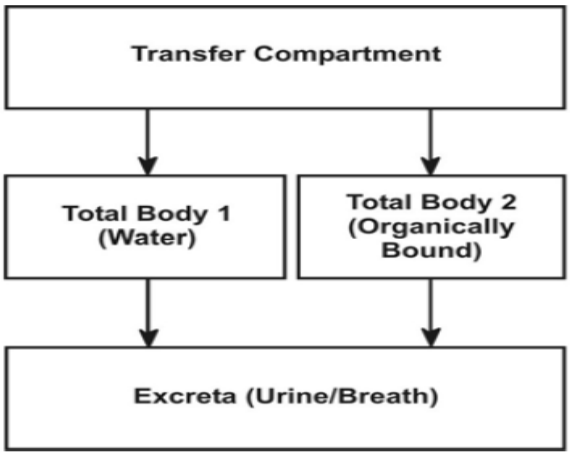


Figure 1.2 ICRP Publication 67 [11]metabolic model for organically bound tritium (OBT).

For organically bound tritium (OBT), it is assumed that half equilibrates rapidly with all body water. The other half is assumed to be metabolized as organic carbon. This is retained longer in the body. ICRP Publication 67 [11] recommended the values of fractional uptake and transfer rates shown in Table 1.2 for substitution in the organically bound tritium (OBT) metabolic model (Figure 1.2).

Table 1.2 Retention half-times and fractional uptake for organically bound tritium (OBT).

Retention Compartment	Fractional Uptake	Half-time (d)
Transfer compartment (Blood)	-	0.25
Total Body 1 (Body water)	50%	10
Total Body 2 (Organic)	50%	40

Assumed metabolism of organically bound tritium (OBT) – for bioassay:

For OBT, the ICRP revised its treatment of urinary excretion . This is now treated explicitly in the metabolic model, by considering that approximately half (14/30) the excreted body water is excreted in the urine (and also, therefore, approximately half the OBT in equilibrium with the body water). The remaining fraction (16/30) is excreted in the breath.

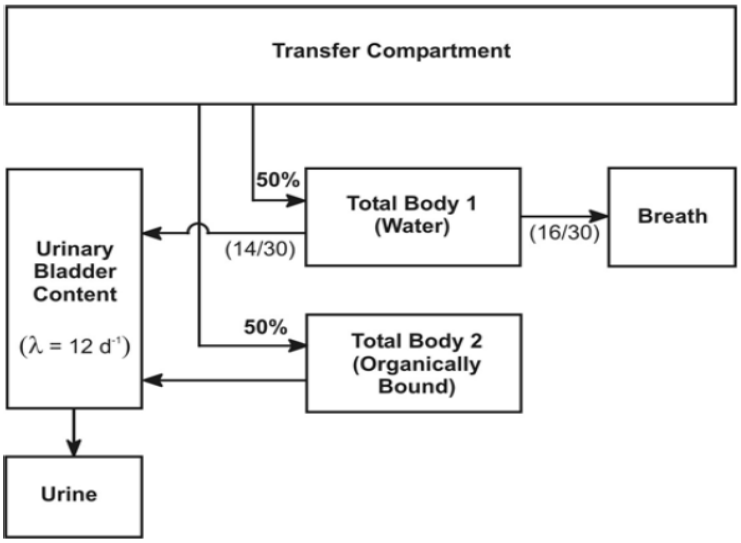


Figure 1.3 ICRP Publication 78 [2] metabolic; model for organically bound tritium (OBT).

Table 1.3 Compounds, absorption types and f_1 values

	f_1	**Intake**
Ingestion	1.0	Ingestion of tritiated water or organically bound tritium
Vapour	[a]	Inhalation of tritiated water vapour
Gas	[a]	Inhalation of tritium gas

[a]Not applicable since all activity deposited in the respiratory tract is instantaneously absorbed

Table 1.4 Emissions

Radiation	**Average energy (MeV)**	**Intensity (%)**
β	0.0057	100

Table 1.5 Measurement techniques

Method of measurement		Typical detection limit
Liquid scintillation counting	Urine	50 Bq L^{-1}

Tritium presents no detection problems.

Table 1.6 Dose coefficients (Sv Bq^{-1}) for intake of tritiated compounds via inhalation / ingestion [2]

Chemical form	$t_{1/2}$	Type	**Inhalation**		**Ingestion**	
			f_1	e(50)	f_1	e(50)
Tritiated water	12.3 y	SR-2	a	1.8×10^{-11}	1.0	1.8×10^{-11}
Tritium gas	12.3 y	SR-2	a	1.8×10^{-15}	----	----
OBT[b]	12.3 y	SR-2	a	4.1×10^{-11}	1.0	4.2×10^{-11}

[a]Not applicable since all activity deposited in the respiratory tract is instantaneously absorbed.

[b]OBT –Organically bound tritium.

Table 1.7 Dose coefficients (Sv Bq^{-1}) for intake of tritiated compounds via a contaminated wound for all wound modelcategories [14].

Radio-nuclide	Weak	Moderate	Strong	Avid	Colloid	Particle	Fragment
^{3}H (HTO)[1]	1.84×10^{-11}	1.84×10^{-11}	1.78×10^{-11}	1.61×10^{-11}	1.56×10^{-11}	1.01×10^{-11}	4.07×10^{-13}
^{3}H (OBT)[2]	4.16×10^{-11}	4.16×10^{-11}	4.01×10^{-11}	3.64×10^{-11}	3.52×10^{-11}	2.28×10^{-11}	9.19×10^{-13}

[1] tritiated water

[2] organically bound tritium

2. Cobalt

The currently recommended biokinetic model for cobalt was adopted in ICRP Publication 67 [11].This is shown in Figure 2.1. Following entry into the transfer compartment, 50% of the cobalt is rapidly excreted with a biological half-life of 0.5 day, 5% is taken up by the liver and 45% is uniformly distributed in all other tissues. Fractions of 0.6, 0.2 and 0.2 are assumed to be lost from the liver and other tissues with biological half-lives of 6, 60 and 800 days, respectively. For activity lost to excretion from systemic compartments, 86% is assumed to be lost to urine and 14% to faeces. ICRP Publication 67 [11] recommended the values of fractional uptake and transfer rates shown in Table 2.1, for substitution in the cobalt metabolic model (Figure 2.1).

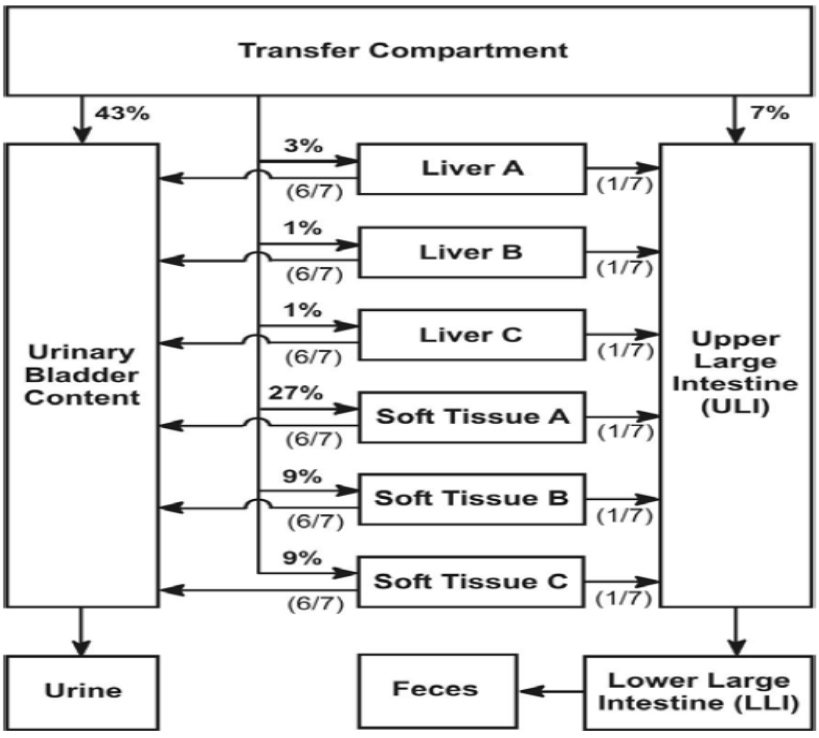


Figure 2.1 ICRP Publication 67 [11]metabolic model for cobalt.

The urinary to faecal excretion ratio for cobalt is 6:1 [11]. A large fraction (50%) of excretion is assumed to occur directly from the transfer compartment.

Table 2.1 Retention half-times and fractional uptake for cobalt metabolic model.

Retention Compartment	Fractional Uptake	Half-time (d)
Transfer compartment (blood)	-	0.5
Liver A	3%	6
Liver B	1%	60
Liver C	1%	800
Soft tissues (other) A	27%	6
Soft tissues (other) B	9%	60
Soft tissues (other) C	9%	800
Direct to Urinary bladder content	(6/7) × 50%	-
Direct to Upper large intestine	(1/7) × 50%	-

Table 2.2 Compounds, absorption types and f_1 values

Intake	f_1	Compound
Ingestion	0.1	Unspecified compounds
Ingestion	0.05	Oxides, hydroxides and inorganic compounds
Inhalation, Type M	0.1	Unspecified compounds
Inhalation, Type S	0.05	Oxides, hydroxides, halides and nitrates

Cobalt-57(half- life =271 d)

Table 2.3 Emissions

Radiation	Energy (MeV)	Intensity (%)
γ	0.122	85.6
γ	0.137	10.6

Table 2.4 Measurement techniques

Method of measurement		Typical detection limit
γ–ray spectrometry in vivo	Whole body	100 Bq
γ–ray spectrometry on biological samples	Urine	1 Bq l^{-1}
	faeces	1 Bq per sample

Cobalt -57 presents no detection problems.

Cobalt -58 (half-life =70.8 d)

Table 2.5 Emissions

Radiation	Energy (MeV)	Intensity (%)
γ	0.511	30
γ	0.811	99

Table 2.6 Measurement techniques

Method of measurement		Typical detection limit
γ–ray spectrometry in vivo	Whole body	100 Bq
γ–ray spectrometry on biological samples	Urine	1 Bq l^{-1}
	Faeces	1 Bq per sample

Cobalt -58 presents no detection problems.

Cobalt -60(half-life = 5.27 y)

Table 2.7 Emissions

Radiation	Energy (MeV)	Intensity (%)
γ	1.17	100
γ	1.33	100

Table 2.8 Measurement techniques

Method of measurement		Typical detection limit
γ –ray spectrometry in vivo	Whole body	100 Bq
γ–ray spectrometry on biological samples	Urine	1 Bq l^{-1}
	Faeces	1 Bq per sample

Cobalt -60 presents no detection problems.

Table 2.9 Dose coefficients ($SvBq^{-1}$**)** for intake of cobalt via inhalation / ingestion [2]

Nuclide	$t_{1/2}$	Type	Inhalation		Ingestion	
			f_1	e(50)	f_1	e(50)
^{57}Co	271 d	M	0.1	3.9×10^{-10}	0.1	2.1×10^{-10}
		S	0.05	6.0×10^{-10}	0.05	1.9×10^{-10}
^{58}Co	70.8 d	M	0.1	1.4×10^{-9}	0.1	7.4×10^{-10}
		S	0.05	1.7×10^{-9}	0.05	7.0×10^{-10}
^{60}Co	5.27 y	M	0.1	7.1×10^{-9}	0.1	3.4×10^{-9}
		S	0.05	1.7×10^{-8}	0.05	2.5×10^{-9}

Table 2.10 Dose coefficients ($SvBq^{-1}$) for intake of cobalt via a contaminated wound for all wound model categories[14].

Radio-nuclide	Weak	Moderate	Strong	Avid	Colloid	Particle	Fragment
^{57}Co	6.36×10^{-10}	6.28×10^{-10}	5.16×10^{-10}	2.59×10^{-10}	1.84×10^{-10}	2.45×10^{-11}	3.00×10^{-12}
^{58}Co	1.53×10^{-9}	1.47×10^{-9}	1.10×10^{-9}	4.01×10^{-10}	2.03×10^{-10}	1.46×10^{-11}	3.07×10^{-12}
^{60}Co	1.94×10^{-8}	1.94×10^{-8}	1.81×10^{-8}	1.49×10^{-8}	1.38×10^{-8}	6.11×10^{-9}	2.47×10^{-10}

3. Strontium

The currently recommended biokinetic model for the alkaline earth elements (including strontium) was adopted in ICRP Publication 67 [11]. This is shown in Figure 3.1 [2]. This model describes in detail the kinetics of alkaline earth elements in bone, which is the main site of deposition and retention, and considers also retention in soft tissues and routes of excretion. It takes account of initial uptake on to bone surfaces, transfer from the surface to bone volume and recycling from bone and soft tissues to blood. It also describes the excretion routes for which no constant ratio is used.

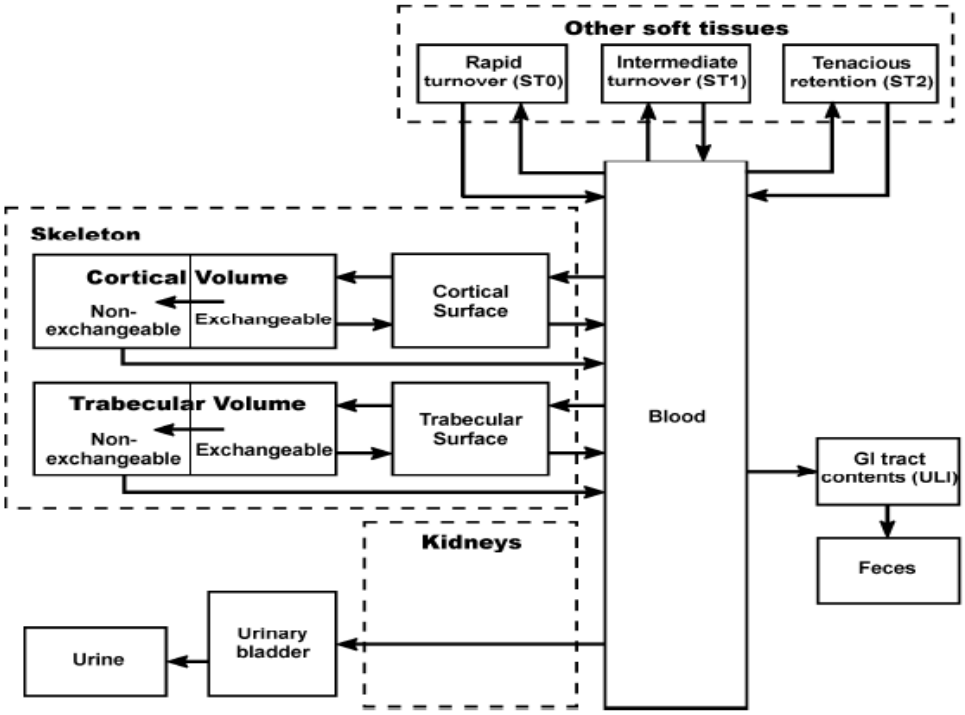


Figure 3.1 ICRP Publication 67 [11] biokinetic model for the alkaline earth elements (calcium, strontium, barium and radium).

ICRP Publication 67 [11] recommended the values of transfer rates shown in Table 3.1 for substitution in the strontium biokinetic model (Figure 3.1).

Table 3.1 Transfer rates for strontium biokinetic model.

Route of transfer between compartments	Transfer rate (d^{-1})
Plasma to ST0	7.5
Plasma to Urinary bladder content	1.73
Plasma to ULI content	0.525
Plasma to ST1	1.50
Plasma to ST2	0.003

Plasma to Trabecular surfaces	2.08
Plasma to Cortical surfaces	1.67
ST0 to Plasma	2.50
ST1 to Plasma	0.116
ST2 to Plasma	0.000380
Bone surfaces to Plasma	0.578
Non-exchangeable trabecular volume to Plasma	0.000493
Non-exchangeable cortical volume to Plasma	0.0000821
Bone surfaces to Exchangeable volume	0.116
Exchangeable bone volume to Bone surfaces	0.00430
Exchangeable bone volume to Non-exchangeable volume	0.00430

Table 3.2 Compounds, absorption types and f_1 values

Intake	**f_1**	**Compounds**
Ingestion	0.3	Unspecified compounds
Ingestion	0.01	Strontium titanate ($SrTiO_3$)
Inhalation, Type F	0.3	Unspecified compounds
Inhalation, Type S	0.01	Strontium titanate ($SrTiO_3$)

Strontium -85 (half-life = 64.8 d)

Table 3.3 Emissions

Radiation	**Energy (MeV)**	**Intensity (%)**
γ	0.51	98

Table 3.4 Measurement techniques

Method of measurement		**Typical detection limit**
γ –ray spectrometry in vivo	Whole body	150 Bq
γ –ray spectrometry on biological samples	Urine	10 Bq d^{-1}

Strontium -89 (half-life = 50.5 d)

Table 3.5 Emissions

Radiation	**Average energy (MeV)**	**Intensity (%)**
β^-	0.58	100

Table 3.6 Measurement techniques

Method of measurement		**Typical detection limit**
β^- counting following chemical separation	Urine	0.015 Bq d^{-1}

Strontium-89 presents no detection problems.

Strontium -90 (half –life = 29.1 y) Yttrium -90 (half –life = 64 h)

Table 3.7 Emissions

Radionuclide	Radiation	Average energy (MeV)	Intensity (%)
^{90}Sr	β^-	0.20	100
^{90}Y	β^-	0.99	100

Table 3.8 Measurement techniques

Method of measurement		Typical detection limit
β^- counting following chemical separation	Urine	0.015 Bq d^{-1}

Table 3.9 Dose coefficients($SvBq^{-1}$)for intake of strontium via inhalation / ingestion [2]

Nuclide	$t_{1/2}$	Type	Inhalation		Ingestion	
			f_1	e(50)	f_1	e(50)
^{85}Sr	64.8 d	F	0.3	5.6 x 10^{-10}	0.3	5.6 x 10^{-10}
		S	0.01	6.4 x 10^{-10}	0.01	3.3x 10^{-10}
^{89}Sr	50.5 d	F	0.3	1.4 x 10^{-9}	0.3	2.6 x 10^{-9}
		S	0.01	5.6 x 10^{-9}	0.01	2.3 x 10^{-9}
^{90}Sr	29.1 y	F	0.3	3.0 x 10^{-8}	0.3	2.8 x 10^{-8}
		S	0.01	7.7 x 10^{-8}	0.01	2.7 x 10^{-9}

Table 3.10 Dose coefficients ($SvBq^{-1}$) for intake of strontium via a contaminated wound for all wound model categories[14].

Radio-nuclide	Weak	Moderate	Strong	Avid	Colloid	Particle	Fragment
^{85}Sr	1.10 ×10^{-9}	1.06×10^{-9}	7.86×10^{-10}	2.83× 10^{-10}	1.39×10^{-10}	9.77×10^{-12}	2.05×10^{-12}
^{89}Sr	3.13 ×10^{-9}	2.98×10^{-9}	2.17×10^{-9}	7.61× 10^{-10}	3.43×10^{-10}	2.24×10^{-11}	4.59×10^{-12}
^{90}Sr	8.80 ×10^{-8}	8.81×10^{-8}	8.66×10^{-8}	8.26× 10^{-8}	8.22×10^{-8}	6.57×10^{-8}	2.87×10^{-9}

4. Iodine

The currently recommended biokinetic model for iodine was adopted in ICRP Publication 67 [11]. This is shown in Figure 4.1. ICRP Publication 78 [2] recommended the values of transfer rates shown in Table 4.1, for substitution in the iodine biokinetic model (Figure 4.1). It is assumed that, of the iodine reaching the blood, a fraction of 0.3 is accumulated in the thyroid gland and 0.7 is excreted directly

in urine. The biological half-life in blood is taken to be 0.25 day. Iodine incorporated into thyroid hormones leaves the thyroid gland with a biological half-life of 80 days and enters other tissues, where it is retained with a biological half-life of 12 days. Most iodine (80%) is subsequently released to the blood and is available in the circulation for uptake by the thyroid gland and urinary excretion; the remainder (20%) is excreted in faeces in organic form.

Pathological states of the thyroid may result in uptake values of 0–0.05 (blocked thyroid) to more than 0.5. When such cases are suspected, then individual values need to be introduced in the dose calculation, especially for accidental exposures leading to significant doses, for which a precise assessment is needed.

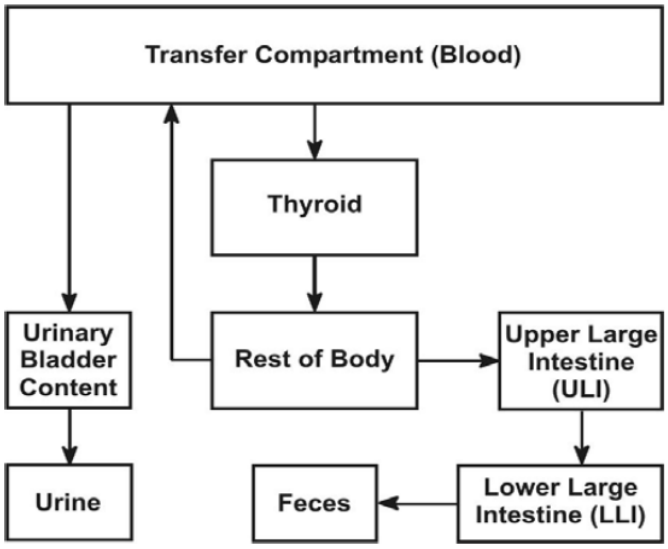


Figure 4.1 ICRP Publication 67 [11] biokinetic model for iodine.

Table 4.1 Transfer rates for iodine biokinetic model.

Route of transfer between compartments	**Transfer rate (d^{-1})**
Blood to Thyroid	0.83178
Blood to Urinary bladder content	0.19408
Thyroid to Rest of body	0.0086643
Rest of body to Blood	0.04621
Rest of body to Upper large intestine content	0.01155

Table 4.2 Compounds, absorption types and f_1 values

Intake	**f_1**	**Compounds**
Ingestion	1.0	All compounds
Inhalation, class SR-1	1.0	Iodine vapour
Inhalation, Type F	1.0	All other compounds

Iodine-125 (half life = 60.1 d)

Table 4.3 Emissions

Radiation	Energy (MeV)	Intensity (%)
χ	0.027-0.032	140
γ	0.035	6.7

Table 4.4 Measurement techniques

Method of measurement		**Typical detection limit**
Photon spectrometry in vivo	Thyroid	25 Bq

Iodine-129 (half life = 1.57E+07 y)

Table 4.5 Emissions

Radiation	**Energy (MeV)**	**Intensity (%)**
β^-	0.049[a]	100
χ	0.030-0.034	70
γ	0.040	7.5

[a]Average energy

Table 4.6 Measurement techniques

Method of measurement		**Typical detection limit**
Photon spectrometry in vivo	Thyroid	25 Bq

Iodine-131 (half life = 8.04d)

Table 4.7 Emissions

Radiation	**Energy (MeV)**	**Intensity (%)**
β^-	0.019	89
γ	0.36	81

Table 4.8 Measurement techniques

Method of measurement		**Typical detection limit**
γ-ray spectrometry in vivo	Thyroid	100 Bq
γ-ray spectrometry on biological sample	Urine	1 Bq d^{-1}

I-131 presents no detection problems.

Table 4.9 Dose coefficients ($SvBq^{-1}$)for intake of iodine via inhalation / ingestion [2]

Nuclide	**$t_{1/2}$**	**Type**	**Class**	**Inhalation**		**Ingestion**	
				f_1	**e(50)**	**f_1**	**e(50)**
^{125}I	60.1 d	F	-	1.0	7.3 x 10^{-9}	1.0	1.5x 10^{-8}
		F	SR-1[a]	1.0	1.4 x 10^{-8}		
^{129}I	1.57E07 y	F	-	1.0	5.1 x 10^{-8}	1.0	1.1 x 10^{-7}
		F	SR-1	1.0	9.6 x 10^{-8}		

^{131}I	8.04 d	F	-	1.0	1.1×10^{-8}	1.0	2.2×10^{-8}
		F	SR-1	1.0	2.0×10^{-8}		

[a] The model for iodine vapour is described in ICRP Publication 68 [41].

Table 4.10 Effective dose coefficients ($SvBq^{-1}$) for intake of iodine via a contaminated wound for all wound model categories[14].

Radio-nuclide	Weak	Moderate	Strong	Avid	Colloid	Particle	Fragment
^{125}I	1.54×10^{-8}	1.48×10^{-8}	1.09×10^{-8}	3.90×10^{-9}	1.87×10^{-9}	1.28×10^{-10}	2.68×10^{-11}
^{129}I	1.07×10^{-7}	1.07×10^{-7}	1.06×10^{-6}	1.05×10^{-7}	1.07×10^{-7}	1.06×10^{-7}	5.87×10^{-9}
^{131}I	2.13×10^{-8}	1.88×10^{-8}	1.19×10^{-8}	4.35×10^{-9}	7.36×10^{-10}	3.85×10^{-11}	2.45×10^{-12}

5. Caesium

The currently recommended metabolic model for caesium was adopted in ICRP Publication 67 [11]. This is shown in Figure 5.1 [2]. Caesium is assumed to be uniformly distributed throughout all organs and tissues of the body. Therefore, a classification of isotopes of the element for the purposes of bone dosimetry is not required [8]. The SEEs are calculated for "Whole Body" as the source. The recommended fractions of caesium eliminated from the total body compartments that are excreted via urine and faeces are 80% and 20% respectively. There is no direct excretion of caesium from the transfer compartment to urine or faeces.

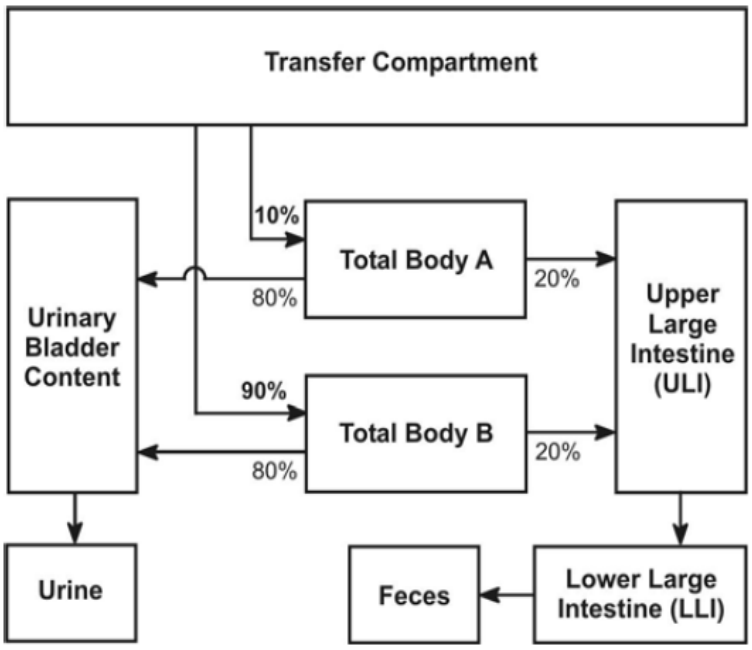


Figure 5.1ICRP Publication 67 [11]metabolic model for caesium.

ICRP Publication 67 [11] recommended the values of fractional uptake and transfer rates shown in Table 5.1, for substitution in the caesium metabolic model (Figure 5.1).

Table 5.1 Retention half-times and fractional uptake for caesium metabolic model.

Retention Compartment	Fractional Uptake	Half-time (d)
Transfer compartment (blood)	-	0.25
Total Body A	10%	2
Total Body B	90%	110

According to ICRP-78 [2], following entry into the transfer compartment, caesium is taken to be distributed uniformly throughout all body tissues; 10% of the activity is assumed to be retained with a biological half-life of two days and 90% with 110 days. A urinary to faecal ratio of 4:1 is recommended.

Table 5.2 Compounds, absorption types and f_1 values

Intake	f_1	Compounds
Ingestion	1.0	All compounds
Inhalation, Type F	1.0	All compounds

Caesium-134 (half life = 2.06 y)

Table 5.3 Emissions

Radiation	Energy (MeV)	Intensity (%)
γ	0.60	98
γ	0.80	85

Table 5.4 Measurement techniques

Method of measurement		Typical detection limit
γ-ray spectrometry in vivo	Whole body	200 Bq
γ-ray spectrometry on biological sample	Urine	1 Bq d^{-1}

^{134}Cs presents no detection problems

Caesium-137 (half life = 30.0 y)

Table 5.5 Emissions

Radiation	Energy (MeV)	Intensity (%)
γ (^{137m}Ba)	0.661	85.1

Table 5.6 Measurement techniques

Method of measurement		Typical detection limit
γ-ray spectrometry in vivo	Whole body	200 Bq
γ-ray spectrometry on biological sample	Urine	1 Bq d^{-1}

Cs-137 presents no detection problems.

Table 5.7 Dose coefficients ($SvBq^{-1}$) for intake of caesium via inhalation / ingestion [2]

Nuclide	$t_{1/2}$	Type	Inhalation		Ingestion	
			f_1	**e(50),**	f_1	e(50)
^{134}Cs	2.06 y	F	1.0	9.6×10^{-9}	1.0	1.9×10^{-8}
^{137}Cs	30.0 y	F	1.0	6.7×10^{-9}	1.0	1.3×10^{-8}

Table 5.8 Dose coefficients ($SvBq^{-1}$) for intake of caesium via a contaminated wound for all wound model categories[14]

Radio-nuclide	Weak	Moderate	Strong	Avid	Colloid	Particle	Fragment
^{134}Cs	1.94×10^{-8}	1.94×10^{-8}	1.71×10^{-8}	1.16×10^{-8}	9.70×10^{-9}	2.45×10^{-9}	1.48×10^{-10}
^{137}Cs	1.36×10^{-8}	1.36×10^{-8}	1.34×10^{-8}	1.28×10^{-8}	1.27×10^{-8}	1.03×10^{-8}	4.77×10^{-10}

6. Polonium

The currently recommended biokinetic model for polonium was adopted in ICRP Publication 67 [11]. This is shown in Figure 6.1. ICRP Publication 67 [11] recommended the values of fractional uptake and transfer rates shown in Table 6.1, for substitution in the polonium metabolic model (Figure 6.1).

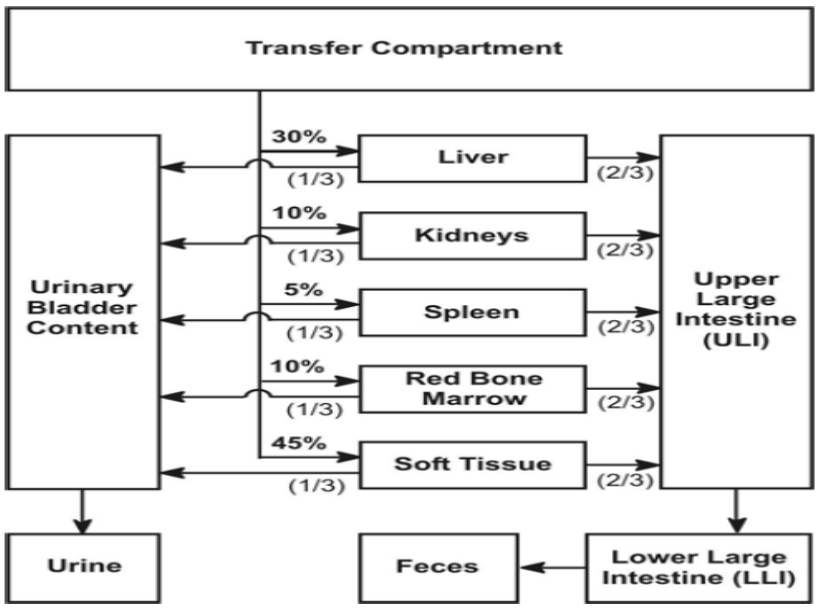


Figure 6.1 ICRP Publication 67 [11] metabolic model for polonium.

Table 6.1 Retention half-times and fractional uptake for polonium metabolic model.

Retention Compartment	Fractional Uptake	Half-time (d)
Transfer compartment (blood)	-	0.25
Liver	30%	50
Kidneys	10%	50
Spleen	5%	50

Red bone marrow	10%	50
Soft tissues (other)	45%	50

For bone dosimetry, it is assumed that all polonium in the skeleton is associated with the bone marrow. The fraction of polonium excretion via urine, $\mathbf{f_u}$, is one-third [11]. There is no direct excretion of polonium from the transfer compartment to urine or faeces.

Table 6.2 Compounds, absorption types and f_1 values

Intake	**f_1**	**Compounds**
Ingestion	0.1	All compounds
Inhalation, Type F	0.1	Unspecified compounds
Inhalation, Type M	0.1	Oxides, hydroxides and nitrates

Polonium-210 (half life = 138.4 d)

Table 6.3 Emissions

Radiation	**Energy (MeV)**	**Intensity (%)**
α	5.3	100
γ	0.80	1.6 E-3

Table 6.4 Measurement techniques

Method of measurement		**Typical detection limit**
α-ray spectrometry on biological sample	Urine	0.5 mBq d^{-1}

Table 6.5 Dose coefficients($SvBq^{-1}$) for intake of polonium via inhalation / ingestion [2]

Nuclide	**$t_{1/2}$**	**Type**	**Inhalation**		**Ingestion**	
			f_1	**e(50)**	**f_1**	**e(50)**
^{210}Po	138.4 d	F	0.1	7.1×10^{-7}	0.1	2.4×10^{-7}
		M	0.1	2.2×10^{-6}		

Table 6.6 Dose coefficients ($SvBq^{-1}$) for intake of polonium via a contaminated wound for all wound model categories [14]

Radio-nuclide	**Weak**	**Moderate**	**Strong**	**Avid**	**Colloid**	**Particle**	**Fragment**
^{210}Po	2.36×10^{-6}	2.31×10^{-6}	1.81×10^{-6}	7.52×10^{-7}	4.60×10^{-7}	4.31×10^{-8}	7.72×10^{-9}

7. Radium

The currently recommended biokinetic model for the alkaline earth elements (strontium, barium, and radium) was adopted in ICRP Publication 67 [11]. This is

shown in Figure 7.1 [2]. ICRP Publication 67 [11] recommended the values of transfer rates shown in Table 7.1, for substitution in the radium biokinetic model (Figure 7.1). The model describes the kinetics of radium in bone, which is the main site of deposition and retention, and also considers retention in liver and other soft tissues as well as routes of excretion. It takes account of initial uptake onto bone surfaces, transfer from surface to bone volume and recycling from bone and other tissues to plasma.

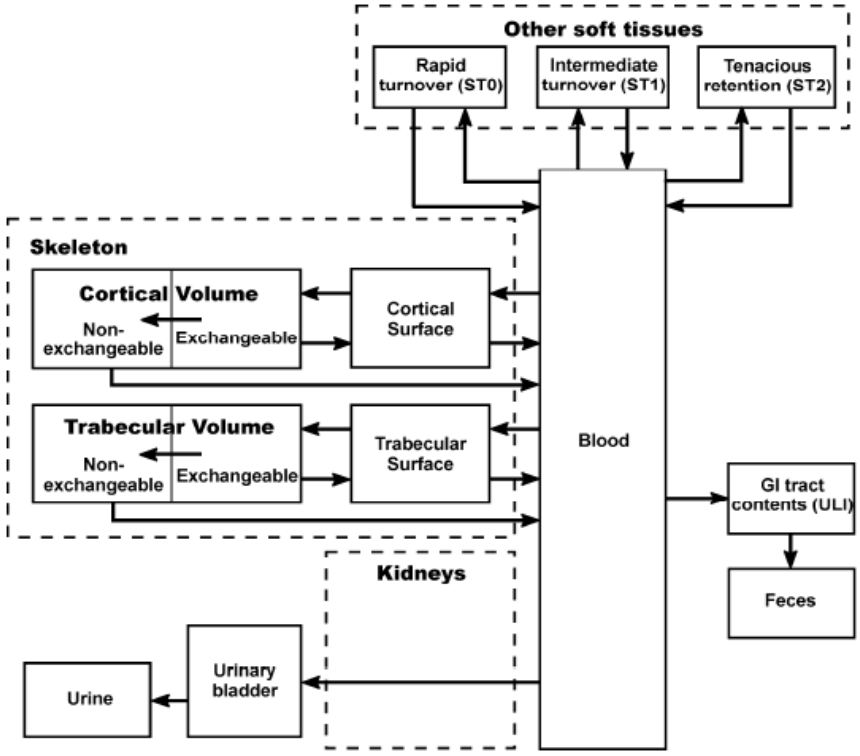


Figure 7.1 ICRP Publication 67[11]biokinetic model for the alkaline earth elements (calcium, strontium, barium, and radium).

Table 7.1 Transfer rates for radium biokinetic model.[(a)]

Route of transfer between compartments	**Transfer rate (d^{-1})**
Plasma to ST0	22.68
Plasma to Urinary bladder content	0.606
Plasma to ULI content	21.79
Plasma to ST1	7.0
Plasma to ST2	0.07
Plasma to Trabecular bone surface	9.72
Plasma to Cortical bone surface	7.78
Plasma to Liver	0.35
Liver to Plasma	0.0139
ST0 to Plasma	7.56
ST1 to Plasma	0.693
ST2 to Plasma	0.00038
Bone surfaces to Plasma	0.578
Non-exchangeable trabecular volume to Plasma	0.000493

Non-exchangeable cortical volume to Plasma	0.0000821
Bone surfaces to Exchangeable bone volume	0.116
Exchangeable bone volume to Non-exchangeable bone volume	0.0046
Exchangeable bone volume to Bone surfaces	0.0185

[(a)]ICRP Publication 78 [2] notes that the values of rate constants are given to sufficient precision [3 significant figures] for calculational purposes and that this may be more precise than the biological data would support.

Table 7.2 Compounds, absorption types and f_1 values

Intake	f_1	Compounds
Ingestion	0.2	All compounds
Inhalation, Type M	0.2	All compounds

Radium-226 (half life = 1.6E03 y)

Table 7.3 Emissions

Radionuclide	Radiation	Energy (MeV)	Intensity (%)
^{226}Ra	α	4.6	6
	α	4.8	94
	γ	0.19	3
^{214}Pb	γ	0.30	19
	γ	0.35	37
^{214}Bi	γ	0.61	46
	γ	1.12	15
	γ	1.76	16

Table 7.4 Measurement techniques

Method of measurement		Typical detection limit
γ-ray spectrometry in vivo	Whole body	1000 Bq
Radiochemical separation and α-spectrometry on biological sample	Urine	0.5 mBq d^{-1}

Radium-228 (half life = 5.75 y)

Table 7.5 Emissions

Radionuclide	Radiation	Energy (MeV)	Intensity (%)
^{228}Ra	β^-	0.01[a]	100
^{228}Ac	γ	0.34	16
	γ	0.91	29
	γ	0.96	23
^{212}Pb	γ	0.24	45

^{208}Tl	γ	0.51	22[b]
	γ	0.58	86[b]
	γ	0.86	12[b]
	γ	2.61	100[b]

[a]Average energy

[b]These intensities refer to one disintegration of ^{208}Tl, the decay of ^{228}Ra to ^{208}Tl involves a branching fraction of 0.36. This should be taken into account when relating these intensities to one disintegration of ^{228}Ra.

Table 7.6 Measurement techniques

Method of measurement		**Typical detection limit**
γ-ray spectrometry in vivo by means of decay products	Whole body	200 Bq
Radiochemical separation and β - counting	Urine	1 Bq L^{-1}

Table 7.7 Dose coefficients ($SvBq^{-1}$) for intake of radium via inhalation / ingestion [2]

Nuclide	**$t_{1/2}$**	**Type**	**Inhalation**		**Ingestion**	
			f_1	**e(50)**	**f_1**	**e(50)**
^{226}Ra	1.6 x 10^3 y	M	0.2	2.2 x 10^{-6}	0.2	2.8 x 10^{-7}
^{228}Ra	5.75 y	M	0.2	1.7 x 10^{-6}	0.2	6.7x 10^{-7}

Table 7.8 Dose coefficients ($SvBq^{-1}$) for intake of radium via a contaminated wound for all wound model categories[14]

Radio-nuclide	**Weak**	**Moderate**	**Strong**	**Avid**	**Colloid**	**Particle**	**Fragment**
^{226}Ra	1.36 ×10^{-6}	1.31×10^{-6}	1.44×10^{-6}	1.62×10^{-6}	1.77×10^{-6}	2.64×10^{-6}	1.65×10^{-7}
^{228}Ra	3.37 ×10^{-6}	3.96×10^{-6}	1.41×10^{-5}	3.81×10^{-5}	4.56×10^{-5}	4.04×10^{-5}	1.13×10^{-6}

8. Thorium

The currently recommended biokinetic model for the actinide elements (thorium, neptunium, plutonium, americium, and curium) was adopted in ICRP Publication 67 [11]. This is shown in Figure 8.1 [2]. For thorium absorbed to blood main sites of deposition are liver and skeleton. ICRP Publication 78 [2] recommended the values of transfer rates shown in Table 8.1 for substitution in the thorium biokinetic model. The model takes account of the initial deposition in bone, liver, gonads, and other tissues, and allows for transfer of activity from bone surfaces to bone volume and marrow, recycling of activity between tissues, as well as loss by excretion.

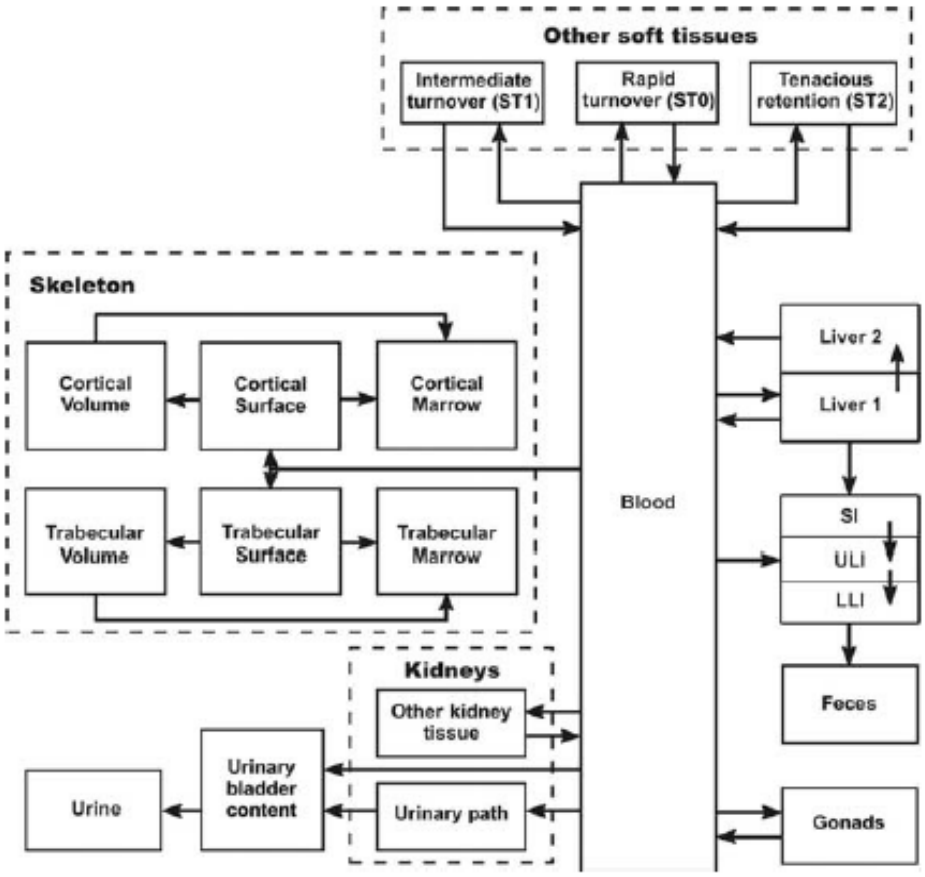


Figure 8.1 ICRP Publication 67 [11] biokinetic model for thorium.

ICRP Publication 78 [2] recommended the values of transfer rates shown in Table 8.1, for substitution in the thorium biokinetic model (Figure 8.1).

Table 8.1 Transfer rates for thorium biokinetic model.[(a)]

Route of transfer between compartments	**Transfer rate (d^{-1})**
Blood to ST0	0.832
Blood to ST1	0.243
Blood to ST2	0.0388
Blood to Trabecular surface	0.6793
Blood to Cortical surface	0.6793
Blood to Urinary bladder content	0.1067
Blood to Kidneys (urinary path)	0.0679
Blood to Kidneys (other tissue)	0.0194
Blood to ULI content	0.0097
Blood to Liver 1	0.097
Blood to Testes	0.00068
Blood to Ovaries	0.00021
ST0 to Blood	0.462
ST1 to Blood	0.00095
ST2 to Blood	0.000019

Liver 2 to Blood	0.000211
Liver 1 to Liver 2	0.00095
Liver 1 to Blood	0.000475
Liver 1 to Small intestine content (SI)	0.000475
Kidneys (other tissue) to Blood	0.00038
Kidneys (urinary path) to Urinary bladder content	0.0462
Trabecular marrow to Blood	0.0076
Cortical marrow to Blood	0.0076
Trabecular surface to Trabecular volume	0.000247
Cortical surface to Cortical volume	0.0000411
Trabecular surface to Trabecular marrow	0.000493
Trabecular volume to Trabecular marrow	0.000493
Cortical surface to Cortical marrow	0.0000821
Cortical volume to Cortical marrow	0.0000821
Testes to Blood	0.00019
Ovaries to Blood	0.00019

[(a)]ICRP Publication 78 [2] notes that the values of rate constants are given to sufficient precision for calculational purposes and that this may be more precise than the biological data would support.

Table 8.2 Compounds, absorption types and f_1 values

Intake	**f_1**	**Compounds**
Ingestion	5.0×10^{-4}	Unspecified compounds
Ingestion	2.0×10^{-4}	Oxides and hydroxides
Inhalation, Type M	5.0×10^{-4}	Unspecified compounds
Inhalation, Type S	2.0×10^{-4}	Oxides and hydroxides

Thorium-228 (half life = 1.91 y)

Table 8.3 Emissions

Radionuclide	**Radiation**	**Energy (MeV)**	**Intensity (%)**
^{228}Th	α	5.3	27
	α	5.4	73
^{212}Pb	γ	0.24	45
^{212}Bi	γ	0.73	12
^{208}Tl	γ	0.51	22[a]
	γ	0.58	86[a]
	γ	0.86	12[a]
	γ	2.6	100[a]

[a]These intensities refer to one disintegration of ^{208}Tl, the decay of ^{228}Th to ^{208}Tl involves a branching fraction of 0.36. This should be taken into account when relating these intensities to one disintegration of ^{228}Th.

Table 8.4 Measurement techniques

Method of measurement		Typical detection limit
γ-ray spectrometry in vivo (measurement of daughter product ^{208}Tl)	Whole body	10 Bq
	lungs	6 Bq
Radiochemical separation and α-spectrometry on biological sample	Urine	0.5 mBq d^{-1}
	Faeces	0.5 mBq

Thorium-232 (half life = 1.4 x 10^{10} y)

Table 8.5 Emissions

Radionuclide	Radiation	Energy (MeV)	Intensity (%)
^{232}Th	α	3.95	23
	α	4.01	77
^{228}Ac	β^-	0.39[a]	40
	β^-	0.61[a]	11
	β^-	0.75[a]	8
	γ	0.34	16
	γ	0.91	29
	γ	0.96	23
^{228}Th	α	5.3	27
	α	5.4	73
^{212}Pb	γ	0.24	45
^{212}Bi	γ	0.73	12
^{208}Tl	γ	0.51	22[b]
	γ	0.58	86[b]
	γ	0.86	12[b]
	γ	2.6	100[b]

[a]Average energy

[b]These intensities refer to one disintegration of ^{208}Tl, the decay of ^{232}Th to ^{208}Tl involves a branching fraction of 0.36. This should be taken into account when relating these intensities to one disintegration of ^{232}Th.

Table 8.6 Measurement techniques

Method of measurement		Typical detection limit
γ-ray spectrometry in vivo (measurement of daughter product Ac-228)	Lungs	10 Bq
Radiochemical separation, NAA and γ-spectrometry on biological sample	Urine	0.0008 mBqd^{-1}
	Faeces	0.0008 mBq per sample

*NAA = Neutron Activation Analysis

Table 8.7 Dose coefficients ($SvBq^{-1}$) for intake of thorium via inhalation / ingestion [2]

Nuclide	$t_{1/2}$	Type	Inhalation		Ingestion	
			f_1	e(50)	f_1	e(50)
^{228}Th	1.91y	M	5.0×10^{-4}	2.3×10^{-5}	5.0×10^{-4}	7.0×10^{-8}
		S	2.0×10^{-4}	3.2×10^{-5}	2.0×10^{-4}	3.5×10^{-8}
^{232}Th	1.40×10^{10} y	M	5.0×10^{-4}	2.9×10^{-5}	5.0×10^{-4}	2.2×10^{-7}
		S	2.0×10^{-4}	1.2×10^{-5}	2.0×10^{-4}	9.2×10^{-8}

Table 8.8 Dose coefficients ($SvBq^{-1}$) for intake of thorium via a contaminated wound for all wound model categories[14]

Radio-nuclide	Weak	Moderate	Strong	Avid	Colloid	Particle	Fragment
^{228}Th	1.18×10^{-4}	1.17×10^{-4}	1.03×10^{-4}	6.86×10^{-5}	5.68×10^{-5}	1.37×10^{-5}	8.64×10^{-7}
^{232}Th	4.52×10^{-4}	4.52×10^{-4}	4.48×10^{-4}	4.4×10^{-4}	4.47×10^{-4}	4.17×10^{-4}	1.92×10^{-5}

9. Uranium

The currently recommended biokinetic model for uranium was adopted in ICRP Publication 69[12]. This is shown in Figure 9.1 [2]. ICRP Publication 78 [2]recommended the values of transfer rates shown in Table 9.1, for substitution in the uraium biokinetic model (Figure 9.1). The model describes the kinetics of uranium in bone, which is the main site of deposition and retention, and also considers retention in liver, kidneys, and other soft tissues as well as routes of excretion. It takes account of initial uptake onto bone surfaces, transfer from surface to bone volume and recycling from bone and other tissues to plasma.

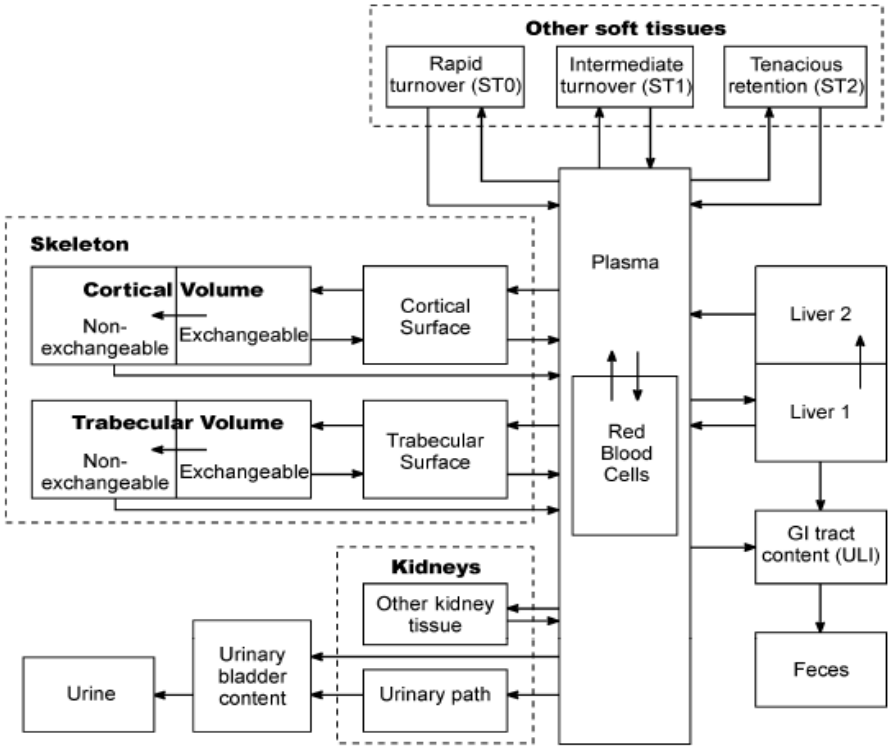


Figure 9.1 ICRP Publication 69 [12]biokinetic model for uranium.

Table 9.1 Transfer rates for uranium biokinetic model. [(a)]

Route of transfer between compartments	Transfer rate (d^{-1})
Plasma to ST0	10.5
Plasma to RBC	0.245
Plasma to Urinary bladder content	15.43
Plasma to Kidneys (urinary path)	2.94
Plasma to Kidneys (other tissue)	0.0122
Plasma to ULI contents	0.122
Plasma to Liver 1	0.367
Plasma to ST1	1.63
Plasma to ST2	0.0735
Plasma to Trabecular surfaces	2.04
Plasma to Cortical surfaces	1.63
ST0 to Plasma	8.32
RBC to Plasma	0.347
Kidneys (other tissue) to Plasma	0.00038
Liver 1 to Plasma	0.092
Liver 2 to Plasma	0.00019
ST1 to Plasma	0.0347
ST2 to Plasma	0.000019
Bone surfaces to Plasma	0.0693
Non-exchangeable trabecular volume to Plasma	0.000493
Non-exchangeable cortical volume to Plasma	0.0000821
Kidneys (urinary path) to Urinary bladder content	0.099
Liver 1 to Liver 2	0.00693
Bone surfaces to Exchangeable volume	0.0693
Exchangeable bone volume to Bone surfaces	0.0173
Exchangeable bone volume to Non-exchangeable volume	0.00578

[(a)] ICRP Publication 69 [12]notes that the values of rate constants are given to sufficient precision [3 decimal places in exponential notation] for calculational purposes and that this may be more precise than the biological data would support.

Table 9.2 Compounds, absorption types and f_1 values

Intake	f_1	Compounds
Ingestion	0.02	Unspecified compounds
Ingestion	0.002	Most tetravalent compounds e.g. UO_2, U_3O_8, UF_4
Inhalation, Type F	0.02	Soluble compounds including hexavalent compounds e.g. UF_6, UO_2F_2, and $UO_2(NO_3)_2$
Inhalation, Type M	0.02	Less soluble compounds, e.g. UO_3, UF_4, UCl_4, and most other hexavalent compounds

Inhalation, Type S	0.002	Highly insoluble compounds, e.g. UO_2 and U_3O_8

Table 9.3 Emissions

Radionuclide	Radiation	Energy (MeV)	Intensity (%)
^{232}U	α	5.32	68.15
		5.26	31.55
^{233}U	α	4.82	84.4
		4.78	13.2
^{234}U	α	4.72	27
	α	4.77	72
^{235}U	α	4.22	6
	α	4.33	5
	α	4.37	18
	α	4.40	56
	α	4.56	4
	α	4.60	5
	γ	0.14	11
	γ	0.19	54
	γ	0.21	5
^{238}U	α	4.15	23
	α	4.20	77

Table 9.4 Measurement techniques

Method of measurement	Radionuclide		Typical detection limit
γ-ray spectrometry in vivo	^{238}U	Lungs	40 Bq
	^{235}U	Lungs	3 Bq
	^{232}U	Lungs [γ-ray spectrometry of ^{212}Pb]	6 Bq
Radiochemical separation and α-spectrometry on biological sample		Urine	0.5 $mBqd^{-1}$
		Faeces	0.5 mBq per sample

Table 9.5 Dose coefficients($SvBq^{-1}$) for intake of uranium via inhalation / ingestion [2]

Nuclide	$t_{1/2}$	Type	Inhalation		Ingestion	
			f_1	e(50)	f_1	e(50)
^{232}U	68.9 y	F	0.02	4.7×10^{-6}	0.02	3.3×10^{-7}
		M	0.02	4.8×10^{-6}	0.002	3.7×10^{-8}
		S	0.002	2.6×10^{-5}		
^{233}U	1.59×10^5 y	F	0.02	6.6×10^{-7}	0.02	5.0×10^{-8}
		M	0.02	2.2×10^{-6}	0.002	8.5×10^{-9}

		S	0.002	6.9×10^{-6}		
^{234}U	2.44×10^{5} y	F	0.02	6.4×10^{-7}	0.02	4.9×10^{-8}
		M	0.02	2.1×10^{-6}	0.002	8.3×10^{-9}
		S	0.002	6.8×10^{-6}		
^{235}U	7.04×10^{8} y	F	0.02	6.0×10^{-7}	0.02	4.6×10^{-8}
		M	0.02	1.8×10^{-6}	0.002	8.3×10^{-9}
		S	0.002	6.1×10^{-6}		
^{238}U	4.47×10^{9} y	F	0.02	5.8×10^{-7}	0.02	4.4×10^{-8}
		M	0.02	1.6×10^{-6}	0.002	7.6×10^{-9}
		S	0.002	5.7×10^{-6}		

Table 9.6 Dose coefficients ($SvBq^{-1}$) for intake of uranium via a contaminated wound for all wound model categories[14]

Radio-nuclide	Weak	Moderate	Strong	Avid	Colloid	Particle	Fragment
^{232}U	1.25×10^{-5}	1.25×10^{-5}	1.27×10^{-5}	1.30×10^{-5}	1.33×10^{-5}	1.19×10^{-5}	5.18×10^{-7}
^{233}U	2.33×10^{-6}	2.33×10^{-6}	2.30×10^{-6}	2.24×10^{-6}	2.25×10^{-6}	1.97×10^{-6}	9.02×10^{-8}
^{234}U	2.27×10^{-6}	2.27×10^{-6}	2.25×10^{-6}	2.18×10^{-6}	2.19×10^{-6}	1.92×10^{-6}	8.75×10^{-8}
^{235}U	2.11×10^{-6}	2.11×10^{-6}	2.09×10^{-6}	2.03×10^{-6}	2.04×10^{-6}	1.78×10^{-6}	8.13×10^{-8}
^{238}U	2.03×10^{-6}	2.03×10^{-6}	2.01×10^{-6}	1.96×10^{-6}	1.97×10^{-6}	1.73×10^{-6}	7.89×10^{-8}

10. **Plutonium**

The currently recommended biokinetic model for the actinide elements (thorium, neptunium, plutonium, americium, and curium) was adopted in ICRP Publication 67[11]. This is shown in Figure 10.1 [2]. For plutonium absorbed to blood the main sites of deposition are the liver and skeleton. The model takes account of the initial deposition in bone, the liver, gonads and other tissues, and allows for transfer of activity from bone surfaces to bone volume and marrow, recycling of activity between tissues, as well as loss by excretion. *ICRP* recommended the values of transfer rates shown in Table 10.1 for substitution in the plutonium biokinetic model.

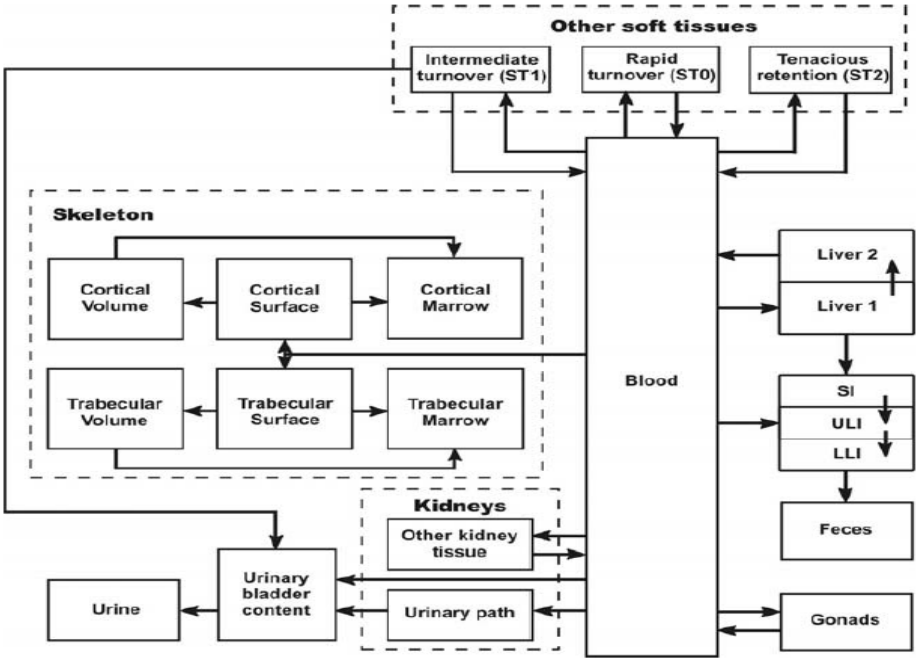


Figure 10.1 ICRP Publication 67 [11]biokinetic model for plutonium.

Table 10.1. Transfer rates for plutonium biokinetic model.[(a)]

Route of transfer between compartments	**Transfer rate (d^{-1})**
Blood to ST0	0.2773
Blood to ST1	0.0806
Blood to ST2	0.0129
Blood to Trabecular surface	0.1941
Blood to Cortical surface	0.1294
Blood to Urinary bladder contents	0.0129
Blood to Kidneys (urinary path)	0.00647
Blood to Kidneys (other tissue)	0.00323
Blood to ULI content	0.0129
Blood to Liver 1	0.1941
Blood to Testes	0.00023
Blood to Ovaries	0.000071
ST0 to Blood	0.693
ST1 To Blood	0.000475
ST1 to Urinary bladder content	0.000475
ST2 to Blood	0.000019
Liver 2 to Blood	0.000211
Liver 1 to Liver 2	0.00177
Liver 1 to Small Intestine (SI)	0.000133
Kidneys (other tissue) to blood	0.00139
Kidneys (urinary path) to Urinary bladder content	0.01386
Trabecular marrow to Blood	0.00760
Cortical marrow to blood	0.00760
Trabecular surface to Trabecular volume	0.000247
Cortical surface to Cortical volume	0.0000411
Trabecular surface to Trabecular marrow	0.000493
Trabecular volume to Trabecular marrow	0.000493
Cortical surface to Cortical marrow	0.0000821
Cortical volume to Cortical marrow	0.0000821
Testes to Blood	0.00019
Ovaries to Blood	0.00019

[(a)]*ICRP Publication 78*[2] notes that the values of rate constants are given to sufficient precision for calculational purposes and that this may be more precise than the biological data would support.

Table 10.2 Compounds, absorption types and f_1 values

Intake	f_1	Compounds
Ingestion	5.0×10^{-4}	Unspecified compounds
Ingestion	1.0×10^{-4}	Nitrates
Ingestion	1.0×10^{-5}	Insoluble oxides
Inhalation, Type M	5.0×10^{-4}	Unspecified compounds
Inhalation, Type S	1.0×10^{-5}	Insoluble oxides

Table 10.3 Emissions

Radionuclide	Radiation	Energy (MeV)	Intensity (%)
^{238}Pu	α	5.46	28
	α	5.50	72
	χ-rays	0.014-0.020	11
^{239}Pu	α	5.11	11
	α	5.14	15
	α	5.16	74
	χ-rays	0.014-0.022	4.6
^{240}Pu	α	5.12	27
	α	5.17	73
	χ-rays	0.014-0.020	10

Table 10.4 Measurement techniques

Method of measurement		Typical detection limit
γ-ray spectrometry in vivo	Lungs	2000 Bq
Radiochemical separation and α-spectrometry on biological sample	Urine	0.5 $mBqd^{-1}$
	Faeces	0.5 mBq per sample

Table 10.5 Dose coefficients ($SvBq^{-1}$) for intake of plutonium via inhalation / ingestion [2]

Nuclide	$t_{1/2}$	Type	Inhalation		Ingestion	
			f_1	e(50)	f_1	e(50)
^{238}Pu	87.7y	M	5.0×10^{-4}	3.0×10^{-5}	5.0×10^{-4}	2.3×10^{-7}
		S	1.0×10^{-5}	1.1×10^{-5}	1.0×10^{-4}	4.9×10^{-8}
					1.0×10^{-5}	8.8×10^{-9}
^{239}Pu	2.41×10^4 y	M	5.0×10^{-4}	3.2×10^{-5}	5.0×10^{-4}	2.5×10^{-7}
		S	1.0×10^{-5}	8.3×10^{-6}	1.0×10^{-4}	5.3×10^{-8}
					1.0×10^{-5}	9.0×10^{-9}
^{240}Pu	6.54×10^3 y	M	5.0×10^{-4}	3.2×10^{-5}	5.0×10^{-4}	2.5×10^{-7}
		S	1.0×10^{-5}	8.3×10^{-6}	1.0×10^{-4}	5.3×10^{-8}
					1.0×10^{-5}	9.0×10^{-9}

Table 10.6 Dose coefficients ($SvBq^{-1}$) for intake of plutonium via a contaminated wound for all wound model categories[14]

Radio-nuclide	Weak	Moderate	Strong	Avid	Colloid	Particle	Fragment
^{238}Pu	$4.44x10^{-4}$	$4.43x10^{-4}$	$4.36x10^{-4}$	$4.18x10^{-4}$	$4.17x10^{-4}$	$3.36x10^{-4}$	$1.41x10^{-5}$
^{239}Pu	$4.89x10^{-4}$	$4.89x10^{-4}$	$4.83x10^{-4}$	$4.66x10^{-4}$	$4.67x10^{-4}$	$3.9x10^{-4}$	$1.67x10^{-5}$
^{240}Pu	$4.89x10^{-4}$	$4.89x10^{-4}$	$4.83x10^{-4}$	$4.66x10^{-4}$	$4.67x10^{-4}$	$3.9x10^{-4}$	$1.67x10^{-5}$

11. Americium

The currently recommended biokinetic model for the actinide elements (thorium, neptunium, plutonium, americium, and curium) was adopted in ICRP Publication 67 [11]. This is shown in Figure 11.1 [2]. For americium absorbed to blood the main sites of deposition are the liver and skeleton. The model takes account of the initial deposition in bone, the liver, gonads and other tissues, and allows for transfer of activity from bone surfaces to bone volume and marrow, recycling of activity between tissues, as well as loss by excretion. *ICRP* recommended the values of transfer rates shown in Table 11.1 for substitution in the americium biokinetic model

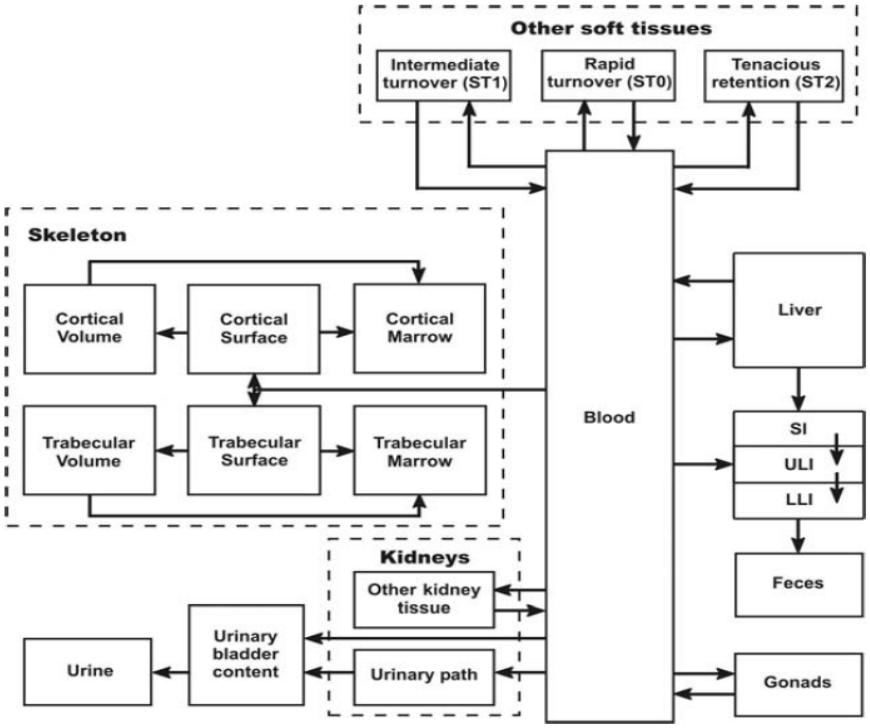


Figure 11.1 *ICRP Publication* 67 [11]biokinetic model for americium.

Table 11.1 Transfer rates for americium biokinetic model.[(a)]

Route of transfer between compartments	Transfer rate (d^{-1})
Blood to ST0	10.0
Blood to ST1	1.67
Blood to ST2	0.466
Blood to Trabecular surface	3.49
Blood to Cortical surface	3.49

Blood to Urinary bladder content	1.63
Blood to Kidneys (urinary path)	0.466
Blood to Kidneys (other tissue)	0.116
Blood to ULI content	0.303
Blood to Liver	11.6
Blood to Testes	0.0082
Blood to Ovaries	0.0026
ST0 to Blood	1.386
ST1 to Blood	0.0139
ST2 to Blood	0.000019
Liver to Blood	0.00185
Liver to Small intestine (SI)	0.000049
Kidneys (other tissue) to Blood	0.00139
Kidneys (urinary path) to Urinary bladder content	0.099
Trabecular marrow to Blood	0.0076
Cortical marrow to Blood	0.0076
Trabecular surface to Trabecular volume	0.000247
Cortical surface to Cortical volume	0.0000411
Trabecular surface to Trabecular marrow	0.000493
Trabecular volume to Trabecular marrow	0.000493
Cortical surface to Cortical marrow	0.0000821
Cortical volume to Cortical marrow	0.0000821
Testes to Blood	0.00019
Ovaries to Blood	0.00019

[(a)]*ICRP Publication* 78 [2] notes that the values of rate constants are given to sufficient precision for calculational purposes and that this may be more precise than the biological data would support.

Table 11.2 Compounds, absorption types and f_1 values

Intake	**f_1**	**Compounds**
Ingestion	5.0×10^{-4}	All compounds
Inhalation, Type M	5.0×10^{-4}	All compounds

Table 11.3 Emissions

Radionuclide	**Radiation**	**Energy (MeV)**	**Intensity (%)**
^{241}Am	α	5.39	1.4
	α	5.44	13
	α	5.49	85
	α	5.51	0.2
	α	5.54	0.3
	γ	0.06	36

Table 11.4 Measurement techniques

Method of measurement		**Typical detection limit**
γ-ray spectrometry in vivo	Lungs	6 Bq
	Urine	0.5 $mBqd^{-1}$

Radiochemical separation and α-spectrometry on biological sample	Faeces	0.5 mBq per sample

Table 11.5 Dose coefficients ($SvBq^{-1}$) for intake of americium via inhalation / ingestion [2]

Nuclide	**$t_{1/2}$**	**Type**	**Inhalation**		**Ingestion**	
			f_1	**e(50)**	**f_1**	**e(50)**
^{241}Am	4.32 x 10^2y	M	5.0 x 10^{-4}	2.7 x 10^{-5}	5.0 x 10^{-4}	2.0 x 10^{-7}

Table 11.6 Dose coefficients ($SvBq^{-1}$) for intake of americium via a contaminated wound for all wound model categories[14]

Radio-nuclide	**Weak**	**Moderate**	**Strong**	**Avid**	**Colloid**	**Particle**	**Fragment**
^{241}Am	3.97x10^{-4}	3.96x10^{-4}	3.91x10^{-4}	3.79x10^{-4}	3.8x10^{-4}	3.23x10^{-4}	1.41x10^{-5}

Annexure II

METHODS FOR MEASUREMENT OF RADON, THORON AND THEIR DAUGHTER PRODUCTS

There are several methods for measurement of radon and its progeny in the working environment. Some of the commonly employed techniques are described in this annexure.

A.1 Scintillation cell technique for radon measurement

One of the simplest and very convenient method for collection of large number of air samples from the mine workings for radon measurements is based on the scintillation cell technique [31]. In this technique a 150 ml metallic cylindrical cell coated inside with ZnS(Ag) with a transparent window is optically coupled to a photomultiplier tube and a counting system. Pre-filtered air is introduced into the evacuated cell which is later counted for alpha activity over a known period, typically for about 600 seconds after ~ 180 minutes post sampling, when the progeny attain equilibrium with radon in the cell. Radon concentration is estimated using the relation;

$$C_{Rn} = \frac{0.0697 \times C}{E \times V \times \exp(-\lambda t) \times [1 - \exp(-\lambda T)]}$$

Where, C_{Rn} = radon concentration, $Bq.m^{-3}$,
C = net count rate (cps),
E = efficiency of counting system (fraction),
V = volume of the sampler (m^3),
λ = decay constant of radon (s^{-1}),
t = time delay post-sampling (s),
T = counting period (s).

For a 10 min (600 s) counting time after 180 min (10800 s) post sampling, the minimum detectable concentration of radon is ~ 40 $Bq.m^{-3}$.

A.2 Low level radon detection system (LLRDS)

Another method for the measurement of radon is based on electro-deposition of freshly formed positively charged (~90 %) ^{218}Po atoms on a negatively charged plate for a predetermined collection and alpha counting period [35]. The system typically consists of a 5 litre cylindrical aluminium chamber with a ~ 5 cm dia Al disc on the top lid. Filtered air is introduced in the system. A negative voltage of – 800 V is applied to the metallic disc for a period of 90 min and the alpha activity from radon progeny deposited on the plate is counted, ideally from 1 to 75 min post collection. The radon concentration is evaluated from the relation,

$$C_{Rn} = \frac{C}{E \times V \times Z \times 0.9 \times (1 - e^{(0.042H - 4.31)})}$$

Where C = net counts observed
E = efficiency of the counting system (fraction)
V = volume of the chamber (m^3)
H = relative humidity in the sample (%)
Z = alpha emission factor

$$Z = \sum_{i=1}^{3} K_i \left(1 - e^{-\lambda_i t}\right)\left(e^{-\lambda_i T_1} - e^{-\lambda_i T_2}\right)$$

K_1 = 277 s, K_2 = 982 s, K_3 = - 5599 s
λ_i – decay constant of the i^{th} radon daughter (s^{-1}), (i=1, 2,3).
T_1 and T_2 are start and end time of the measurement and t is total collection period.

This system is also used for measurement of radon in the exhaled breath of workers to evaluate internally deposited radium inhaled in the form of ore dust [35].

There are many other methods for measurement of radon in the working environment like continuous radon monitoring system [36][37] [38].

A.3 Measurement of radon daughter concentrations

The short-lived radon progeny are more important from the point of view of hazard evaluation. Hence there are several techniques and instruments available for evaluation of radon progeny detection and estimation. One of the most commonly used methods is the Kusnetz's method.

Kusnetz's method

In the Kusnetz's method [39] radon daughters are collected on a filter paper by drawing air with a pump for about 10 min and counting the alpha activity on the filter at any time between 40 and 90 min. This most widely used method in mines gives radon daughter working level concentrations within an error of ± 13 %, irrespective of the equilibrium condition, which is acceptable for a mine atmosphere which has many other variables. The radon daughter concentration in units of working level (WL) is given by relation,

$$WL = \frac{100 \times C}{E \times V \times F(T,t)}$$

Where

WL - radon progeny concentration (Working Level)
C - Count rate at the delay time of 't'
E - Efficiency of the alpha counter (%)
V - Sampling rate (lpm)
F(T,t) - is the Kusnetz's factor which is a function of T and t (given in the table below)*

t - Counting delay after end of sampling (min.)
T - Sampling time (min.).

*Kuznet's factor for determination of Rn daughter concentration in working level (WL)

Time after sampling (min)	Kusznet's factor, F(T,t)
40	150
50	130
60	110
70	90
80	75
90	60

A.4 Measurement of Thoron daughter concentration

Similar to the measurement of radon daughter concentration, a simple methodology has been developed for evaluation of thoron daughter working level [40]. An air sample is collected by passing air through a high efficiency filter paper for 10 min. The alpha activity on the filter is measured for 10 minutes at any time between 180 minutes to 600 minutes post sampling. The thoron working level concentration is obtained using the relation,

$$T(WL) = \frac{C}{E \times V \times T \times Z(WL)}$$

Where C - α count rate (cpm) on the filter paper beyond 180 min,
E - efficiency of counting (fraction),
V - sampling flow rate (lpm),
T - duration of sampling (min),

Z(WL) – theoretical correction factor for thoron daughter at the counting instant (equivalent of Kusnetz's factor for thoron daughters).

The thoron concentration of 275 $Bq.m^{-3}$ in equilibrium with its daughter products gives a potential alpha energy equivalent to 1 WL (20.8 $\mu J.m^{-3}$). For a given mixture of air containing 275 $Bq.m^{-3}$ of Th-B (^{214}Pb), at extreme equilibrium ratios with Th-C (^{214}Bi), the alpha activity decays with the decay rate of Th-B. But at Th-B to Th-C ratio of 1:0, the alpha activity is initially zero and gradually grows to a maximum value at 180 min and thereafter decays with the decay rate of Th-B. The decay curve beyond 180 min may be approximately given by the relation,

$$Z(WL) = 246 \times e^{-\lambda_b \times t}$$

Where λ_b is the decay constant of Th-B and t is the counting time, for instant beyond 180 min.The maximum error due to different equilibrium conditions is within 10%.

A - 5 SSNTD based radon evaluation

For evaluation of very low levels of radon in the environment or in dwellings and for dosimetry of mine workers a system for integrated measurement of radon using SSNTD can be used.

The device comprises of a 60 ml cylindrical aluminium chamber covered with a permeable membrane which allows only radon to diffuse in, due to its relatively longer half life, and serves as a barrier for Rn-219, Rn-220 and aerosols. A 1.8 cm x 3 cm SSNTD film is placed between two TLD chips mounted on a disc at the base of the chamber. The SSNTD film and the TLD in the chamber are replaced every two months. The TLDs are processed to give cumulative gamma radiation exposure. The SSNTD film is etched with 10 % sodium hydroxide solution at a temperature of 60ºC for 90 min. The tracks are electronically counted which are correlated to radon exposure using a calibration system [41]. The cumulative radon exposure is obtained from the track density as,

$$\mathbf{E\ (Bq \times h \times l^{-1}) = 0.554\ T}$$

Where

E – Cumulativeexposure
T – Track density per cm^2
h – Exposure period (h)
0.554 is calibration factor between track density T and Rn concentration $Bq.l^{-1}$)

When used for personnel dosimetery, the exposure from radon progeny (ERp) is given as,

$$ERp = 4.4 \times 10^{-4} \times F \times T$$

Where

F – Equilibrium factor (0.4 for mines)
4.4×10^{-4} is the proportionality constant.

REFERENCES

[1] ATOMIC ENERGY REGULATORY BOARD, SAFETY CODE on Radiation Protection for Nuclear Fuel cycle Facilities (AERB/NF/SC/RP (2012).

[2] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Individual Monitoring for Internal Exposure of Workers, ICRP Publication 78, Ann. ICRP 27 (3-4) (1997).

[3] NATIONAL RADIOLOGICAL PROTECTION BOARD, Optimisation of Monitoring for Internal Exposure (OMINEX), NRPB-W60 (2004).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of Occupational Exposure due to Intakes of Radionuclides, Safety Standard Series No. RS-G-1.2, IAEA, Vienna(1999).

[5] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, General Principles for the Radiation Protection of the Workers, ICRP Publication 75. Ann. ICRP 27 (1) (1997).

[6] Manual on Internal Dose Computation and Reporting. Sawant, P. D; Sawant, J. V; Gurg, R. P; Rudran, K; Gupta, V. K; Abani, M. C; BARC/1999/E/010, pp.73. Bhabha Atomic Research Centre, Mumbai, India (1999).

[7] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4) (2008).

[8] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Limits for the Intakes of Radionuclides by Workers, ICRP Publication 30, Part 1. Ann. ICRP 2 (3/4) (1979).

[9] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Age–dependent Doses to Members of the Public from Intake of Radionuclides, ICRP Publication 56, Part 1. Ann. ICRP 20 (2) (1989).

[10] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Human Respiratory Tract Model for Radiological Protection, ICRP Publication 66, Ann. ICRP 24 (1-3) (1994).

[11] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Age–dependent Doses to Members of the Public from Intake of Radionuclides, ICRP Publication 67, Part 2 Ingestion Dose Coefficients. Ann. ICRP 23 (3/4) (1993).

[12] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Age–dependent Doses to Members of the Public from Intake of Radionuclides, ICRP Publication 69, Part 3 Ingestion Dose Coefficients. Ann. ICRP 25 (1) (1995).

[13] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Age–dependent Doses to Members of the Public from Intake of Radionuclides, ICRP Publication 71, Part 4, Inhalation Dose Coefficients,, Ann. ICRP 25(3-4) (1995).

[14] NATIONAL COUNCIL ON RADIATION PROTECTION & MEASUREMENTS, Development of a Biokinetic Model for Radionuclide-Contaminated Wounds and Procedures for Their Assessment, Dosimetry and Treatment, .NCRP Report No. 156, (2006).

[15] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Compendium of Dose Coefficients based on ICRP Publication 60, ICRP Publication 119, Ann. ICRP 41 (Suppl.) (2012).

[16] INTERNATIONAL ATOMIC ENERGY AGENCY, General Safety Requirements Part 3 (GSR Part 3), Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA, Vienna (2014).

[17] Dose Co-efficients for Intakes of Radionuclides via Contaminated Wounds, Toohey R.E; Bertelli, L; Sugarman, S. L; Wiley, A. L; Christensen, D. M; Radiation Emergency Assistance

Center/Training Site, REAC/TS-2012, Oak Ridge Institute for Science and Education, Oak Ridge (2012).

[18] United States of America Department of Energy Standard on Internal Dosimetry, DOE-STD 1121 (2008).

[19] INTERNATIONAL ATOMIC ENERGY AGENCY, Direct Methods for Measuring Radionuclides in the Human Body. Safety Series No. 114, IAEA, Vienna (1996).

[20] Ultrasonic Measurements of Chest Wall Thickness for the Assessment of Internal Contamination due to Actinides in Indian Radiation Workers, Nadar, M. Y; Singh, I. S; Chaubey, A; Kantharia, S; Bhati, S; Paper presented at NSRP-18 held at M.L. Sukhadia University Udaipur during 19-21 November, 2009.

[21] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Adult Reference Computational Phantoms, ICRP Publication 110, Ann. ICRP 39 (2), (2009).

[22] Monte Carlo Simulation of Skull and Knee Voxel Phantoms for the Assessment of Skeletal Burden of Low-energy Photon Emitters, Nadar, M. Y; Akar, D. K; Patni, H. K; Singh, I. S; Mishra, L; Rao, D. D; Pradeepkumar, K. S; Rad.. Prot. and Dosim. Vol 162(4), doi:10.1093/rpd/nct367 (2014).

[23] INTERNATIONAL ATOMIC ENERGY AGENCY, Indirect Methods for Assessing Intakes of Radionuclides causing Occupational Exposure, Safety Reports Series No. 18, IAEA, Vienna (2000).

[24] Technical Basis for using Nose Swab Bioassay Data for Early Internal Dose Assessment, Guilmette, R; Bertelli, L; Miller, G; Little T; Rad. Prot. Dosim. Prot., 127, 356, (2007).

[25] Applying Low Energy HPGe detector Gamma Ray Spectrometric technique for the evaluation of Pu/Am ratio in biological samples, Singh, I. S; Mishra, L; Yadav, J. R; Nadar, M. N; Rao, D. D; Pradeepkumar, K. S; Appl. Rad.Iso. 104, 49-54 (2015).

[26] Feasibility Studies to Assess the use of ^{236}Pu as a Radiochemical Yield Monitor in Bioassay Samples, Sawant, P. D; Kalsi, P. C; J. Radioanal. Nucl. Chem., 274 (1), 67 – 69,(2007).

[27] Estimation of Trace Levels of Plutonium in Urine Samples by Fission Track. Sawant, P. D; Prabhu, S. P; Kalsi, P. C; Pendharkar K. A; J. Radioanal. and Nucl. Chem., 279 (1), 179 – 183, (2009).

[28] Thorium in Human Blood Serum, Clot, and Urine - Comparison with ICRP Excretion Model. Sunta, C. M; Dang, H. S; Jaiswal, D. D; Soman S. D; J. Radioanal. & Nuc. Chem., Articles, 138 (1), 139-144, (1990).

[29] Determining the Normal Concentration of Uranium in Urine and Application of the Data to its Biokinetics. Dang, H. S; Pullat, V. R; Pillai, K. C; Health Phys. 62 (6):562–566, (1992).

[30] Current Status of Bioassay Procedures to Detect and Quantify Previous Exposures to Radioactive Materials, Boecker, B; Hall, R; Inn, K; Lawrence, J; Ziemer, P; Eisele, G; Wachholz, B; Burr, W., Jr.; Health Phys. 60, (1), 45 – 100 (1991).

[31] An Inexpensive Radon Scintillation Cell. Raghavayya, M; Health Phys. 40, 894-896, (1981).

[32] Database of Calculated Values of Retention and Excretion for Members of the Public following Acute Intake of Radionuclides. Ishigure, N; Nakano, T; Matsumoto, M; Enomoto, H; Rad. Prot. Dosim., 105 (1-4) 311- 316 (2003).

[33] IMBA Professional Plus: A Flexible Approach to Internal Dosimetry. Birchall, A; Puncher, M; Marsh, J. W; Davis, K; Bailey, M.R; Jarvis, N. S; Peach, A. D; Dorrian, M. D; James, A.C; Radiat. Prot. Dosim.125 (1-4): 194-197 (2007).

[34] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Dose Coefficients for Intakes of Radionuclides by Workers, ICRP Publication 68, Ann. ICRP 24 (4) (1994).

[35] A Low Level Radon Detection System. Srivastava, G. K; Raghavayya, M; Khan, A. H; Kotrappa, P; Health. Physics J., 46 (1), 225–228, (1984).

[36] A Study of the Technologically Modified Sources of ^{222}Rn and its Environmental Impact in an Indian U Mineralized Belt. Jha, S; Khan, A. H; Mishra, U. C; J. Environ. Radioact. 53, 183–197 (2001).

[37] Radon Measurement Studies with Indigenously Developed Continuous Radon Monitor (CRM), Ashokkumar, P; Raman, A; Srivastava, V. S; Patnaik, R. L; Babu, D. A. R; Sharma, D. N; Sixteenth National Symposium on Environment, Guru Jambeshwar University of Science and Technology, Hissar, Haryana, July 16 -18, 2008.

[38] Development of a Continuous Radon Monitor and Instant Working Level Meter. Khan, A. H; Raghavayya, M; Sharma, D. N; Raman, N; Jethmalani, P; Krishnamachari G; Iyer, M. R; Bull. of Rad. Prot., 12 (3), 56 - 61 (1989).

[39] Radon Daughters in Mine Atmosphere. Kusnetz, H. L; Amer. Ind. Hyg.Assoc. Qurtly. 17, 1, p. 85 (1956).

[40] Measurement of the Potential Alpha Energy Exposure of Thoron Progeny in Air. Khan, A . H; Gohel, C. D; Raghavayya, M; Nambiar P.P.V.J; Proceedings of the 15th Annual Conference of the IARP, CCMB, Hyderabad, March 7-9, 1988.

[41] Dose Estimate for Jaduguda Mine Workers. Raghavayya, M; Khan, A. H; Jha, G; Bulletin of Radiation Protection, 8, 69-73, (1986).

BIBLIOGRAPHY

1. Modification of the Tsivoglou method for radon daughters in air, Thomas, J. W.,Health. Phys. Vol. 19, p-691 (1970).
2. Radiation protection- Dose assessment for monitoring of workers for internal radiation exposure, ISO/DIS 27048; International Organization for Standardization, Geneva (2011).
3. Radiation protection - Performance criteria for Radiobioassay, ISO/DIS28218;International Organization for Standardization, Geneva (2010).
4. Internal Dosimetry Program Guide for Use with Title 10, Code of Federal Regulations, Part 835, DOE G 441.1-3A, Occupational Radiation Protection, U.S. Department of Energy Washington D.C. 20585 (2005).
5. DOE Standard-1121: Internal Dosimetry, U.S. Department of Energy Washington D.C. 20585 (2008).
6. Pacific Northwest National Laboratory (PNNL). Methods and Models of the Hanford Internal Dosimetry Program, PNNL-MA-860. Report PNNL-15614, Rev. 1, Richland, Washington (2009).
7. INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION,Occupational Intakes of radionuclides (Part I), ICRP Publication 130 Ann. ICRP 44(2), (2015).

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4.	AERB/SG/G-8; 2001	Criteria for Regulation of Health and Safety of Nuclear Power Plant Personnel, the Public and the Environment
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