IAEA SAFETY STANDARDS SERIES

Assessment of Occupational Exposure Due to Intakes of Radionuclides

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INTERNATIONAL ATOMIC ENERGY AGENCY AND THE
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SAFETY GUIDE

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ASSESSMENT OF OCCUPATIONAL EXPOSURE DUE TO INTAKES OF RADIONUCLIDES

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

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FOREWORD

by Mohamed ElBaradei
Director General

One of the statutory functions of the IAEA is to establish or adopt standards of safety for the protection of health, life and property in the development and application of nuclear energy for peaceful purposes, and to provide for the application of these standards to its own operations as well as to assisted operations and, at the request of the parties, to operations under any bilateral or multilateral arrangement, or, at the request of a State, to any of that State's activities in the field of nuclear energy.

The following advisory bodies oversee the development of safety standards: the Advisory Commission on Safety Standards (ACSS); the Nuclear Safety Standards Advisory Committee (NUSSAC); the Radiation Safety Standards Advisory Committee (RASSAC); the Transport Safety Standards Advisory Committee (TRANSSAC); and the Waste Safety Standards Advisory Committee (WASSAC). Member States are widely represented on these committees.

In order to ensure the broadest international consensus, safety standards are also submitted to all Member States for comment before approval by the IAEA Board of Governors (for Safety Fundamentals and Safety Requirements) or, on behalf of the Director General, by the Publications Committee (for Safety Guides).

The IAEA's safety standards are not legally binding on Member States but may be adopted by them, at their own discretion, for use in national regulations in respect of their own activities. The standards are binding on the IAEA in relation to its own operations and on States in relation to operations assisted by the IAEA. Any State wishing to enter into an agreement with the IAEA for its assistance in connection with the siting, design, construction, commissioning, operation or decommissioning of a nuclear facility or any other activities will be required to follow those parts of the safety standards that pertain to the activities to be covered by the agreement. However, it should be recalled that the final decisions and legal responsibilities in any licensing procedures rest with the States.

Although the safety standards establish an essential basis for safety, the incorporation of more detailed requirements, in accordance with national practice, may also be necessary. Moreover, there will generally be special aspects that need to be assessed by experts on a case by case basis.

The physical protection of fissile and radioactive materials and of nuclear power plants as a whole is mentioned where appropriate but is not treated in detail; obligations of States in this respect should be addressed on the basis of the relevant instruments and publications developed under the auspices of the IAEA.

Non-radiological aspects of industrial safety and environmental protection are also not explicitly considered; it is recognized that States should fulfil their international undertakings and obligations in relation to these.

The requirements and recommendations set forth in the IAEA safety standards might not be fully satisfied by some facilities built to earlier standards. Decisions on the way in which the safety standards are applied to such facilities will be taken by individual States.

The attention of States is drawn to the fact that the safety standards of the IAEA, while not legally binding, are developed with the aim of ensuring that the peaceful uses of nuclear energy and of radioactive materials are undertaken in a manner that enables States to meet their obligations under generally accepted principles of international law and rules such as those relating to environmental protection. According to one such general principle, the territory of a State must not be used in such a way as to cause damage in another State. States thus have an obligation of diligence and standard of care.

Civil nuclear activities conducted within the jurisdiction of States are, as any other activities, subject to obligations to which States may subscribe under international conventions, in addition to generally accepted principles of international law. States are expected to adopt within their national legal systems such legislation (including regulations) and other standards and measures as may be necessary to fulfil all of their international obligations effectively.

PREFACE

Occupational exposure to ionizing radiation can occur in a range of industries, medical institutions, educational and research establishments and nuclear fuel cycle facilities. Adequate radiation protection of workers is essential for the safe and acceptable use of radiation, radioactive materials and nuclear energy.

In 1996, the Agency published Safety Fundamentals on Radiation Protection and the Safety of Radiation Sources (IAEA Safety Series No. 120) and International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (IAEA Safety Series No. 115), both of which were jointly sponsored by the Food and Agriculture Organization of the United Nations, the IAEA, the International Labour Organisation, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization. These publications set out, respectively, the objectives and principles for radiation safety and the requirements to be met to apply the principles and to achieve the objectives.

The establishment of safety requirements and guidance on occupational radiation protection is a major component of the support for radiation safety provided by the Agency to its Member States. The objective of the Agency's Occupational Protection Programme is to promote an internationally harmonized approach to the optimization of occupational radiation protection, through the development and application of guidelines for restricting radiation exposures and applying current radiation protection techniques in the workplace.

Guidance on meeting the requirements of the Basic Safety Standards for occupational protection is provided in three interrelated Safety Guides, one giving general guidance on the development of occupational radiation protection programmes and two giving more detailed guidance on the monitoring and assessment of workers' exposure due to external radiation sources and from intakes of radionuclides, respectively. These Safety Guides together reflect the current internationally accepted principles and recommended practices in occupational radiation protection, with account taken of the major changes that have occurred over the past decade.

The three Safety Guides on occupational radiation protection are jointly sponsored by the IAEA and the International Labour Office.

The present Safety Guide addresses the assessment of exposure due to intakes of radionuclides in the workplace. Such intakes can occur via a number of pathways whenever unsealed sources are present, and the monitoring of workers and the workplace in such situations is an integral part of any occupational radiation protection programme. The assessment of exposure due to intakes depends critically upon knowledge of the biokinetics of the radionuclides, and the present Safety Guide reflects the major changes over the past decade in international practice in internal dose assessment.

EDITORIAL NOTE

An appendix, when included, is considered to form an integral part of the standard and to have the same status as the main text. Annexes, footnotes and bibliographies, if included, are used to provide additional information or practical examples that might be helpful to the user.

The safety standards use the form 'shall' in making statements about requirements, responsibilities and obligations. Use of the form 'should' denotes recommendations of a desired option.

The English version of the text is the authoritative version.

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. Occupational exposure due to radioactive materials can occur as a result of various human activities. These include work associated with the different stages of the nuclear fuel cycle, the use of radioactive sources in medicine, scientific research, agriculture and industry, and occupations which involve the handling of materials containing enhanced concentrations of naturally occurring radionuclides. In order to control this exposure, it is necessary to be able to assess the magnitude of the doses involved.

1.2. The IAEA Safety Fundamentals publication Radiation Protection and the Safety of Radiation Sources [1] presents the objectives, concepts and principles of radiation protection and safety. Requirements designed to meet the objectives and apply the principles specified in the Safety Fundamentals, including requirements for the protection of workers exposed to sources of radiation, are established in the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (commonly referred to as the Basic Safety Standards or BSS), jointly sponsored by the Agency and five other international organizations [2].

1.3. Three interrelated Safety Guides, prepared jointly by the IAEA and the International Labour Office (ILO), provide guidance on the application of the requirements of the Basic Safety Standards with respect to occupational exposure. Reference [3] gives general advice on the exposure conditions for which monitoring programmes should be set up to assess radiation doses arising from external radiation and from intakes of radionuclides by workers. More specific guidance on the assessment of doses from external sources of radiation can be found in Ref. [4] and the present Safety Guide deals with intakes of radioactive materials.

1.4. Recommendations related to occupational radiation protection have also been developed by the International Commission on Radiological Protection (ICRP) [5]. These and other current recommendations of the ICRP [6] have been taken into account in preparing this Safety Guide.

OBJECTIVE

1.5. The purpose of this Safety Guide is to provide guidance for regulatory authorities on conducting assessments of intakes of radioactive material arising from occupational exposure. This Guide will also be useful to those concerned with the

planning, management and operation of occupational monitoring programmes, and to those involved in the design of equipment for use in internal dosimetry and workplace monitoring.

SCOPE

1.6. This Safety Guide presents the main considerations for monitoring for internal exposures in both routine and accident situations, using direct and indirect methods. It also introduces monitoring of levels of radionuclides in the working environment as a basis for assessing intakes. The biokinetic and dosimetric models needed for more specific estimates of doses to individuals, to be used in the case of accidents or incidents, or when operations could result in doses approaching regulatory limits, are also presented.

1.7. This Safety Guide does not cover the medical exposure of patients or exposure of members of the public, nor does it give specific advice on monitoring of workers in mining and milling.

1.8. Technical details and advice on the assessment of internal contamination by direct methods has been published by the IAEA [7]. Practical advice on the use of indirect methods as well as interpretation of measurements in terms of the amount of radioactive material taken into the body and the associated radiation doses will be given in future IAEA publications.

STRUCTURE

1.9. The primary dosimetric and derived operational quantities used in radiation protection that relate to the assessment of doses from intakes of radionuclides are summarized in Section 2. In Section 3, the principles involved in the development of monitoring programmes and the need for individual or area monitoring are discussed. The selection of individuals and the choice of either direct or indirect methods for assessing the extent of any internal contamination in routine and accident situations are also reviewed in Section 3. The methods that have been developed for directly assessing the body or organ/tissue content of radionuclides by external counting of photon emissions emanating from the body are reviewed in Section 4. The use of indirect methods for assessing either the body content of a radionuclide or for investigating whether an intake has occurred from biological or physical samples is considered in Section 5. Models for describing the behaviour of radionuclides in the body are summarized in Section 6. Their application to calculate levels of radionuclides in the

body and hence radiation doses from measurements made by either direct or indirect methods is illustrated in Section 7. Requirements for record keeping, for measurements both on individuals and from area monitoring, are considered in Section 8. Finally, guidance on quality assurance procedures is given in Section 9.

1.10. Two appendices and an annex provide additional information. Appendix I provides suggested criteria to indicate whether individual monitoring is necessary. Appendix II defines procedures for calculating detection limits for measurement methods. The Annex provides, for ease of reference, some basic data relevant to the assessment of occupational exposure due to intakes of radionuclides, namely tissue weighting factors and dose coefficients (committed doses per unit intake) and derived air concentrations (DACs) for selected chemical forms of some common radionuclides.

2. DOSIMETRIC QUANTITIES

2.1. The quantities adopted in the BSS to express the doses received from intakes of radionuclides for radiological protection purposes are the effective dose E and the equivalent dose H_T in tissue or organ T. These quantities are briefly discussed in the related Safety Guide [3] and are formally defined in the BSS [2]. The quantity of primary interest for internal dose assessment is the intake, which is defined here as the activity of a radionuclide taken into the body[1]. The dose coefficient (committed effective dose per unit intake) for radionuclide j — by ingestion, $e(g)_{j,ing}$, or by inhalation, $e(g)_{j,inh}$, as appropriate — is used to determine the committed effective dose from an estimated intake. For occupational exposure, all exposed persons are adults and therefore the period of time over which the committed effective dose is assessed is 50 years, irrespective of the age at intake.

2.2. Internal doses cannot be measured directly; they can only be inferred from measured quantities such as body activity content, excretion rates or airborne concentrations of radioactive material. Section 7 provides an illustration of the assessment of doses from such measurements.

[1] In the BSS, intake is defined as “the process of taking radionuclides into the body by inhalation or ingestion or through the skin”. In this Safety Guide, the term intake is used both in this descriptive sense and in the more specific sense described in the text.

2.3. In situations of exposure due to a single radionuclide by inhalation or ingestion, with no external exposure, the limit on intake $I_{j,L}$ corresponding to the relevant limit L on effective dose is given by:

$$I_{j,L} = \frac{L}{e(g)_j}$$

where $e(g)_j$ is the relevant value of committed effective dose per unit intake. When there is exposure due to a range of radionuclides and/or external exposure, the total effective dose will need to be calculated. Requirements for and guidance on dose assessment in these circumstances are given in the BSS [2] and in Ref. [3].

2.4. Values of the committed effective dose per unit intake by ingestion and by inhalation for occupational exposure are given in Table II–III of the BSS [2] (except for radon progeny and thoron progeny). Values for selected radionuclides are reproduced in Table A–I of the Annex to the present Safety Guide.

2.5. The limits on intake and exposure for radon progeny and thoron progeny are given in Table II–I of the BSS [2] and are summarized in the companion Safety Guide [3].

2.6. The fraction of an intake that remains in the body (for direct methods) or that is being excreted from the body (for indirect methods) at time t after an intake may be designated $m(t)$ [8, 9]. This fraction depends on the radionuclide, its chemical and physical form and the route of intake, as well as t. To estimate the intake for dose assessment, the measured body content or excretion rate must be divided by the appropriate value of $m(t)$ (see Section 7). The committed dose can be seriously underestimated if the dose coefficient $e(g)_j$ is applied directly to the measured body content rather than to the inferred intake.

2.7. The potential for inhalation of radionuclides should be assessed when necessary by measuring activity levels in air samples. The derived air concentration (DAC, expressed in Bq/m^3) is defined as that concentration of airborne activity which would result in the intake of $I_{j,inh,L}$ by a worker exposed continuously for one year (taken to be 2000 working hours). For a standard breathing rate of 1.2 m^3/h, the DAC would thus be given by:

$$\mathrm{DAC} = \frac{I_{j,inh,L}}{2000 \times 1.2}$$

2.8. For example, for inhalation by a worker of ^{137}Cs as an aerosol with an AMAD of 5 µm, $e(g)_{j,inh}$ is 6.7×10^{-9} Sv/Bq. If L is assumed to be the occupational dose limit of 20 mSv/a (0.02 Sv/a) [3], then:

$$I_{j,inh,L} = \frac{0.02}{6.7 \times 10^{-9}} = 3 \times 10^{6}\,\text{Bq}$$

and

$$\text{DAC} = \frac{3 \times 10^{6}}{2000 \times 1.2} = 1.3 \times 10^{3}\,\text{Bq/m}^{3}$$

In practice, the DAC would be rounded to 1×10^3 Bq/m^3. Table A–2 of the Annex gives example values of DACs.

2.9. The measured airborne activity concentration, expressed as a fraction of the DAC, may be multiplied by the exposure time in hours to obtain an estimate of intake expressed in units of DAC·h. By definition, 2000 DAC·h corresponds to an intake of $I_{j,inh,L}$.

3. MONITORING PROGRAMME

GENERAL OBJECTIVE

3.1. The general objective of operational monitoring programmes is the assessment of workplace conditions and individual exposures. The assessment of doses to workers routinely or potentially exposed to radiation through intakes of radioactive material constitutes an integral part of any radiation protection programme and helps to ensure acceptably safe and satisfactory radiological conditions in the workplace.

3.2. Measures to meet the general requirements for the radiation protection of workers are described in the related Safety Guide on the application of occupational radiation protection principles [3]. The specific aspects of monitoring that relate to exposure due to intakes of radioactive material are described below.

3.3. Typical methods of individual monitoring for intakes are whole body counting, organ counting (such as thyroid or lung monitoring) and analysis of samples of excreta. Sampling of the breathing zone with personal air samplers is also used.

3.4. In many circumstances involving exposure due to radionuclides, workplace monitoring will be needed. Monitoring procedures may be introduced to demonstrate satisfactory working conditions or in cases where individual monitoring is unable to provide adequate protection of the worker. Such workplace monitoring may also be appropriate when levels of contamination are low, for example in a research laboratory using small quantities of radioactive tracers.

3.5. Monitoring for the estimation of doses from intakes of radionuclides may include one or more of the following techniques:

(a) Sequential measurements of radionuclides in the whole body or in specific organs;
(b) Measurements of radionuclides in biological samples such as excreta or breath;
(c) Measurement of radionuclides in physical samples such as filters from personal or fixed air samplers, or surface smears.

Measurements can be used to calculate the intake of a radionuclide, which, when multiplied by the appropriate dose coefficient, leads to an estimate of committed effective dose. Dose coefficients for a wide range of radionuclides are given in the BSS [2], and those for selected radionuclides are reproduced in Table A–1 of the Annex. In some circumstances, the results of direct measurement may be used to calculate dose rates to the whole body or to individual organs.

Need for monitoring

3.6. The designation of areas as controlled areas or supervised areas and the need for individual monitoring will be determined from a knowledge of conditions in the workplace and the potential for worker exposure. In general, the decision to enrol a worker in an internal exposure monitoring programme should be based on the likelihood that the individual could receive an intake of radioactive material exceeding a predetermined level. Guidance on the designation of controlled and supervised areas is given in the related Safety Guide [3]. If operational procedures are set up to prevent or reduce the possibility of intake, a controlled area will, in general, need to be established [3].

3.7. The need or otherwise for individual or area monitoring for internal exposure will depend on the amount of radioactive material present and the radionuclide(s) involved, the physical and chemical form of the radioactive material, the type of containment used, the operations performed and the general working conditions. For example, workers handling sealed sources, or unsealed sources in reliable containment, may need to be monitored for external exposure, but not necessarily for internal exposure. Conversely, workers handling radionuclides such as tritium, ^{125}I or ^{239}Pu may need monitoring for internal exposure, but not for external exposure.

3.8. It can be difficult to determine whether monitoring a worker for intakes of radioactive material is necessary. Such monitoring should be used routinely only for workers who are employed in areas that are designated as controlled areas specifically in relation to the control of contamination and in which there are grounds for expecting significant intakes. If experience has shown that it is unlikely that committed effective doses from annual intakes of radionuclides from occupational exposure would exceed 1 mSv, then individual monitoring may be unnecessary, but workplace monitoring should be undertaken.

3.9. Examples of situations in which experience has shown that it is necessary to give consideration to routine individual monitoring for internal exposure include the following:

(a) Handling of large quantities of gaseous or volatile materials, for example tritium and its compounds in large scale production processes, in heavy water reactors and in luminizing;
(b) Processing of plutonium and other transuranic elements;
(c) Mining, milling and processing of thorium ores, and the use of thorium and its compounds (which can lead to internal exposure due both to radioactive dusts and to thoron (^{220}Rn) and its progeny);
(d) Mining, milling and refining of high grade uranium ores;
(e) Processing of natural and slightly enriched uranium, and reactor fuel fabrication;
(f) Bulk production of radioisotopes;
(g) Working in mines and other workplaces where radon levels exceed a specified action level;
(h) Handling of large quantities of radiopharmaceuticals, such as ^{131}I for therapy;
(i) Maintenance of reactors, which can lead to exposure due to fission and activation products.

3.10. For some radionuclides, individual monitoring may not be feasible because of the radiation type(s) emitted and the detection sensitivity of monitoring methods, and reliance must be placed on workplace monitoring. Conversely, for some other

radionuclides such as tritium, individual monitoring may be more sensitive than workplace monitoring.

3.11. For new operations, individual monitoring is likely to be needed and should be considered. As experience in the workplace is accumulated, the need for routine individual monitoring should be kept under review. Workplace monitoring may be found to be sufficient for radiological protection purposes.

3.12. Some guidance on and examples of criteria to determine whether individual monitoring is necessary are given in Appendix I.

Design of a routine monitoring programme

3.13. Routine internal exposure monitoring is that conducted on a fixed schedule for selected workers. Internal exposure monitoring has several limitations that should be considered in the design of an adequate monitoring programme.

3.14. Firstly, monitoring does not measure directly the committed effective dose to the individual. Biokinetic models are needed to relate the activity level in an excreta sample to that in the body at the time the sample was taken, to relate the body content at the time the sample was taken to the original intake, and to calculate the committed effective dose from the estimated intake.

3.15. Secondly, measurements may be subject to interference from other radionuclides present in the body, such as ^{40}K present naturally, ^{137}Cs from global fallout, uranium naturally present in the diet or radiopharmaceuticals administered for diagnostic or therapeutic purposes. It is therefore important to establish the body content of both naturally occurring and artificial radionuclides from previous intakes. This is particularly important when the non-occupational intakes are elevated, for example in mining areas having above average domestic exposure due to radon. All workers should undergo bioassay measurements before commencing work with radioactive materials, in order to determine a 'background' level.

3.16. Radiopharmaceuticals may interfere with bioassay measurements for some time after administration, depending on the properties of the agent administered and on the radionuclides present at the workplace. Workers should be requested to report any administration of radiopharmaceuticals to their supervisors, so that it can be determined whether or not adequate internal exposure monitoring can be performed.

3.17. Thirdly, the results of an individual monitoring programme for the estimation of chronic intakes might depend on the time at which the monitoring is performed.

For certain radionuclides with a significant early clearance component of excretion, there may be a significant difference between measurements taken before and after the weekend. Such cases should be reviewed individually if chronic exposure is possible [8–10]. Additionally, for nuclides with long effective half-lives, the amount present in the body and the amount excreted depend on, and will increase with, the number of years for which the worker has been exposed. In general, the retained activity from previous years' intakes should be taken to be part of the background for the current year.

3.18. Finally, the analytical methods used for individual monitoring sometimes do not have adequate sensitivity to detect the activity levels of interest (see Appendix II). If individual monitoring is not feasible, then a system of workplace and personnel monitoring should be employed to determine, as far as possible, the amounts of radionuclides that may have been taken in by an individual. Fixed (static) air samplers or personal air samplers (PASs) may be used to determine the concentration of airborne radioactive material, which can be combined with standard or site specific assumptions about the physicochemical form of the material and the breathing rate and exposure time of the worker to estimate inhalation intakes. Similarly, surface monitoring may be used to indicate the potential for intake or the need for more detailed area monitoring, but the models for estimating intakes from surface contamination are particularly uncertain.

3.19. Exposure due to radon is of particular concern in underground mines, in buildings constructed with material containing significant levels of radium, in offices, factories and other premises with elevated levels of uranium in the ground, and in buildings where large amounts of groundwater are processed. In 1993, the ICRP issued recommendations on protection against ^{222}Rn at home and at work [11]. Safety Series No. 26 [12] covers radiation protection in the mining and milling of radioactive ores.

Methods of measurement

3.20. Intakes of radionuclides can be determined by either direct or indirect measurement methods. Direct measurements of gamma or X ray photons (including bremsstrahlung) emitted from internally deposited radionuclides are frequently referred to as body activity measurements, whole body monitoring or whole body counting. Indirect measurements are measurements of activity in samples which may be either biological (e.g. excreta) or physical (e.g. air filters). Each type of measurement has advantages and disadvantages, and the selection of one rather than another is largely dependent on the nature of the radiation to be measured. Direct methods are useful only for those radionuclides which emit photons of sufficient energy, and in sufficient numbers, to escape from the body and be measured by an external detector.

Many fission and activation products fall into this category. Incorporated radionuclides which do not emit energetic photons (e.g. ^{3}H, ^{14}C, $^{90}Sr/^{90}Y$, ^{239}Pu) can usually be measured only by indirect methods. However, some beta emitters, especially those with high energy emissions such as ^{32}P or $^{90}Sr/^{90}Y$, can sometimes be measured 'directly' via the bremsstrahlung produced. Such bremsstrahlung measurements, because of their relatively high minimum detectable activities (see Appendix II), are not usually employed for routine monitoring.

3.21. Direct measurements, where they are possible, offer the advantage of a rapid and convenient estimate of the total activity in the body or a defined part of the body at the time of measurement; when it is sufficiently sensitive, for example for ^{131}I and ^{137}Cs, direct measurement of body or organ content is therefore to be preferred. Whole body and individual organ measurements are less dependent on biokinetic models than indirect monitoring measurements, but they suffer from greater calibration uncertainties, especially for low energy photon emitters. Direct measurements may necessitate the worker being removed from work involving radiation exposure for the period over which the retention characteristics are measured, and usually need special, well shielded (and therefore expensive) facilities and equipment.

3.22. Direct measurements are useful in qualitative as well as quantitative determinations of radionuclides in a mixture that may have been inhaled, ingested or injected. In addition, direct measurements can assist in identifying the mode of intake by determining the distribution of activity in the body [13, 14]. Sequential measurements, where they are possible, can reveal the redistribution of activity and give information about the total body retention and the biokinetic behaviour of radionuclides in the body.

3.23. Indirect measurements generally interfere less with worker assignments, but require access to a radiochemical analytical laboratory; such a laboratory may also be used for measuring environmental samples, but high level (e.g. reactor water chemistry) and low level (e.g. bioassay or environmental samples) measurements should be performed in separate laboratories. Excreta measurements determine the rate of loss of radioactive materials from the body by a particular route, and must be related to the body content and intake by a biokinetic model. Because of the ability of radiochemical analyses to detect low levels of activity, measurements of excreta usually give sensitive detection of activity in the body.

3.24. Measurements of air samples can be difficult to interpret, because they measure the concentration of radionuclides in the air at the location of the sampler, not necessarily in the breathing zone of the worker. However, a personal air sampler (PAS) placed on the worker's lapel or protective headgear can collect a sample that is

representative of the activity concentration in air which the worker has inhaled, except in cases where the sample comprises only a few particles. Air concentration measurements, combined with assumptions about breathing rates and volumes and measured exposure times, can be used to estimate the intake. However, the use of PASs allows only estimates of intake and cannot be used to refine a dose estimate based on individual retention characteristics. Furthermore, PAS measurements cannot be repeated if an analytical result is suspect or is lost. They can, however, provide estimates of intakes for radionuclides such as ^{14}C (in particulate form), ^{239}Pu, ^{232}Th and ^{235}U, for which direct and other indirect methods of assessment of body activity are not sufficiently sensitive. This method of monitoring depends for its interpretation on the dose coefficients and the derived air concentrations (DACs), which are defined in Section 2 and discussed further in Section 7. Dose coefficients and DACs for various chemical forms of selected radionuclides are provided in Tables A–I and A–II.

3.25. Particle size influences the deposition of inhaled particulates in the respiratory tract and so information about the distribution of particle sizes is needed for the correct interpretation of bioassay results and subsequent dose assessment. In many situations the airborne particle size distribution should be determined using cascade impactors or other methods. As a minimum, air sample measurements should include measurement of the concentration of the respirable fraction of airborne particulates. Some models for interpreting PAS results discriminate against non-respirable particles [15]. In general, the more site and material specific information that is available, the better will be the dose assessment.

3.26. Measurement methods have limits of detection arising from the presence of naturally occurring radioactive materials, from statistical fluctuations in counting rates, and from factors related to sample preparation and analysis. Appendix II describes the concepts of minimum significant activity (MSA) and minimum detectable activity (MDA), which are used to characterize the limits of detection of any measurement method.

Frequency of monitoring

3.27. As stated in para. I.35 (Appendix I) of the BSS [2]: “The nature, frequency and precision of individual monitoring shall be determined with consideration of the magnitude and possible fluctuations of exposure levels and the likelihood and magnitude of potential exposures.” In order to determine the appropriate frequency and type of individual monitoring, the workplace should be characterized. The radionuclides in use and, if possible, their chemical and physical forms should also be known. If these forms are likely to change under accident conditions (e.g. the release of uranium hexafluoride into the atmosphere results in the production of HF and uranyl fluoride),

this should also be considered. The chemical and physical forms (e.g. particle size) of the material determine its behaviour on intake and its subsequent biokinetics in the human body. These in turn determine the excretion routes and rates, and hence the type of excreta samples to be collected and their frequency.

3.28. A consideration in setting a bioassay sampling schedule is to minimize the uncertainty in estimates of intake due to the unknown time of an intake within the monitoring period. The ICRP [8, 9] recommend that monitoring periods should generally be selected so that assuming an intake to have occurred at the mid-point of the monitoring period would not lead to underestimation of the intake by a factor of more than three.

3.29. Another consideration in scheduling a worker for monitoring, whether by direct or indirect methods, is to ensure that an intake above a predetermined level is not 'missed' [16]. An intake could be missed if, as a result of radioactive decay and biological clearance, the body content or daily excretion of the radionuclide were to decline to a level below the minimum significant activity (MSA) of the measurement during the time interval between the intake and the measurement (see Appendix II for further details). The fraction of an intake remaining in the body for direct measurement or being excreted from the body for indirect measurement, $m(t)$, depends on both the physical half-life and the biokinetics of the radionuclide, and is a function of the time since intake. Thus, an intake I and the resulting committed effective dose $E(50)$ would be missed if $I \times m(t)$ is less than the MSA. Typically, the frequency of monitoring should be set so that intakes corresponding to more than 5% of the annual dose limit are not missed.

3.30. The frequency of monitoring will thus be driven to a great extent by the sensitivity of the measurement technique. Although techniques for measurement should be as sensitive as possible, the costs of using the most sensitive techniques and the shortest possible sampling interval should be balanced against the radiation detriment associated with doses that might be underestimated or missed if less sensitive methods or less frequent measurements are used.

3.31. In any case, the bioassay method and measurement frequency adopted should be capable of detecting an intake that results in a specified fraction of the dose limit. Sometimes this goal cannot be realized because of a lack of analytical sensitivity, unacceptably long counting times for direct measurements, or unacceptably short sampling intervals for excreta collection, particularly in the case of faecal sampling to monitor inhalations of insoluble particulates. In such cases, additional methods such as improved workplace monitoring and personal air sampling should be used to ensure adequate worker protection.

Reference levels

3.32. Reference levels are helpful in the management of operations. They may be expressed in terms of measured quantities or in terms of other quantities to which measured quantities can be related, and if they are exceeded, some specified action or decision should be taken. The various types of reference level are described in the related Safety Guide [3]. In relation to intakes of radionuclides, reference levels are generally based on the committed effective dose $E(50)$. The appropriate fraction of the dose limit corresponding to each type of reference level (see below) should be established with other sources of exposure taken into account. Investigation levels and recording levels are of relevance to monitoring for internal contamination in the case of occupational exposures.

Investigation level

3.33. An investigation level is "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" [2]. For intakes of radionuclides, the investigation level relates to a value of committed effective dose above which a monitoring result is regarded as sufficiently important to justify further investigation. The investigation level set by management will depend upon the objectives of the programme and the type of investigation to be carried out.

3.34. For routine monitoring, the investigation level for an intake of a radionuclide is set in relation to the type and frequency of monitoring, as well as the expected level and variability of intakes. The numerical value of the investigation level depends on a knowledge of the conditions in the workplace. An investigation level may be set for individuals involved in a particular operation, either routinely or on an occasional basis, or may be devised for individuals within a workplace without reference to a particular operation.

3.35. As an example, for a routine operation with routine monitoring, an investigation level IL may be set on the basis of a committed effective dose of 5 mSv (0.005 Sv) from a year's intakes. Thus, for N monitoring periods per year, the investigation level (in Bq) for the intake of any radionuclide j in any monitoring period would be given by:

$$\mathrm{IL}_j = \frac{0.005}{N\, e(g)_j}$$

where $e(g)_j$ is the appropriate dose coefficient for inhalation or ingestion.

Recording level

3.36. A recording level is defined as "a level of dose, exposure or intake specified by the regulatory authority at or above which values of dose, exposure or intake received by workers are to be entered in their individual exposure records" [2]. As an example, the recording level RL for an intake of a radionuclide could be set to correspond to a committed effective dose of 1 mSv (0.001 Sv) from a year's intakes. Thus, for N monitoring periods per year, the recording level for intake of radionuclide j in a monitoring period would be given by:

$$\mathrm{RL}_j = \frac{0.001}{N\, e(g)_j}$$

Derived levels

3.37. The quantities actually measured in individual bioassay programmes are radionuclide activities in the body or excreta samples, and it is therefore convenient to establish reference levels for the measurement results themselves. These are termed derived investigation levels (DILs) and derived recording levels (DRLs). They are measurement results that imply radionuclide intakes or committed effective doses at the corresponding reference levels. Derived investigation and recording levels are calculated separately for each radionuclide. They are specific to the radiochemical form in the workplace, and are a function of time since intake. For the examples given above,

$$\mathrm{DIL}_j = \frac{0.005}{N\, e(g)_j} \times m(t_0)$$

where t_0, the typical time elapsed since intake when a bioassay sample is taken, is usually calculated as $365/2N$ days, based on the assumption that the intake occurs at the mid-point of the monitoring period, and

$$\mathrm{DRL}_j = \frac{0.001}{N\, e(g)_j} \times m(t_0)$$

Even if the resulting dose is below that associated with the recording level, the measurement results should always be maintained in the radiation monitoring records for the workplace and for the individual [17] (see also Section 8). In cases of worker exposure to external radiation or to multiple radionuclides, management may decide to reduce the derived levels for individual radionuclides appropriately.

Use of material specific and individual specific data

3.38. Biokinetic models for most radionuclides in their commonly encountered forms, with reference parameter values, have been published by the ICRP (see Section 6). These models are based on Reference Man [18] and the observed behaviour of radionuclides in humans and animals. They have been developed for defined chemical forms of radionuclides and are generally of use for planning purposes. As mentioned above, the particular workplace conditions should be characterized to determine which forms are actually present. It is likely that, in some circumstances, the chemical or physical forms of the radionuclides in use in a given workplace will not correspond to the reference biokinetic models. In this circumstance, material specific models may need to be developed.

3.39. If intakes are small, for example corresponding to a few per cent of the dose limit, the reference models are likely to be adequate for estimating the resulting doses. However, if the estimate of an intake corresponds to about a quarter or more of the dose limit, biokinetic model parameters specific to the material(s) and individual(s) in question may need to be developed to estimate the committed effective dose more accurately. Such biokinetic models can be developed from sequential direct and indirect measurements of the exposed workers. Analysis of workplace air and surface contamination samples can also assist in the interpretation of bioassay measurements, for example by measuring the ratio of ^{241}Am to $^{239+240}$Pu when direct measurement of ^{241}Am in the lung is used to assess plutonium intakes or for assessing the solubility of inhaled particles [13, 14].

3.40. A common example of the need for material specific information is where the size of the particles that a worker would be likely to inhale differs significantly from the assumption of 5 μm AMAD recommended by the ICRP as a default value for the workplace [19]. In this case, the fractions of inhaled radioactive materials deposited in the various regions of the respiratory tract would have to be determined from the ICRP respiratory tract model (see Section 6) [19] and an appropriate dose coefficient calculated. More specific information may also be needed on the solubility characteristics of the material after inhalation or ingestion as appropriate. This can be obtained from experimental studies in animals or by in vitro solubility studies. Retrospective determination of particle characteristics following an exposure may be difficult and consideration should be given to obtaining material specific information when setting up worker monitoring programmes.

3.41. Even if all of the assumptions in the reference biokinetic models are appropriate for a given workplace, there will still be differences between individuals in excretion rates and other biokinetic parameters for the same intake of a radionuclide. The

variability between individuals, and even in the daily excretion rate for the same individual, will often be more significant than the differences between a reference biokinetic model and one developed specifically for a given individual. To reduce some of this variability, collection periods for excreta samples should be sufficiently long, for example 24 hours for urine and 72 hours for faeces. The use of individual specific model parameters should be rare under routine circumstances.

Task related monitoring

3.42. Task related monitoring is, by definition, not routine, i.e. it is not regularly scheduled. Such monitoring is conducted to provide information about a particular operation and to give, if necessary, a basis for decisions on the conduct of the operation. It is particularly useful when short term procedures are carried out under conditions which would be unsatisfactory for long term use. Task related monitoring is usually conducted in the same way as routine monitoring, unless the circumstances of the operation dictate otherwise, for example if the radionuclides involved may be different or if the probability or potential magnitude of internal exposure may be significantly greater.

Special monitoring

3.43. Special monitoring may be necessary as a result of a known or suspected exposure, or an unusual incident, such as a loss of containment of radioactive materials as indicated by an air or surface sample, or following an accident. It is most often prompted by a result of a routine bioassay measurement that exceeds the derived investigation level. It may also result from occasional samples such as nose blows, swipes or other monitoring.

3.44. Special monitoring prompted by an incident is not usually conducted any differently from a routine measurement in terms of measurement techniques, although improved sensitivity or a faster processing time may be needed. The laboratory should be advised that the sample analysis or the direct measurement has priority over routine measurements, and the frequency of subsequent monitoring may be changed. The laboratory should also be informed that samples may have a higher than normal level of activity, so that the measurement technique can be tailored to the special monitoring situation and any necessary precautions taken to prevent contamination of other samples.

ASSESSMENT FOLLOWING ACCIDENTS OR INCIDENTS

3.45. There will be situations involving the use of radioactive material in which the operational controls break down. Accidents or incidents may result in releases of

radioactive materials into the working environment with the potential for high doses to the workforce.

3.46. After an accident has occurred, the radiological consequences may be complicated by trauma or other health effects incurred by the workers. Medical treatment of injuries, especially those that are potentially life threatening, generally takes priority over radiological operations, including exposure assessment. In such cases, post-accident exposure assessment should be conducted when the situation has been brought under control.

3.47. Once assessment of internal exposure has commenced, as much information should be gathered as is practicable. For example, information will be needed on the time and nature of the incident and the radionuclides involved, and on the timing of bioassay samples and measurements of body activity. This information may be necessary not only for exposure assessment, but also to assist in medical assessment, to guide medical treatment of the victim (which may include chelation therapy or wound excision), and to assist later in reconstruction of the accident or incident itself and in long term medical follow-up of the victim [20, 21].

3.48. Because intakes associated with accidents or incidents can result in committed effective doses which approach or exceed dose limits, individual and material specific data are normally needed for exposure assessment. These data include information on the chemical and physical forms of the radionuclide(s), the particle size, airborne concentrations, surface contamination levels, the retention characteristics in the individual affected, nose blows, face wipes and other skin contamination levels and external dosimetry results. The various items of data will often seem to be inconsistent or contradictory, particularly if the intake period is uncertain. An adequate assessment of dose can be made only after considering all of the data, resolving the sources of inconsistency as far as is possible, and determining the most likely and worst possible scenarios for the exposure and the magnitude of any intake.

Direct and indirect methods

3.49. The primary factor in deciding between direct and indirect methods of internal exposure monitoring after an accident or an incident will be the radiological characteristics of the radionuclides involved. If the victim is externally contaminated with gamma emitting radionuclides, direct measurements should normally be delayed until the victim has been decontaminated, both to prevent interference with the measurement and also to avoid contamination of the direct measurement facility [22, 23]. Occasionally, the urgency of assessment may preclude complete decontamination, in which case the individual could be wrapped in a clean sheet to minimize

contamination of the facility. The result of this initial direct measurement would set an upper limit for the body content, but more measurements would be needed after further decontamination [24]. External contamination with alpha or pure beta emitters will normally not interfere with direct measurements, unless bremsstrahlung is produced by the beta emitter(s). External contamination will not interfere with indirect methods, provided that care is taken to avoid transfer of contamination to excreta samples. On rare occasions, intakes may be so high that special techniques are needed for either direct or indirect measurements to avoid interference with equipment response, such as excessive electronic dead times [22, 23].

3.50. Following an accident or incident, analyses of samples of urine and faeces should be considered to verify the intake of radioactive material. However, the results of such analyses are frequently difficult to interpret, because of the potential for multiple routes of intake and imprecise knowledge of the amount of radionuclide transferred to the blood from points of intake. Excreta sample measurements are generally not useful for intake assessment immediately after an accident or incident because of the delay between intake and excretion; this is particularly the case for faecal excretion. In addition, rapid early components of urinary excretion can be difficult to interpret as they are not fully defined in some biokinetic models. Nevertheless, all excreta should be collected following an accident or incident; the early detection of radioactive materials in a urine sample can be a useful indication of the solubility of the radioactive material involved and of the potential for effective treatment. Excreta analyses can be the only reliable method of assessing intakes if large amounts of external contamination interfere with direct measurements.

3.51. In view of the general principle of emphasizing non-invasive procedures, invasive procedures such as blood sampling will usually be justified only in accident situations in which large intakes may have occurred. Blood sampling can provide data on the solubility and biokinetics of the material involved, but is generally of limited value for providing quantitative estimates of the intake because of the rapid clearance of most radionuclides to other tissues.

3.52. Workplace monitoring samples, such as air filters and surface contamination wipes, should be analysed to determine the radionuclides involved, isotopic ratios and their physicochemical characteristics.

Follow-up monitoring

3.53. Both direct and indirect follow-up monitoring programmes should be conducted at reasonable intervals for an extended period after an accident or incident. This information will help in establishing the biological half-lives of radionuclides in

the body tissues and their excretion rates. This, in turn, can help to improve the accuracy of dose assessment.

Schedule of sampling

3.54. Following an accident or incident, excreta samples for indirect monitoring should be collected until such time as a reasonable estimate can be made of the temporal pattern of excretion. If decorporation therapy, such as the administration of chelating agents, is used [20], samples should continue to be collected in order to determine the effectiveness of the treatment. Once excretion patterns have stabilized, individual samples collected during the course of a day may be combined into 24-hour samples, and appropriate aliquots taken for analysis.

3.55. If direct measurements are feasible, they should be continued at regular intervals if the subject's medical condition permits. The frequency of direct measurements will be determined by the clearance and decay rates of the internally deposited radioactive materials. Sequential direct measurements of specific organs or body regions can also assist in determining the biokinetics of the activity. For example, sequential measurements of inhaled ^{241}Am can demonstrate the clearance from lung and translocation to bone and liver [25]. In the case of deposits in cuts or wounds of some insoluble forms of radioactive materials, follow-up monitoring may reveal deposition in regional lymph nodes as a consequence of lymphatic clearance, with slow clearance from these sites [26, 27].

4. DIRECT METHODS

INTRODUCTION

4.1. The most accurate assessments of internal dose can be made when the distribution and total body content of an incorporated radionuclide can be determined reliably by direct in vivo counting of emissions from the body. Nevertheless, biokinetic modelling of retention and biophysical modelling of energy deposition may still be needed to calculate the intake and the committed effective dose, so direct methods can also depend on the interpretation of rates of excretion, which often vary markedly over time and between individuals.

4.2. Direct measurement is possible when the incorporated radionuclide(s) emit(s) penetrating radiation (normally X ray or gamma photons, including bremsstrahlung)

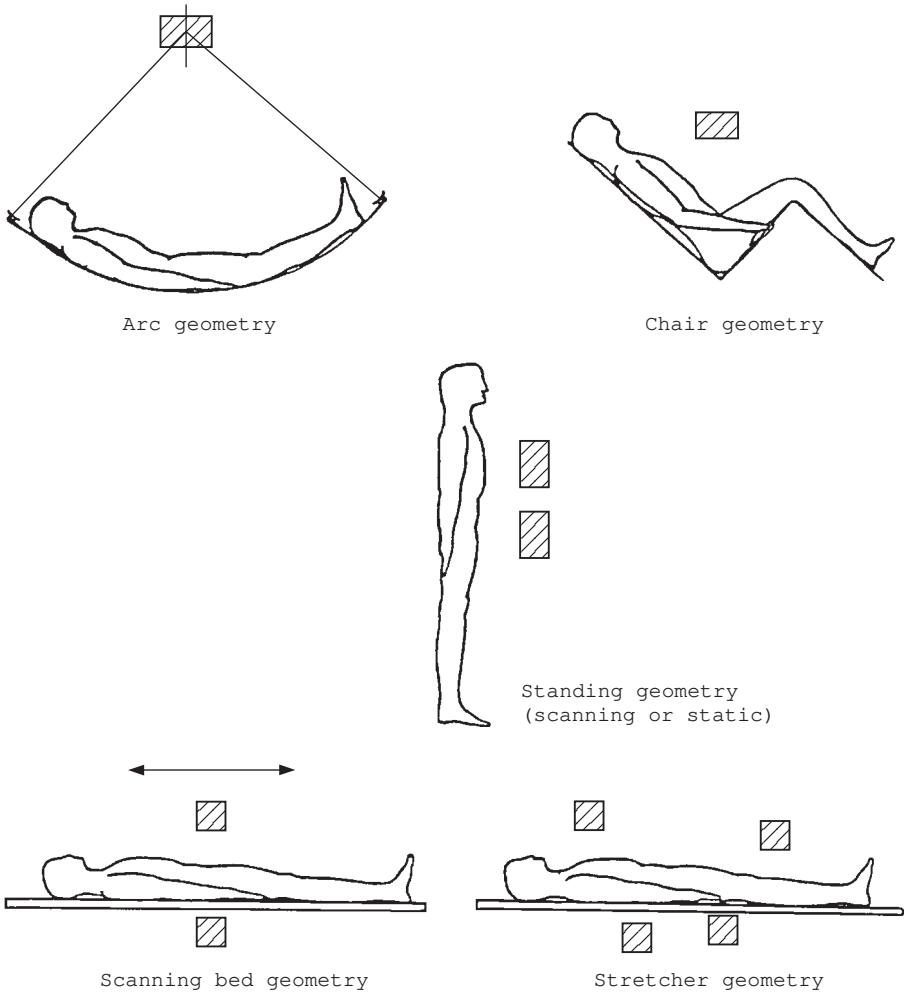


FIG. 1. Various geometries used for whole body monitoring.

of sufficient energy and yield to be detectable (Appendix II) outside the body. A detailed description of the methods commonly used in direct measurement can be found in Ref. [7]. For most in vivo counting applications, photon detectors are positioned at specified locations around the body, usually with at least partial shielding of the detector and/or the subject to reduce interference from ambient external sources.

MEASUREMENT GEOMETRIES

4.3. A variety of physical arrangements of detectors has been developed to serve specific purposes. For radionuclides which are distributed throughout the body, counting of the whole body, or a large fraction of it, provides the greatest sensitivity. Whole body counting is carried out either using a static geometry, with one or more detectors, or by scanning — moving the subject with respect to static detectors or moving detectors around a static subject. Static geometries commonly comprise an array of detectors distributed along a standing or supine subject, or a single detector directed towards the centre of a subject on a tilted chair or curved frame. Some examples of counting geometries are shown in Fig. 1.

4.4. For other radionuclides which are at least temporarily concentrated in particular organs or tissues of the body, monitoring of specific sites is recommended. Examples are radioiodine, which is taken up by the thyroid, and inhaled radioactive particles which are retained in the lungs. Localized monitoring is also recommended when intake is through a wound, or when there are other reasons for determining the distribution of the radionuclide(s) within the body.

4.5. In all cases the method should be to compare the signal measured from the subject with that obtained under the same conditions from an anthropomorphic phantom, or other surrogate, containing known quantities of the radionuclide in question. The distribution of the radionuclide in the calibration phantom should match that expected in the human subject as far as possible, although some measurement techniques are more sensitive than others to this distribution. Whole body counting is unlikely to fail completely to detect a significant amount of localized activity, but might not provide an accurate estimate of the amount or give good information on its spatial distribution.

METHODS OF DETECTION

4.6. A variety of detection systems are in use for different purposes. Inorganic crystals of high atomic number materials, usually thallium-activated sodium iodide, NaI(Tl), are commonly used to detect energetic photons (above 100 keV), such as those emitted by many fission and activation products. Scintillations produced by the crystal's interaction with high energy photons are detected by photomultiplier tubes; these generate electronic pulses which are processed to produce a spectrum reflecting that of the radiation absorbed by the crystal. This type of measurement system is most suited to cases where a small number of radionuclides are present; the energy resolution is limited, so that even deconvolution techniques may be unable to determine the radionuclides giving rise to a complex spectrum, such as that from a fresh fission

product mixture. For many circumstances, however, this approach provides the most sensitive method of quantifying body content.

4.7. Semiconductor detectors have major advantages in energy resolution, and so allow almost unambiguous identification of the radionuclides in a mixture, but are inconvenient in that they need cooling to liquid nitrogen temperatures. High purity germanium (HPGe) detectors can tolerate cycles to room temperature but need cooling during operation. Furthermore, many semiconductor detectors are available only in fairly small sizes, so that their sensitivity is decreased relative to inorganic crystals and other scintillators. Compact arrays of three to six detectors are becoming standard for monitoring contamination in specific organs such as the lungs.

4.8. Low energy photons, such as those emitted by ^{239}Pu (13–20 keV) and by ^{241}Am (60 keV), can be detected with thin NaI(Tl) crystals, which have a similar detection efficiency to larger crystals but much lower background. The addition of a second crystal, usually of CsI(Tl), as an anticoincidence guard improves the detection sensitivity by eliminating the contribution of high energy photons. Such a device, which is commonly known as a phoswich (phosphor sandwich) detector, can lower the detection limit for these photons by more than an order of magnitude. Arrays of HPGe detectors are increasingly used for the detection of low energy photons, because of their high resolution and low background. For low energy photon counting (using, for example, phoswich or HPGe detectors), account must be taken of the overlying tissue thickness in determining detection efficiency.

4.9. Miniature semiconductor detectors, in particular those using cadmium telluride (CdTe) operating at room temperatures, are becoming increasingly available. CdTe detectors offer high sensitivity for detection of low energy photons. Their small size (approximately 10 mm in diameter and 2 mm thick) make them ideal for localized wound monitoring. Their additional advantages are that there is no need to confine a worker in a shielded enclosure and that quick assessment of the success of a surgical excision procedure is possible. These small size detectors are not, however, suitable for the identification and quantification of radionuclides by spectrometry.

4.10. In setting up an advanced in vivo monitoring facility it would generally be recommended that a variety of detection systems be installed, appropriate for the specific radionuclide(s) likely to be of concern.

MEASUREMENT PROCEDURES

4.11. Subjects for direct measurements should be free of external surface contamination and in fresh clothing, often disposable paper garments. Accessories such as

jewellery, watches and spectacles should be removed. Such precautions help to avoid false identifications of internal activity, and also prevent the transfer of contamination to the counting equipment. Individuals should, to the extent practicable, be in a defined counting position, to ensure reproducibility in serial measurements and to improve comparison with calibration results. In some cases the subject will need to remain stationary for periods of up to an hour for satisfactory precision in the measurement. Some means of communication should be provided for subjects in enclosed shielding, particularly when extended counting periods are necessary.

4.12. 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 activity in the shielding and other equipment;
(c) Radiation from natural radioactivity in the subject;
(d) Radiation scattered into the detector by interactions of the subject with ambient radiation.

For counting systems based on scintillation counting (NaI(Tl) crystals or phoswich detectors), background counts for the detector system should therefore 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 results. However, exact matching will not be possible and factors such as ^{40}K content cannot be controlled, and therefore better results can be obtained from matched control groups, or from measurements on the specific individual made before starting work. Measurements of background in the counter should be made as close as possible in time to the measurement of the subject, ideally just before and just after. When using semiconductor detectors, background counting with matching phantoms is not necessary.

4.13. Quality assurance requirements are discussed in Section 9.

5. INDIRECT METHODS

INTRODUCTION

5.1. Indirect monitoring is based on the determination of activity concentrations in biological materials separated from the body — usually urine, faeces, breath or blood — or in physical samples taken from the work environment, such as samples of air or of contamination from surfaces.

5.2. Indirect methods are most suitable for those radionuclides, such as tritium, that do not emit strongly penetrating radiation to any significant extent. For some other radionuclides, such as those which emit only low energy photons, the insensitivity of, and uncertainties in, the direct monitoring measurement may be such that an indirect method can provide a more reliable estimate of intake, despite its dependence on the interpretation of measurements through biokinetic models of processes which may vary with time and between individuals. In other cases, indirect methods may be more practicable than direct monitoring and be sufficiently accurate.

BIOLOGICAL SAMPLES

5.3. The biological samples most commonly used for the estimation of intakes are urine and faeces, but breath, blood or other samples are used in special cases. For example, the analysis of activity in a nose blow or nasal swab provides an early estimate of the identities and relative levels of radionuclides in an inhaled mixture. In this case, however, the relationship between the activity concentration in the sample and the intake is so uncertain that such data can provide only a crude indication of the size of the intake.

5.4. The choice of bioassay sample will depend not only on the major route of excretion, as determined from the physicochemical form of the intake and the biokinetic model for the element(s) involved, but also on such factors as ease of collection, analysis and interpretation. Urine samples are readily obtained and analysed and generally provide information on the intake of radionuclides in chemical forms that are readily transferred to the blood. Intakes of insoluble material can often be reliably assessed only from faecal samples.

Urine

5.5. Following the entry of radionuclides into the blood and systemic circulation, clearance from the body will generally be via the urine. Urine contains waste and

other materials, including water, extracted by the kidneys from the blood, and collected for up to several hours or more in the bladder before voiding. Because of this mixing in the bladder, radionuclide levels in samples of urine obtained soon after an acute intake should be interpreted with caution. The bladder should be cleared soon after the intake, and then a second and subsequent samples obtained. All samples should be analysed.

5.6. After the first few days, 24-hour samples of urine normally provide the best basis for assessing intake. In circumstances where 24-hour samples have not been obtained, total excretion can be estimated from creatinine measurements. In routine monitoring for radionuclides with prompt components of excretion, consideration should be given to the day on which samples are taken, since there can be significant differences between samples taken before and after even short periods free from exposure.

5.7. For intakes of tritiated water, the concentration of tritium in urine is the same as in body water and can be used to assess body content and dose rate without reference to an excretion model.

Faeces

5.8. Faecal samples contain water, cellular debris lost from the wall of the gastrointestinal tract, unabsorbed waste products transported through the gastrointestinal tract, including insoluble materials cleared from the lung, and metabolic products cleared from the liver in bile. The mass and composition of individual faecal voidings can be quite variable, and depend strongly on diet. For this reason, reliable estimates of daily faecal excretion rates of radioactive materials can usually be based only on total collections over 3–4 days. Single samples should, in most cases, only be used for screening purposes.

5.9. Post-vacation measurements allow for differentiation between the fraction of inhaled radionuclides cleared rapidly through the gastrointestinal tract and the delayed clearance of systemic activity and long term deposits of insoluble forms of radionuclides in the lung. In the monitoring of workers chronically exposed to long lived radionuclides, therefore, faecal samples should ideally be collected after a vacation (at least ten days absence from work) and prior to return to the working environment.

Breath

5.10. Breath is a significant route of excretion only for those few materials which are exhaled directly or metabolized to gases or volatile liquids. However, for these cases, breath samples can provide a convenient way of measuring the activity of excreta,

free from most other sources of radioactive contamination. For radon and thoron produced in the body from intakes of ^{226}Ra and ^{228}Ra, models are available that have been used for dose assessment purposes [28].

Blood

5.11. Blood samples provide the most direct source for estimating radionuclides present in the systemic circulation, but are not often used because of medical constraints on the sampling process. With only a few exceptions (e.g. HTO, ^{59}Fe and ^{51}Cr in labelled erythrocytes), blood samples provide very limited information on the total systemic activity following an intake, because of rapid clearance from the blood stream and deposition in tissues.

Nose blows

5.12. Nose blows should not be used to estimate an intake, but can be very useful in task related and special monitoring to indicate the need for additional sampling and analysis, especially when exposure due to actinides may have occurred. They can also be used to identify the components in a mixture of radionuclides.

Tissue samples

5.13. For localized deposits of radionuclides with high radiotoxicity (e.g. transuranic elements) in a wound, it is usually advisable, subject to medical advice, to excise the contamination soon after the intake. Radiochemical analysis of excised tissue by destructive and/or non-destructive methods can provide information on the radionuclides and their relative concentrations, and may assist in assessing the uptake to blood and in determining the course of further actions.

5.14. Other biological samples, such as hair and teeth, can be used to assess intakes although, in general, they cannot be used for quantitative dose assessments. Tissue samples taken at autopsy may also be used to assess the body content of radionuclides.

PHYSICAL SAMPLES

5.15. Physical samples include air samples, surface wipes and smears, and other materials from the workplace that can be used to identify the physicochemical form of radioactive contaminants. Assessments of intakes based on activity in physical samples are particularly uncertain because of the large variability in workplace conditions and the potential for intakes by individuals. Furthermore, assessment must

usually be based on a single sample from the initial step in the exposure process, which cannot be repeated. Nevertheless, for those radionuclides which emit no strongly penetrating radiation and which are found in only low concentrations in excreta, such as some inhaled actinides, interpretation of such physical samples can provide a basis for assessment. These samples can also serve as an indication of the need for additional individual monitoring.

Air samples

5.16. Air samples may be drawn from the ambient room atmosphere by fixed samplers, or from the breathing zones of workers by personal air samplers (PASs). For compounds that disperse readily in air, such as radioactive gases and vapours (e.g. $^{14}CO_2$ and tritiated water), samples from static samplers can provide a reasonable representation of inhaled radioactive material, especially in small rooms. For other sources, however, such as resuspended particulates, such samples may lead to estimates of the activity of the material inhaled that are wrong by an order of magnitude or more, depending on the relative locations of the source, the sampler and the worker.

5.17. More representative samples will be derived from PASs, which are self-powered systems carried by the worker that draw samples from the immediate breathing zone at a regular rate. Even these samples, however, may lead to over- or under-estimation of intakes, depending on the applicability of assumptions about particle size and breathing rates. To reduce this uncertainty, some PAS systems prevent particles of non-respirable size from reaching the filter [15].

5.18. Both forms of sampling rely on the extraction of radioactive material from the passing air on a collection medium. This medium will be specific, to some extent, to the material to be collected. For example, particulate material can be removed on coarse fibre filters, while charcoal beds are employed to sample radon gas and iodine vapour, and tritiated water can be collected in a water trap.

5.19. Analysis of the particle size and solubility of samples of airborne radioactive material can assist in the development of biokinetic models for dose assessment (Section 6). Direct comparison of air samples with values of derived air concentrations (Section 2) can be used as an input to the evaluation of workplace conditions and to the estimation of doses.

Surface samples

5.20. Because modelling of the transfer of radioactive materials from surfaces into the body is particularly uncertain, samples of radionuclide concentrations on surfaces

are used primarily to indicate the potential for significant intakes and the need for individual monitoring. Such samples can also indicate the relative amounts of various radionuclides in a mixture and the presence of any radionuclides not detected in a bioassay sample.

5.21. Surface samples are usually obtained by wiping a defined area of the surface with materials such as filter papers or cotton swabs. These materials are chosen for their ability to transfer the expected contaminants from the surface and to release them as needed for analysis. The efficiency of collection should be determined for the particular combination of surface and wiping material, but is likely to be around 10% for a moist swab on a moderately porous surface.

HANDLING OF SAMPLES

5.22. Special care should be taken in the handling of samples to be used for the assessment of internal exposure; firstly, to avoid the transfer of radioactive or biological contamination during handling and, secondly, to ensure a traceable link between the analytical result and the original sample, as required by the quality assurance programme (see Section 9).

5.23. With respect to the potential hazard from contamination, both biological and radioactive contaminants should be considered. Biological samples may contain pathogens, such as bacteria and viruses. These pathogens will be potentially active until the complete sample has been turned into ash or otherwise sterilized. All such samples should, therefore, be stored at reduced temperature, preferably frozen, until analysis. This treatment will also reduce unwanted biological degradation of those materials, such as organically bound tritium, for which the molecular form is an important factor in the subsequent analysis. Another way to prevent degradation is to treat the sample with acid.

5.24. To establish traceability, a chain of custody should be maintained such that at each step in the collection, transport and analysis of the samples, documentation is created to describe and verify the transfers that have occurred.

5.25. Urine, faeces and other biological samples should not be collected in contaminated areas, to ensure that activity measured in the sample is representative of body clearance. The sample should be clearly marked to show the worker's identity and the date and time of sample collection.

5.26. Those responsible for decisions concerning the type(s) of analysis to be performed on the sample should be informed about the areas in which the worker may have been exposed, especially if the sample is likely to have high levels of activity, as may be the case for special monitoring (Section 3). It is also important that they be aware of the use of any medication or treatment that may interfere with the sample analysis or its interpretation.

METHODS OF ANALYSIS

5.27 Analysis of biological or physical samples involves the detection and quantification of emissions from the radionuclides present by appropriate instrumentation. In many cases, the radionuclides must first be separated from the sample matrix to allow sensitive and reproducible detection. In some other cases, limitations of the detectors prevent discrimination between radionuclides that have similar emissions (e.g. some actinides); in these cases, the samples must be subjected to chemical separation of the elements (radiochemical separation) before counting.

Detection

5.28. Instrumentation for radiometric assessment can be divided into three classes, that for measuring alpha particles, that for beta particles and that for photon emissions.

5.29. Alpha particles can be detected by a variety of techniques, each having advantages and disadvantages. The simplest gross count of total alpha activity can be made using a ZnS detector or a gas flow proportional counter. These methods are efficient, but do not discriminate between alpha particles of different energies and therefore cannot identify or quantify individual radionuclides in a mixture. After radiochemical separation, alpha spectroscopy methods using semiconductor detectors or gridded ionization chambers can quantify individual radionuclides, provided that their energies are sufficiently different, but generally need long counting times to achieve adequate sensitivity. Other methods, such as alpha track etching, are even more sensitive for special applications, but can take periods of a month or more for a complete analysis and may not be able to separate different alpha particle energies.

5.30. Beta particles are most commonly detected by liquid scintillation counting, especially for low energy beta emitters. In some cases separation of two or more beta emitters in a mixture, such as tritium, ^{14}C and ^{32}P, can be achieved by setting energy windows on the detector response. Gross measurements of high energy beta emitters deposited on planchettes or filters can be obtained using gas flow Geiger–Müller or proportional detectors.

5.31. Photon emissions from physical or biological samples are usually detected by NaI(Tl) scintillators or semiconductor detectors such as high purity Ge (see Section 4). Special counting methods are needed for very low energy X rays, such as those emitted by several radioisotopes of transuranic elements.

5.32. Non-radiometric techniques are also available. For example, ultraviolet radiation fluorimetry can be used for the assay of uranium, irrespective of the degree of enrichment. Other techniques, such as fission track analysis, neutron activation analysis and inductively coupled plasma mass spectrometry (ICP/MS), can be used to measure specific radionuclides, but are expensive and will be necessary only in special circumstances. Counting times for all of these methods will depend upon the activity in the sample, the measurement equipment employed and the precision needed (see Appendix II).

Radiochemical separation

5.33. In many cases, radionuclides should be separated from the sample matrix, or from radioisotopes of other elements, before counting, in order to reliably quantify activity. This process is, to a large extent, specific to the elements being separated, but generally includes sample preparation and preconcentration, purification, source preparation and yield determination. In general, a variety of approaches can be applied to isolate a specific radionuclide from sources of interference in order to improve detection. An essential element of the process is to trace the recovery of the radionuclide through each step so that the final result can be reliably related to the concentration in the initial sample. Appropriate blank samples should be prepared to measure the background.

6. BIOKINETIC MODELS FOR INTERNAL DOSIMETRY

INTRODUCTION

6.1. Intakes of radionuclides can occur via a number of pathways. In occupational exposure, the main route of intake is by inhalation, although a fraction of any material deposited in the respiratory system will be transferred to the throat and swallowed, giving the opportunity for absorption in the gastrointestinal tract. Intakes by direct ingestion may occur, as, for some radionuclides, may absorption through the intact skin. Damage to the skin by cuts or other wounds can also result in intakes of radionuclides (Fig. 2(a)).

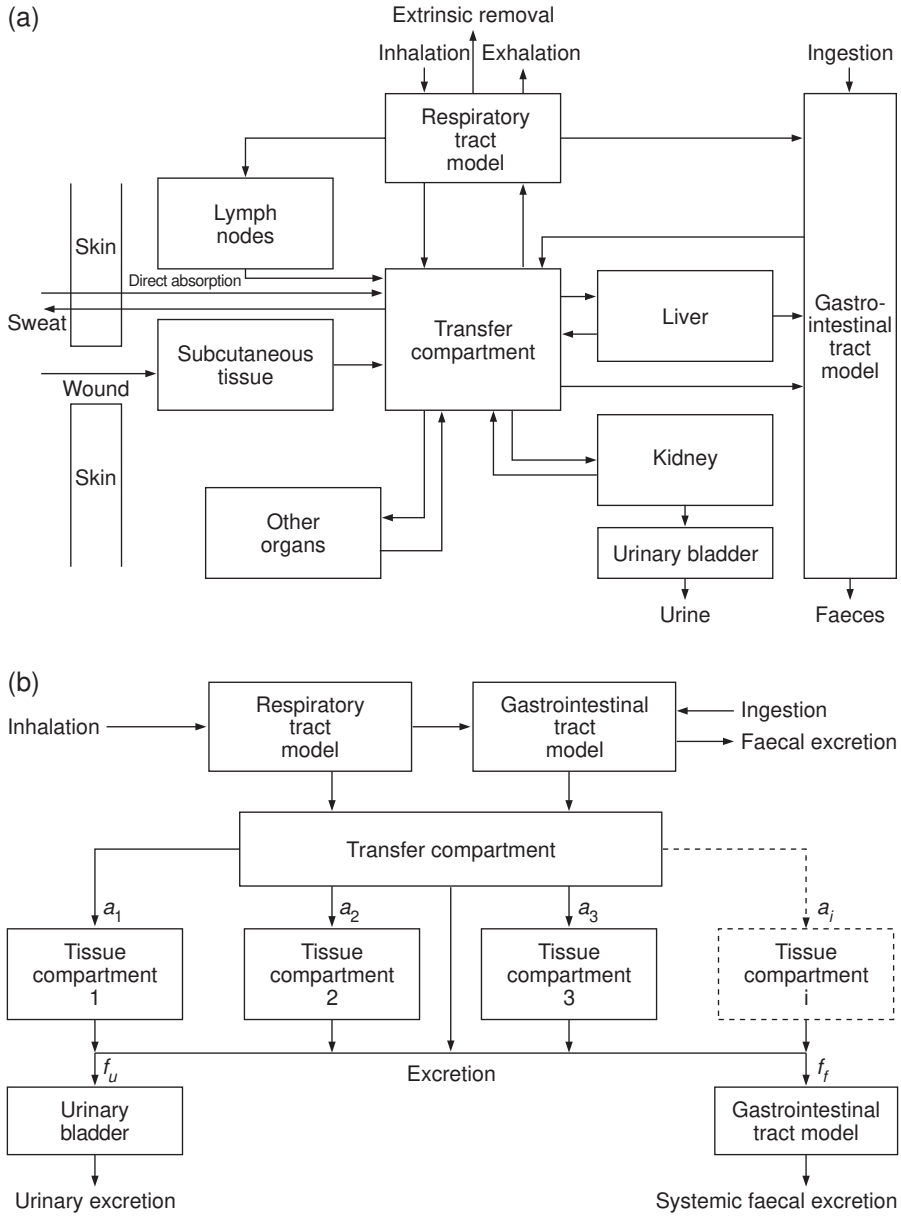


FIG. 2. (a) Routes of intake, transfers and excretion (based on [8]); (b) general model used to represent the kinetics of radionuclides in body compartments (exceptions are noted in the metabolic data for individual elements) (based on [8, 9]).

6.2. Recommendations have been made by the ICRP on methods for assessing intakes of radionuclides, and the resulting doses, from monitoring data [8, 9]. For workers who are occupationally exposed, the ICRP has developed a suite of models to represent the behaviour of radionuclides that have entered the body either by inhalation or ingestion. These models can be applied for regulatory control of the workplace.

6.3. For other routes of exposure, intakes are only likely to occur as a result of accidents, the exact nature of which cannot readily be predicted. Almost no internationally accepted models have therefore been developed that relate to entry of radionuclides through the intact skin or through wounds, although some information on the latter has been published [26]. An exception is tritiated water, which is readily absorbed through the intact skin. This may be assumed to result in an additional intake of tritium, equal to 50% of the activity of tritium inhaled, for exposure in the workplace [29] and is regulated by setting appropriate DACs. Thus, a more useful reference value for tritiated water in air would be two-thirds of the DAC given in Table A–III.

6.4. In Publication 26, the ICRP [30] introduced the use of (tissue) weighting factors w_T to calculate the committed effective dose equivalent from individual tissue dose equivalents. This provided a common way of expressing doses from external radiation, which are relatively uniform to all body tissues, and from intakes of radionuclides, which can be very heterogeneous. The ICRP applied this advice in the various parts and supplements of its Publication 30 [29, 31–33], which described the biokinetic models used for calculating dose equivalents to organs and tissues from intake by inhalation and ingestion of a wide range of radionuclides in different chemical forms. The models given in Publication 30 did not, however, fully describe the biokinetics of radionuclides within the body. They were generally simple compartment models, with transfer of material between the compartments being modelled by first order kinetics. The basis of the model structure adopted by the ICRP in Publication 30 is illustrated in Fig. 2(b). These models were designed primarily for the purposes of calculating prospective doses from incorporated radionuclides and setting limits on intake. They were not intended for the interpretation of bioassay data, although they were used for this purpose in Publication 54 [8], and are adequate for protection purposes when intakes of radionuclides are low. These models are progressively being replaced by more physiologically based models [9].

6.5. In the BSS [2] and in the 1990 Recommendations of the ICRP [6], the approach to calculating the committed effective dose is based on that used for the calculation of committed effective dose equivalent, although as a result of improved information on the late effects of radiation on the tissues of the body some changes have been made to the values of tissue weighting factors and a greater number of tissues now have specified weighting factors (see Table A–III).

6.6. The biokinetic models developed by the ICRP are intended for use in normal situations, for example for the evaluation of doses from measurements performed according to routine monitoring programmes. The evaluation of doses in accident situations needs more specific information about the time and pattern of intake, about the physicochemical form of the radionuclides and about the characteristics of the individual (e.g. body mass). Individual specific data on the biokinetics of radionuclide(s) may be obtained through special monitoring, i.e. by repeated direct measurements of the whole body or specific sites and measurements of excretion.

MODELS FOR DIFFERENT ROUTES OF ENTRY

Inhalation

6.7. Details of a new model of the human respiratory tract for radiological protection purposes have been issued by the ICRP [19]; the main features of this model are described below. This model was used in the calculation of the inhalation dose coefficients given in the BSS [2]. As in the earlier model [29], deposition and clearance are treated separately.

6.8. The main difference in approach is that, whereas the ICRP Publication 30 model calculates only the average dose to the lungs, the new model calculates doses to specific tissues of the respiratory tract (RT), and takes account of differences in radiosensitivity. In the new model, the RT is represented by five regions (Fig. 3). The extrathoracic (ET) airways are divided into the anterior nasal passage, ET_1, and ET_2, which consists of the posterior nasal and oral passages, the pharynx and the larynx. The thoracic regions are bronchial (BB), bronchiolar (bb) and alveolar–interstitial (AI), the gas exchange region. Lymphatic tissue is associated with the extrathoracic and thoracic airways respectively (LN_{ET} and LN_{TH}). Reference values of dimensions and scaling factors are specified in the model.

6.9. Deposition of inhaled particulates is calculated for each region of the respiratory tract, with account taken of both inhalation and exhalation. This is done as a function of particle size, breathing parameters and/or work load, and is assumed to be independent of chemical form. Age dependent default deposition parameters are given for a range of particle sizes from 0.6 nm activity median thermodynamic diameter (AMTD) to 100 μm activity median aerodynamic diameter (AMAD). Default deposition parameters for occupationally exposed individuals are given, based on average daily patterns of activity. Inhalation dose coefficients are given in the BSS [2] for an AMAD of 5 μm, which is now considered to be the most appropriate default particle size for radionuclides in the workplace [19]. Dose coefficients are also given

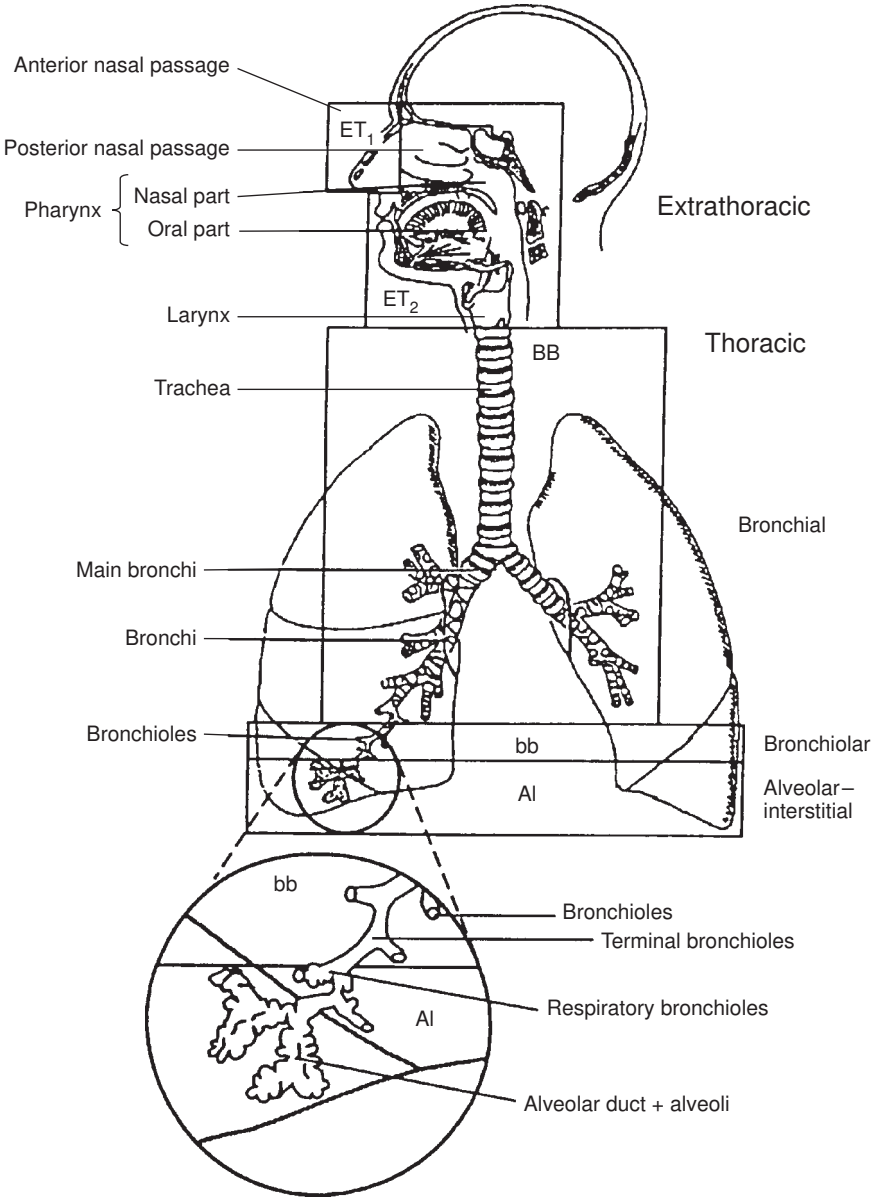


FIG. 3. Respiratory tract regions defined in the new ICRP model [19]. The extrathoracic (ET) airways are divided into ET_1, the anterior nasal passage, and ET_2, which consists of the posterior nasal and oral passages, the pharynx and larynx. The thoracic regions are bronchial (BB: trachea, and main bronchi), bronchiolar (bb: bronchioles) and alveolar–interstitial (AI: the gas exchange region). Lymphatic tissue is associated with the extrathoracic and thoracic airways (LN_{ET} and LN_{TH} respectively).

for an AMAD of 1 μm, the default value used in Publication 30 (see Table II–III of the BSS). An AMAD of 1 μm is used as a default for members of the public (Table II–VII of the BSS).

6.10. Clearance from the respiratory tract is treated as two competing processes: particle transport (by mucociliary clearance or translocation to lymph nodes) and absorption to blood.

6.11. Particle transport is treated as a function of deposition site in the respiratory tract but is taken to be independent of particle size and material. For most regions, time dependent mechanical transport is modelled by considering the region to be made up of several compartments with different clearance half-times. For example, the AI region is divided into three compartments, which clear to bb with biological half-lives of about 35, 700 and 7000 days. Similarly, bb and BB have fast and slow clearance compartments. Clearance from the AI region also involves transfer to lymphatic tissue. For bb, BB and ET, there are compartments to represent material that is sequestered in tissue and transported to lymphatic tissue.

6.12. Absorption into the blood depends on the physicochemical form of the radionuclide deposited in the respiratory system, but is taken to be independent of deposition site, with the exception of ET_1, from which no absorption is assumed. The model allows for changes in dissolution and absorption into blood with time. The use of material specific dissolution rates is recommended, but default absorption parameters are given for use when no specific information is available, namely types F (fast), M (moderate) and S (slow). These correspond broadly to the Publication 30 default lung classes D (days), W (weeks) and Y (years) respectively, although the lung classes referred to overall clearance rates from the lung.

6.13. The absorption rates for the different absorption types can be expressed as approximate biological half-lives and corresponding amounts of material deposited in each region that reach body fluids, as shown in Table I. For all three absorption types, all the material deposited in ET_1 is removed by extrinsic means, such as nose blows. In other regions, most of the deposited material that is not absorbed is cleared to the gastrointestinal tract by particle transport. The small amounts transferred to lymph nodes continue to be absorbed into body fluids at the same rate as in the respiratory tract.

6.14. For radionuclides inhaled by workers in particulate form, it is assumed that entry into and regional deposition in the respiratory tract are governed only by the size distribution of the aerosol particles. The situation is different for gases and vapours, for which respiratory tract deposition is material specific. Almost all inhaled

TABLE I. ABSORPTION TYPES

	Biological half-lives	Examples
Type F	100% absorbed with a biological half-life of 10 min. There is rapid absorption of almost all material deposited in BB, bb and AI. Half of the material deposited in ET_2 is cleared to the gastrointestinal tract by particle transport and half is absorbed.	All compounds of caesium, iodine
Type M	10% absorbed with a biological half-life of 10 min and 90% with a biological half-life of 140 d. There is rapid absorption of about 10% of the deposit in BB and bb; and 5% of material deposited in ET_2. About 70% of the deposit in AI eventually reaches body fluids by absorption.	All compounds of radium and americium
Type S	0.1% absorbed with a biological half-life of 10 min and 99.9% with a biological half-life of 7000 d. There is little absorption from ET, BB or bb, and about 10% of the deposit in AI eventually reaches body fluids by absorption.	Insoluble compounds of uranium and plutonium

gas molecules contact airway surfaces, but usually return to the air unless they dissolve in, or react with, the surface lining. The fraction of an inhaled gas or vapour that is deposited in each region thus depends on its solubility and reactivity. Generally, however, the regional deposition of a gas or vapour cannot be predicted on a mechanistic basis, from knowledge of its physical and chemical properties, but has to be obtained from an in vivo experimental study.

6.15. The new model assigns gases and vapours to three default solubility/reactivity (SR) classes, on the basis of the initial pattern of respiratory tract deposition, as shown in Table II. Subsequent retention in the respiratory tract and absorption to body fluids are determined by the chemical properties of the gas or vapour. The use of the model to calculate dose coefficients for workers is described in ICRP Publication 68 [34].

6.16. The guidance given by the ICRP and in the BSS [2] on the deposition and clearance of gases and vapours is similar to that for the respiratory tract clearance of radionuclides inhaled in particulate form. For those elements for which inhalation of radionuclides in gas or vapour form is potentially important, default SR classes and absorption types (type F or type V, very rapid absorption) are recommended to be used for gases and vapours in the absence of further information. Consideration is given only to the behaviour of gases and vapours at low mass concentrations. Dose coefficients for inhalation of soluble or reactive gases and vapours are given in Table II–IX of the BSS [2].

TABLE II. SOLUBILITY/REACTIVITY CLASSES

	Description	Examples
Class SR-0	Insoluble and non-reactive: negligible deposition in the respiratory tract	^{41}Ar, ^{85}Kr, ^{133}Xe
Class SR-1	Soluble or reactive: deposition may occur throughout the respiratory tract	Tritium gas, ^{14}CO, ^{131}I vapour, ^{195}Hg vapour
Class SR-2	Highly soluble or reactive: complete deposition in the extrathoracic airways (ET_2). For the purpose of calculation they are treated as though they were injected directly into the blood.	^{3}H in organic compounds and tritiated water

6.17. The new human respiratory tract model is more complex than the lung model given in ICRP Publication 30 [29], and has significant advantages in that it is a more realistic description of the behaviour of inhaled radioactive material and can be used with material specific data, both for assessing doses and for interpreting bioassay information.

Ingestion

6.18. The model used for the BSS [2] to describe the behaviour of radionuclides ingested by workers is that given in ICRP Publication 30 [29]. It has four compartments, representing the stomach, the small intestine, the upper large intestine and the lower large intestine. The mean residence times in the gastrointestinal tract compartments are 1, 4, 13, and 24 h respectively. The uptake to blood takes place from the small intestine and is specified by fractional uptake (f_1) values. The only changes to the model parameters for the calculation of the dose coefficients for workers given in the BSS [2] compared to those given in ICRP Publication 30 [29] were to some of the f_1 values.

Entry through wounds and intact skin

6.19. Entry through wounds and through intact skin are additional pathways by which radionuclides can enter the body. Although much of the material may be retained at the wound site, soluble material can be transferred to the blood and hence to other parts of the body. Insoluble material will be slowly translocated to regional

lymphatic tissue, where it will gradually dissolve and eventually enter the blood. A variable fraction of insoluble material can be retained at the wound site or in lymphatic tissue for the remainder of the individual's life. If particulate material enters the blood directly it deposits principally in phagocytic cells in the liver, spleen and bone marrow.

6.20. For insoluble radioactive materials retained at a wound site, the tissues around the wound will be the most exposed to radiation. Consideration may need to be given, in consultation with a physician, to the excision of contaminated local tissues. For this, the variation with depth of contamination at the wound site must be accurately determined. The absorbed dose at the wound site and in the regional lymph nodes can be assessed from the activity of the deposited material, the characteristics of the radionuclides involved, the mass of tissue irradiated and the time since exposure. If the materials are soluble, then they may translocate from the wound site to the blood at a rate which depends on their solubility. The distribution of this soluble component will, in most instances, be similar to that of material entering the blood from the lungs or gastrointestinal tract, but there may be exceptions for some chemical forms of radionuclides which enter the blood directly.

6.21. A number of materials, such as tritium labelled compounds, organic carbon compounds and compounds of iodine, can penetrate intact skin. In these cases, a fraction of the activity will enter the blood. Specific models need to be developed to assess doses from such intakes [35]. For example, the behaviour of tritiated organic compounds following direct absorption through the skin will be significantly different from that after inhalation or ingestion. For skin contamination, both the equivalent dose to the area of skin contaminated and the effective dose will need to be considered.

6.22. The biokinetic models developed by the ICRP can only be used for the calculation of the effective dose arising from the soluble component, once the systemic uptake has been determined [26].

SYSTEMIC ACTIVITY

6.23. The fraction of an intake entering the systemic circulation is referred to as the uptake. For the calculation of the dose coefficients in the BSS [2], the models recommended by the ICRP were used to describe the behaviour of radionuclides that have entered the systemic circulation. As a result of a review of data available on the behaviour of radionuclides in the body, the models recommended for a number of elements in ICRP Publication 30 [29, 31–33] have been revised, as described in

Publications 56, 67, 69 and 71 [36–39]. These revised models were also used in the calculation of the dose coefficients for workers given in the BSS [2]. The sources of the biokinetic models used for adults are shown in Table III.

6.24. A number of the revised systemic models for adults retain the model structure given in ICRP Publication 30, but with minor changes to the distribution of radionuclides between body compartments and the retention functions. In addition, the models for a number of elements have been extensively revised, in particular to take account of recycling of radionuclides between compartments. In ICRP Publication 30, a number of radionuclides (e.g. ^{239}Pu) were assumed to be 'bone surface seekers', i.e. to be retained on bone surfaces. This was known to be a conservative assumption, particularly for alpha emitting radionuclides. Evidence from animal studies and human data indicate that a fraction of plutonium becomes buried as a result of bone growth and turnover whilst a further fraction is desorbed and re-enters the blood. Of this, some may be redeposited in the skeleton and liver or be excreted. In contrast, 'bone volume seeking' radionuclides, such as ^{90}Sr and ^{226}Ra, were assumed in Publication 30 to be instantaneously distributed throughout the bone volume. In practice, the process is progressive, although it occurs more rapidly than for bone surface seeking radionuclides such as plutonium. To allow for the known behaviour of radionuclides and to take account of present knowledge of the physiology of bone, generic models for plutonium and other actinides (Cm, Am, Np and Th) [37–39] and for the alkaline earth metals (Ca, Sr, Ba and Ra) [37, 39] have been developed. The model for alkaline earth metals has also been applied, with some modifications, to lead and to uranium [37, 38].

6.25. A number of radionuclides decay to nuclides that are themselves radioactive. The usual assumption in ICRP Publication 30 was that these decay products would follow the biokinetics of their parents, although there were a few exceptions for decay products which are isotopes of noble gases or iodine. In the revised biokinetic models, separate systemic biokinetics have been applied to the parent and its decay products for intakes of radioisotopes of lead, radium, thorium and uranium.

EXCRETION

6.26. In the biokinetic models described in ICRP Publication 30, no specific information was given on excretion in urine and faeces, although the models were used in Publication 54 [8] for interpreting excretion data. In the 1990 Recommendations of the ICRP [6], however, the urinary bladder and the colon are given explicit w_T values, and in the revised biokinetic models for workers given by the ICRP [34], specific information is given on excretion pathways in the urine and faeces.

TABLE III. LUNG ABSORPTION TYPES AND SOURCES OF BIOKINETIC MODELS FOR SYSTEMIC ACTIVITY USED TO CALCULATE INHALATION DOSE COEFFICIENTS FOR WORKERS

Element	Lung absorption type(s)[a]	ICRP Publication for details of biokinetic model[b]
Hydrogen	G	56
Beryllium	M, S	30, Part 3
Carbon	G	56
Fluorine	F, M, S	30, Part 2
Sodium	F	30, Part 2
Magnesium	F, M	30, Part 3
Aluminium	F, M	30, Part 3
Silicon	F, M, S	30, Part 3
Phosphorus	F, M	30, Part 1
Sulphur	F, M, G	67
Chlorine	F, M	30, Part 2
Potassium	F	30, Part 2
Calcium	M	30, Part 2
Scandium	S	30, Part 3
Titanium	F, M, S	30, Part 3
Vanadium	F, M	30, Part 3
Chromium	F, M, S	30, Part 2
Manganese	F, M	30, Part 1
Iron	F, M	69
Cobalt	M, S	67
Nickel	F, M, G	67
Copper	F, M, S	30, Part 2
Zinc	F, M, S	67
Gallium	F, M	30, Part 3
Germanium	F, M	30, Part 3
Arsenic	M	30, Part 3
Selenium	F, M	69

TABLE III. (cont.)

Element	Lung absorption type(s)[a]	ICRP Publication for details of biokinetic model[b]
Bromine	F, M	30, Part 2
Rubidium	F	30, Part 2
Strontium	F, S	67
Yttrium	M, S	30, Part 2
Zirconium	F, M, S	56 and 67
Niobium	M, S	56 and 67
Molybdenum	F, S	67
Technetium	F	67
Ruthenium	F, S, G	56 and 67
Rhodium	F, M, S	30, Part 2
Palladium	F, M, S	30, Part 3
Silver	F, M, S	67
Cadmium	F, M, S	30, Part 2
Indium	F, M	30, Part 2
Tin	F, M	30, Part 3
Antimony	F, M	69
Tellurium	F, M, G	67
Iodine	F, G	56 and 67
Caesium	F	56 and 67
Barium	F	67
Lanthanum	F, M	30, Part 3
Cerium	M	56 and 67
Praseodymium	M, S	30, Part 3
Neodymium	M, S	30, Part 3
Promethium	M, S	30, Part 3
Samarium	M	30, Part 3
Europium	M	30, Part 3
Gadolinium	F, M	30, Part 3
Terbium	M	30, Part 3

TABLE III. (cont.)

Element	Lung absorption type(s)[a]	ICRP Publication for details of biokinetic model[b]
Dysprosium	M	30, Part 3
Holmium	M	30, Part 3
Erbium	M	30, Part 3
Thulium	M	30, Part 3
Ytterbium	M, S	30, Part 3
Lutetium	M, S	30, Part 3
Hafnium	F, M	30, Part 3
Tantalum	M, S	30, Part 3
Tungsten	F	30, Part 3
Rhenium	F, M	30, Part 2
Osmium	F, M, S	30, Part 2
Iridium	F, M, S	30, Part 2
Platinum	F	30, Part 3
Gold	F, M, S	30, Part 2
Mercury (inorganic)	F, M	30, Part 2
Mercury (organic)	F	30, Part 2
Thallium	F	30, Part 3
Lead	F	67
Bismuth	F, M	30, Part 2
Polonium	F, M	67
Astatine	F, M	30, Part 3
Francium	F	30, Part 3
Radium	M	67
Actinium	M, S	30, Part 3
Thorium	F, M, S	69
Protactinium	M, S	30, Part 3
Uranium	F, M, S	69
Neptunium	M	67
Plutonium	M, S	67

TABLE III. (cont.)

Element	Lung absorption type(s)[a]	ICRP Publication for details of biokinetic model[b]
Americium	M	67
Curium	M	71
Berkelium	M	30, Part 4
Californium	M	30, Part 4
Einsteinium	M	30, Part 4
Fermium	M	30, Part 4
Mendelevium	M	30, Part 4

[a] For particulates, F (fast), M (moderate) or S (slow); G denotes gases and vapours.

[b] Also for ingestion dose coefficients.

6.27. For assessing doses from systemic activity lost into the faeces, the model for the gastrointestinal tract is used, assuming secretion of radionuclides from the blood into the upper large intestine. A model for the urinary bladder has been adapted for calculating doses to the bladder wall [37].

DOSE COEFFICIENTS

6.28. Dose coefficients (committed effective doses per unit intake) are given in the BSS [2] for intakes by ingestion and inhalation. Dose coefficients for selected radionuclides are also given in this report, in Table A–I. These values of committed effective dose are for specific routes of intake, and cannot be used directly for assessing doses from injection into the blood or from transfer to the blood from wound sites or absorption through the skin.

6.29. For many radionuclides, dose coefficients are given for different lung absorption types and/or for different f_1 values. The most appropriate choice of value for a given situation should be based on a knowledge of the physicochemical characteristics of the materials present in the workplace. Guidance is given in the BSS (Tables II–IV and II–V) on the values of gut transfer factors (f_1) and lung absorption types for various chemical forms of the elements. In some cases, little information may be available on the characteristics of the intake, in which case the most restrictive value (i.e. the one indicating the highest dose) should be used.

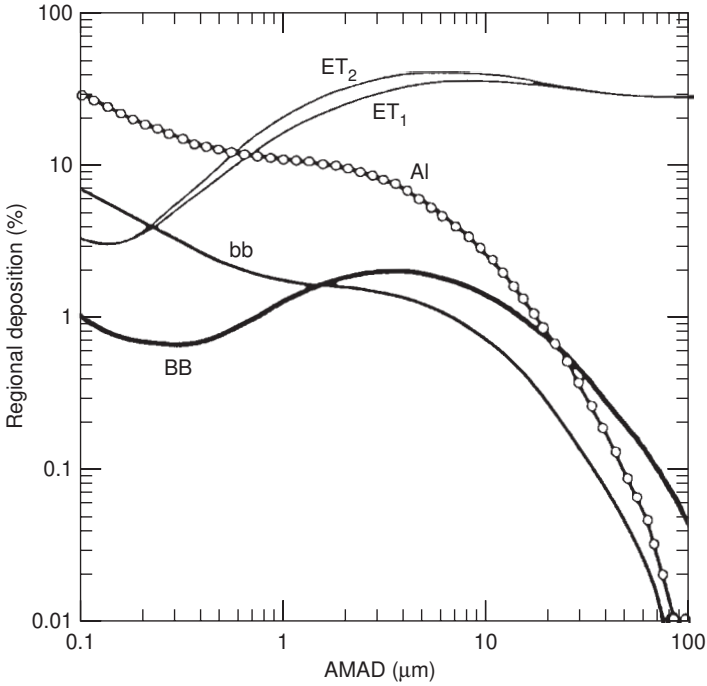


FIG. 4. Influence of particle size on deposition in the various regions of the respiratory tract [19].

WORKPLACE SPECIFIC ASSESSMENTS

6.30. In the case of significant accidental exposures, it will often be necessary to use parameter values in the calculation of tissue or organ equivalent doses and effective dose that are specific to the conditions of exposure and to the individual. Similarly, in routine situations it may be necessary to take account of the particular circumstances of exposure rather than using default parameters. The new model for the respiratory tract [34] adopts an AMAD of 5 μm as a default particle size when no specific information is available. Regional deposition of airborne particles is subject to the mechanisms of sedimentation, impaction and diffusion. Deposition throughout the respiratory system and hence inhalation dose coefficients depend upon aerosol parameters, such as the AMAD. Similarly, ingestion dose coefficients depend upon the choice of an appropriate f_1 value.

6.31. On the basis of the new respiratory tract model [19], deposition in the thorax of aerosols of occupational concern is highest in the AI region, but progressively decreases with increasing particle size (Fig. 4). The extent of deposition in each region, as well as the chemical form inhaled, can have an appreciable influence on the effective dose. Thus, for ^{239}Pu the committed effective dose for both Type M and Type S compounds decreases progressively with increasing AMAD, reflecting the decreasing deposition in the AI region and the conducting airways (BB and bb) (Figs 4 and 5). For this example, the assumption of Type M characteristics will be more restrictive than Type S for the calculation of effective dose. Calculation shows that the other aerosol characteristics, such as particle density and shape factor, have only slight influence on the committed effective dose [40].

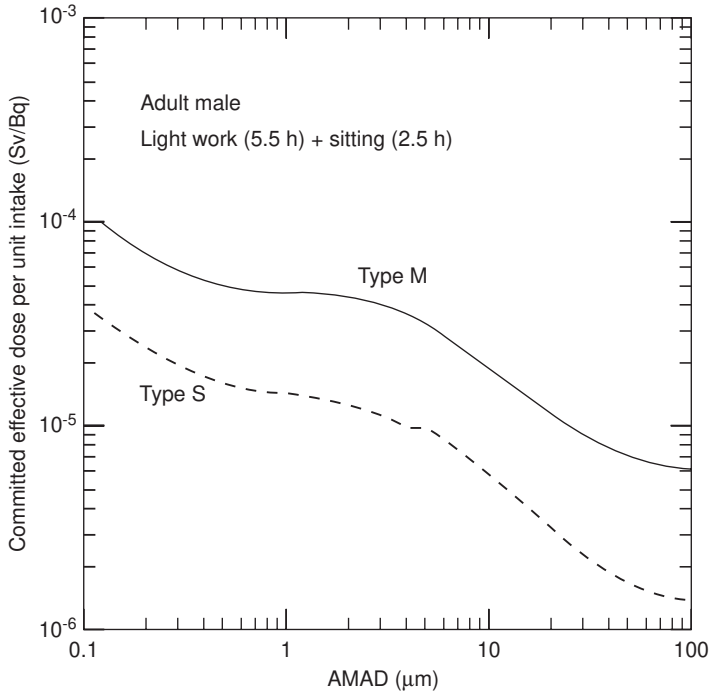


FIG. 5. Influence of AMAD on the committed effective dose from ^{239}Pu inhaled as type M or type S compounds [40].

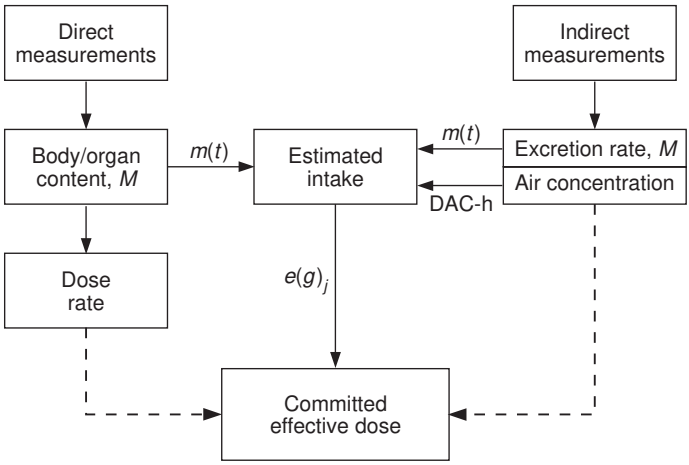


FIG. 6. General scheme for the interpretation of the results of monitoring measurements (possible alternative approaches for calculation are indicated as dashed lines).

7. INTERPRETATION OF MEASUREMENTS

INTRODUCTION

7.1. Direct or indirect measurements provide information about the amount(s) of radionuclides present in the body, in parts of the body such as specific organs or tissues, in a biological sample or in a sample from the working environment. The first use of these data is likely to be an estimation of the intake of the radionuclide by the worker. Biokinetic models which describe body and organ contents, and activity in excreta, as a function of time following intake, and exposure models which relate intake to workplace conditions, are used for this purpose. Alternatively, measurements of activity in the body can be used to estimate dose rates directly. The calculation of committed doses from direct measurements still involves the assumption of a biokinetic model if sufficient measurements are not available to determine retention functions.

7.2. The purpose of this section is to provide a general overview of the interpretation of measurements and to illustrate this process using an example of a simple dose assessment for an intake of ^{131}I, based on both direct and indirect monitoring results. Figure 6 summarizes the general approach.

7.3. To calculate an estimate of intake, the measured body content or excretion rate, M, is divided by the fraction $m(t)$ of the intake retained in the whole body (direct measurement) or having been excreted from the body (indirect measurement) at time t (usually in days) after intake:

$$\text{Intake} = \frac{M}{m(t)}$$

The ICRP has published generic values of $m(t)$ for selected radionuclides in tissues or excreta, together with retention functions for systemic activity [8]. Further information is provided in ICRP Publication 78 [9] using more recent biokinetic models.

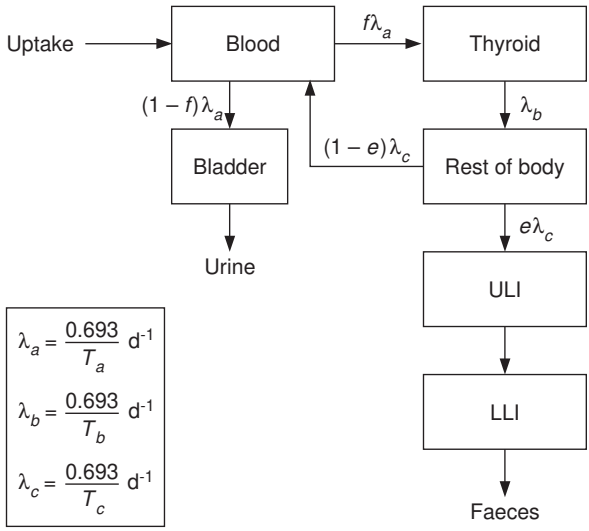
7.4. When significant intakes may have occurred, more refined calculations based on individual specific parameters (special dosimetry) should be made (Section 3). If multiple measurements are available, a single best estimate of intake may be obtained, for example, by the method of least squares [41, 42].

EXAMPLE OF DOSE ASSESSMENT FOR AN INTAKE OF ^{131}I

Sources of measurement data

7.5. Occupational exposure due to radioiodine occurs in the nuclear industry, in nuclear medicine and in research. One common exposure is due to ^{131}I, a short lived radioisotope (half-life 8 d) which decays with the emission of both beta particles (average energy for main emission 0.19 MeV) and gamma radiation (main emission 0.36 MeV) [43]. Iodine is rapidly absorbed into the circulation following inhalation or ingestion, is concentrated in the thyroid, and is excreted predominantly in urine [34, 36]. Thus, after an intake, ^{131}I may be detected directly by measurement of activity in the thyroid, or indirectly in urine samples.

7.6. Where occupational exposures due to ^{131}I can occur, a routine monitoring programme may be based on direct thyroid measurement or on indirect monitoring of urine or workplace samples. The choice of monitoring method will depend on factors such as the availability of instrumentation locally (since the isotope is short lived) and the relative costs of the analyses, as well as on the sensitivity that is needed (see Section 3). Although direct measurement of activity in the thyroid provides the basis for the most accurate dose assessment, other methods may provide adequate monitoring and may be better suited to particular circumstances.



Model parameters for iodine

Age	f_1	Uptake by thyroid, f	Faecal excretion, e	Biological half-time (d)		
				Thyroid T_a	Thyroid T_b	Rest of body T_c
Adult	1	0.3	0.2	0.25	80	12

FIG. 7. Biokinetic model for iodine in adults (based on [36]).

Biokinetic information

7.7. All common forms of iodine are readily taken up by the body. For inhalation of iodine in particulate form, lung absorption type F is assumed, while elemental iodine vapour is assigned to class SR-1 (soluble or reactive) with absorption type F. The absorption of iodine from the gastrointestinal tract is assumed to be complete, i.e. $f_1 = 1$. Dose coefficients for these forms of intake are given in Table A–I.

7.8. The most recent biokinetic model for systemic iodine recommended by the ICRP [36] (Fig. 7) is similar to that described in ICRP Publication 30 [29]. For adults, it is assumed that, of the iodine reaching the blood, 30% is transported to the thyroid

gland and the other 70% is excreted directly in urine via the urinary bladder. The biological half-life in blood is taken to be 6 h. Iodine incorporated into thyroid hormones leaves the gland with a biological half-life of 80 d and enters other tissues, where it is retained with a biological half-life of 12 d. Most iodine (80%) is subsequently released and is available in the circulation for uptake by the thyroid or direct urinary excretion; the remainder is excreted via the large intestine in the faeces. Because of the short physical half-life of ^{131}I, this recycling is not important in terms of the committed effective dose.

Direct measurements

7.9. Iodine-131 in the body is normally monitored directly by measuring activity in the thyroid using a simple NaI(Tl) detector [7]. Where a mixture of radioisotopes of iodine may be encountered, spectroscopic determination of the ^{131}I gamma emission may be necessary.

7.10. As an example, suppose that in a routine monitoring programme, with a monitoring period of 14 days, a thyroid content of 3000 Bq ^{131}I is detected in a male worker. Because of the operations under way in this workplace, it is assumed that any exposures will be due to inhalation of a particulate rather than vapour form (although for ^{131}I this assumption is not critical). Similarly, intakes by ingestion would also lead to the same pattern of retention and excretion [8, 9], and the same committed effective dose calculated from the monitoring data.

7.11. If the intake pattern is not known, and the monitoring period is consistent with the guidance given in Section 3, it should be assumed that an acute intake occurred in the middle of the monitoring period, provided that intakes are uncommon. With this assumption, it can be shown from the biokinetic model that 7.4% of the radioactive substance inhaled in a particulate (type F) form with a default AMAD of 5 μm is retained in the thyroid after 7 d [8]. Thus, $m(7) = 0.074$, and the example monitoring result from the previous paragraph would indicate an intake of 41 kBq. Application of the dose coefficients given in the BSS [2] and in Table A–I gives a committed effective dose of 450 μSv from such an intake. Such a dose may require follow-up investigation (see Section 3).

Indirect measurements

Urine

7.12. One day after the direct thyroid measurement, the worker in the example submits a 24-h urine sample, which is found to contain 30 Bq of ^{131}I. From the

biokinetic model for a type F particulate, $m(8)$ for daily urinary excretion is 1.1×10^{-4} [9]. On this basis, an intake of 270 kBq, and a committed effective dose of 3 mSv (for an aerosol with an AMAD of 5 μm), would be calculated. For this example no account is taken of any previous intakes.

Workplace air measurements

7.13. In the example, a review of workplace air measurements over the monitoring period, in the facilities where the exposure may have occurred, demonstrated that concentrations of ^{131}I were generally low but variable. Maximum concentrations between 10 and 20 kBq/m^3 (12 to 25 times the DAC value, see Table A–II) were recorded for short periods several times during the period, and in several locations. At the default breathing rate of 1.2 m^3/h, an intake of 24 kBq can be received while working for one hour without respiratory protection in a concentration of 20 kBq/m^3. Were the worker to have done so, or to have worked for a somewhat longer period with limited respiratory protection, the intake calculated from air monitoring would be consistent, within the accuracy normally achievable by such methods, with that calculated from bioassay measurements.

Dose assessment

7.14. The large discrepancy between the estimates of intake calculated on the basis of the direct thyroid measurement and of the measurement of radioactive material excreted in urine suggests that at least one of the default assumptions used to derive these estimates is not correct. Although there are significant individual differences in iodine uptake and metabolism, these differences cannot generally account for a discrepancy of a factor of nearly ten. On the other hand, the rate of excretion of ^{131}I in urine decreases markedly with time after intake, by a factor of more than 1000 over the monitoring period, so the default assumption concerning the time of intake is a probable source of error. If the intake were assumed to have occurred three days before the urine sample was submitted (i.e. two days before the end of the monitoring period), rather than at the midpoint of the monitoring period (eight days before the sample), the intake estimated from the urine measurement would be 21 kBq, and that from the thyroid measurement would be 25 kBq, a satisfactory agreement.

7.15. According to the biokinetic model, the fraction of inhaled ^{131}I retained in the thyroid only changes by about a factor of three over the whole monitoring period. In the absence of better evidence from a review of the sources of possible workplace exposure, this refined assumption provides a more reliable basis for dose assessment. The committed effective dose for this example would then be 270 μSv. A second urine sample obtained after a few more days should be used to verify this conclusion.

7.16. The committed effective dose calculated from direct thyroid monitoring results is relatively insensitive to assumptions about the time of intake. It is because of the rapid change in urinary excretion with time after exposure that direct measurement provides a much more reliable basis for interpreting routine monitoring measurements for radioiodine, although urine screening may still be adequate to detect significant intakes.

7.17. The measurement of air concentrations substantially exceeding a DAC would have triggered individual monitoring of workers who had been present in the workplace. However, because of their direct dependence on the period of exposure, breathing rates, levels of protection and other factors that will be known only approximately, estimates of intake based on air monitoring for ^{131}I are much less reliable than those based on individual measurements.

UNCERTAINTIES IN DOSE ASSESSMENTS

7.18. The models that have been developed by the ICRP for describing the behaviour of radionuclides in the body, and hence for assessing intakes, provide the most up-to-date methods available for dose assessment. There are, however, a number of uncertainties that should be considered when interpreting monitoring data.

7.19. Direct methods rely on the results of either whole or partial body monitoring. The accuracy of any measurements will depend principally on the level of activity, but also on the accuracy of calibration of the monitoring equipment. Limits of detection for any particular radionuclide can be calculated from a knowledge of the sensitivity of the equipment and the background count in the region of interest.

7.20. For indirect methods, the accuracy of measurements of levels of activity in physical or biological samples depends on similar considerations. It is, however, generally possible to define the counting geometry accurately, and counting times can be extended if necessary to obtain acceptable counting statistics for all samples except those with very low activity (or very short half-lives).

7.21. From an assessment of the activity in the whole body or in samples of tissues or excreta, the models used to describe the behaviour of radionuclides in the body are then used to assess intake and dose. The reliability of the estimates of dose therefore depend upon the accuracy of the models, and any limitations on their application in particular circumstances. This will depend upon many factors. In particular, knowledge of the time of the intake(s) and of whether the intake was acute or chronic is essential for a reliable dose estimate.

7.22. When the sampling period does not enable the biological half-life of the radionuclide to be estimated, assuming a long period of retention in the body for the purpose of dose assessment may result in an underestimate of the intake, and hence of the committed effective dose. The degree of over- or under-estimation of the dose will depend upon the overall pattern of retention in the body.

7.23. The behaviour of radionuclides that enter the body by ingestion or inhalation will depend upon their physicochemical characteristics. For inhaled radionuclides, the particle size is particularly important in influencing deposition in the respiratory system, while for ingestion the gut absorption factor f_1 can substantially influence effective dose. For routine monitoring when exposures are well within limits on intake, the default parameters recommended in the BSS [2] may be sufficient for assessing intakes. For exposures approaching or exceeding these limits, however, more specific information on the physical and chemical form of the intake, and the characteristics of the individual, may be needed to improve the accuracy of the model predictions.

DOSE COEFFICIENTS AND DERIVED AIR CONCENTRATIONS

7.24. The Annex gives dose coefficients $e(g)_j$ from the BSS [2] and DACs for selected radionuclides that are likely to be of concern in the workplace. The DACs are calculated on the basis of an effective dose limit of 20 mSv in a year, a working time of 2000 h per year and a standard breathing rate of 1.2 m^3/h, and are given for AMADs of 1 and 5 μm.

8. DOSE RECORD KEEPING AND REPORTING

GENERAL

8.1. Dose record keeping is the making and keeping of individual dose records for radiation workers. It is an essential part of the process of monitoring the exposures of individuals to radiation and supports the overall objectives (Section 3). General guidance on record keeping and reporting is given in a related Safety Guide [3]. Further information that relates specifically to doses from incorporated radionuclides is given below.

8.2. Records should provide support for decision making, demonstrate and facilitate regulatory compliance, provide for the reconstruction of results at any later time, and

facilitate co-ordination with other required records such as those for external monitoring and area monitoring. They should therefore be easily retrievable and be protected against loss. Such protection is usually obtained by maintaining duplicate sets of records in well separated locations, so that both copies cannot be destroyed in a single incident. Records should be consolidated for each monitored individual, identified by site, purpose, date and originator, and should be legible and intelligible to a qualified person, complete and accurate. Consideration may need to be given to any applicable national requirements or international agreements concerning the privacy of individual data records.

RECORD KEEPING FOR INDIVIDUAL MONITORING

8.3. The purpose of record keeping, the nature and scope of the records and the extent of record keeping systems depend on national requirements. The records should include the results of individual monitoring for both external radiation and intakes of radioactive material.

8.4. Typical records generated in an internal exposure monitoring programme include both directly relevant data and supporting documentation. The records should ensure the traceability of the measurements and the dose assessment. Directly relevant information includes sample data, such as the date and time of collection and evidence of a 'chain of custody', raw data from measurement devices, such as counting rates in specific energy bands, measurements of backgrounds and standards and calibration data for the counters, calculated results such as activity content of the body or daily excretion rates and their statistical analyses, calculated estimates of intake and the biokinetic models from which they were derived, and estimated committed effective doses and the dose conversion factors used. Supporting documentation includes working procedures and practices, training records, quality assurance (QA) procedures, quality control data such as background trends, estimates of minimal detectable activity, results of sample analyses, equipment calibration procedures and records, and the traceability of standard sources.

RECORD KEEPING FOR WORKPLACE MONITORING

8.5. The requirements for maintaining area monitoring records, such as air sampling and surface contamination survey records, are similar to those for individual monitoring records. Although such records may also be maintained for operational radiation protection purposes, this discussion is limited to their use for internal exposure assessment. If an internal exposure assessment has been based on an air sample, then

all data relating to that sample and the equipment with which it was collected should be maintained just as the data from direct measurements or excreta analyses are maintained. Even if area monitoring data are not used for internal exposure assessment, they should be maintained for future verification of workplace conditions.

8.6. Records documenting the designation and location of controlled and supervised areas should be kept. Records should also be kept of radiation surveys, including the date, time, location, and the radiation levels measured, and any comments relevant to the measurements made. Records should identify the instrument(s) used and the individual performing the survey.

REPORTING OF INFORMATION TO THE MANAGEMENT

8.7. The procedures and criteria to be used for reporting individual and workplace monitoring results should be clearly specified by the management or regulatory authority. Information reported should be clearly identifiable and understandable. Normally only final results are reported.

8.8. In accident situations, or for a potential intake that may be close to or above a regulatory limit, interim results should be supplied so that appropriate administrative and other response actions can be instituted. The results should include the result of the measurement, the implied intake value, based on the appropriate biokinetic model, and the implied committed effective dose based upon the corresponding dose coefficient $e(g)_j$. Recommendations for follow-up monitoring and for workplace restrictions may be made if appropriate. The source of the information reported should be clearly identified, as should a point of contact for any additional information. Finally, the uncertainty in the measured and computed values should always be reported, accompanied by a statement of which sources of variability have been considered, quantified and propagated in the quoted uncertainty.

9. QUALITY ASSURANCE

INTRODUCTION

9.1. The continued effectiveness of any radiation protection programme relies on those in charge implementing its various components, including the adoption of an effective QA programme. General QA requirements related to occupational exposure are given in the BSS [2] and general guidance is given in the related Safety Guide [3].

The following section deals specifically with issues related to the assessment of exposure due to intakes of radionuclides.

IMPLEMENTATION AND MANAGEMENT

9.2. The nature and extent of the QA programme should be consistent with the number of workers monitored, and the magnitude and likelihood of exposures expected in the workplaces to be covered by the monitoring programme.

9.3. All persons involved in the internal exposure assessment programme are responsible for its quality and therefore for implementing its QA programme and quality control (QC) procedures. Responsibility for the quality of a particular operation should be delegated to the person actually performing the operation. Such persons should be actively involved in the development of QC procedures, and trained in methods of detecting non-compliance. Management should motivate staff to detect, report and correct non-compliance. Quality assurance built into a programme from the bottom up is more effective than QA imposed from the top down. For the QA programme to be effective, all personnel must be confident that management expects and encourages performance that meets its objectives.

9.4. An analytical laboratory or direct measurement facility should have a designated QA representative. This representative should monitor QC procedures, perform internal audits of the programme, and be responsible for training all personnel in QA, both in general terms and in the specific quality aspects of their individual work.

9.5. Specific QC measures for direct methods of assessing internal exposures are provided in Ref. [7]. National regulations may require that facilities concerned with measurement and internal dose assessment be accredited. Such accreditation programmes will have specifications for QA and QC to be implemented.

Documentation

9.6. The QA programme related to internal exposure assessment should be thoroughly documented. A QA plan should be prepared that contains general instructions on implementing the programme and the various steps in its operation. Written procedures should describe every task and specify QC criteria. For example, a radiochemistry analytical procedure should contain acceptable limits for chemical yield. Quality control procedures should document the use of control charts and other methods for tracking instrument backgrounds, efficiencies and other performance measures, and should contain instructions for reporting and correcting deviations, as well as for taking account of changes in the operation. Procedures for documenting

and reporting results should also be prepared, as should procedures for record preparation, maintenance and archiving. The documentation should provide sufficient information for an auditor to trace the operation from start to finish and assess its validity. Once the written procedures have been approved, any departures from them or modifications should be authorized and documented.

Training of personnel

9.7. Adequate training of dosimetry service personnel is essential to ensure that they can perform their jobs reliably. Such training should include:

(a) Their particular responsibility within the quality system;
(b) The basic philosophy and strategy of internal dose assessment;
(c) The principles and details of the methods and procedures used, and their limitations;
(d) The technical details and potential problems of the processes in which they are involved;
(e) The relation their work has to other parts of the programme;
(f) Guidance on recognition and reporting of problems that arise;
(g) Knowledge of the overall quality system and its objectives.

Laboratory facilities

9.8. It is difficult to achieve high quality results in a substandard environment. Adequate laboratory and office space should be available to accommodate the necessary equipment and personnel. Equipment should be reliable and stable, and appropriate to the task for which it is intended, and procedures should be in place to prevent contamination of measurement equipment with radionuclides. A preventative maintenance programme should be instituted to minimize the chance that equipment will fail at a critical time, such as in an emergency. Activities that are not directly related to the performance of dosimetry service operations should be separated to avoid unnecessary interference. The general safety of working conditions should also be considered.

9.9. Ventilation, fume hoods and bench space are necessary for radiochemical operations. Shielded facilities should be provided for detectors, including those in direct assessment facilities. Access control to all facilities is necessary, both to protect sensitive equipment and to maintain appropriate confidentiality of records. The facility should have an appropriate continuous floor covering (e.g. vinyl) to facilitate cleaning and decontamination.

9.10. The workplace controls should be adequate to ensure that no equipment is subject to conditions likely to affect its performance. Factors that should be controlled include temperature, humidity, light levels, dust and reactive chemical vapours.

9.11. A stable power supply is needed so that the voltage and AC frequency remain within the specifications of the equipment in use. Stray electric and magnetic fields should be minimized to avoid affecting equipment.

9.12. Changing rooms and showers should be provided at facilities used for direct measurements.

PERFORMANCE ASSESSMENT

9.13. A system should be established to provide a quality indicator of the overall internal dosimetry service performance. One such system is a routine programme of equipment and procedure testing. All test results should be documented, together with any resulting modifications to procedures.

9.14. As part of its QA programme, a measurement facility may establish performance criteria to be applied to the analysis of spiked samples, i.e. samples for which the radionuclide content is known beforehand. Performance criteria should be well defined limits on the acceptability of the measurement results, as functions of the radionuclide content in the sample versus the MDA of the method. For example, an acceptable result for the analysis of ^{239}Pu in faeces may be a measured value that lies within the range of 0.75 to 1.5 times the true value, when the true value is at least five times the MDA of the method. Similarly, an acceptable result for the direct measurement of ^{241}Am in the lungs of a phantom could be a measured value within a range of 0.75 to 1.5 times the true value, when the true value is ten times the MDA of the method [44].

9.15. Performance criteria should also be set for the precision of replicate measurements, e.g. a variation of not more than 30% in successive measurements of the same sample, if the true content is five times the MDA of the method. If the activity is low so that random statistical errors are dominant, performance criteria cannot be more restrictive than statistical fluctuations permit.

9.16. Analyses of samples intended for performance assessment should be performed in at least a single blind fashion, i.e. the analyst should not know the true value beforehand, although the assessment sample may be identified as being intended for

performance assessment. In a double blind assessment, the analyst is not informed that the sample is anything other than routine. Although a double blind assessment may give a truer picture of the laboratory's capabilities, such assessments are logistically difficult to perform.

9.17. Laboratories performing direct or indirect measurements for internal dosimetry should participate in national and international intercomparison exercises. National intercomparison programmes for direct measurements are available in many countries and international intercomparisons are also being co-ordinated [45–49]. Likewise, both national and international intercomparisons for indirect measurements have been co-ordinated, for example, by the French Atomic Energy Commission (CEA) [50]. In addition, periodic audits or reviews should be performed to verify compliance with the QA programme and the effectiveness of the internal dosimetry programme. Guidance for the conduct of audits and reviews is given in the related Safety Guide [3].

CONTRACTING FOR A MONITORING SERVICE

9.18. It may be necessary for many operators (registrants and licensees) to obtain dosimetry services under contract from commercial suppliers. This is especially true for operators with small workforces, such as university laboratories and small hospitals, who may have limited knowledge and/or experience in radiation protection and internal dosimetry. However, in contracting for commercial dosimetry services, they should ensure that there is adequate communication and understanding with the suppliers to ensure an effective dosimetry programme. The following items should be considered:

(a) Regulatory requirements;
(b) Types of direct and indirect methods of dose assessment, and their limitations;
(c) Quality records, references or certificates for equipment and services;
(d) Selection of monitoring periods;
(e) Methods for collecting bioassay samples or preparing for body monitoring;
(f) Dose record keeping, reporting of results, customer dose entries, accessibility and confidentiality;
(g) Interpretation of results (quantities, dose limits, natural background, net dose, lower and upper limit of detection of the dosimetry system, etc.);
(h) Procedures for ordering, changing and cancelling monitoring services;
(i) Information needed from the operator;
(j) Costs;

(k) The amount of time to be allowed to make an order (or cancellation) effective;
(l) Information on routine and/or special services provided by the commercial service, such as immediate reporting by telephone or telex in the event of unusually high doses, emergency processing and advice on technical, scientific and legal matters.

Appendix I

SUGGESTED CRITERIA FOR INDIVIDUAL MONITORING

I.1. The BSS [2, para. I.33] require that: "For any worker who is normally employed in a controlled area, or who occasionally works in a controlled area and may receive significant occupational exposure, individual monitoring shall be undertaken where appropriate, adequate and feasible."

I.2. Many factors should be considered in determining the appropriateness of individual monitoring. These include, but are not limited to, the quantity of radioactive material present, the radiotoxicity of the material, the nature of the operations being performed and the containment employed. The advice of a qualified expert may be used to assist in decisions on monitoring.

I.3. In assessing whether individual monitoring is required, based on the potential for committed effective doses of 1 mSv or greater in a year, a number of factors need to be taken into account, including the following:

(i) The physical form safety factor f_{fs}, based on the physical and chemical properties of the material being handled. In the majority of cases, f_{fs} should be given a value of 0.01. However, in some cases, where it can be shown to be justified, a value of 0.001 may be used;
(ii) The handling safety factor f_{hs}, based on experience of the operation being performed and the form of the material; and
(iii) The protection safety factor f_{ps}, based on the use of permanent laboratory protective equipment (e.g. glove box, fume hood).

I.4. Although personal protective measures (e.g. the use of face masks) provide an additional element of safety, they should not be taken into account in decisions about the need for individual monitoring. If personal protective measures are employed, individual monitoring should be conducted to confirm their effectiveness.

I.5. Suggested values of f_{hs} and f_{ps} for general application are given in Tables IV and V respectively [51], but due regard should be given to the circumstances affecting individual cases. The form of the material being used (e.g. volatile liquid, powder) may sometimes be taken into account both directly (i.e. through f_{fs}) and indirectly, through the relative efficiency of the protective measures being taken (i.e. through f_{hs} and/or f_{ps}). The following illustrates how the above factors may be applied in determining whether monitoring is required.

TABLE IV. HANDLING SAFETY FACTORS

Process	Handling safety factor f_{hs}
Storage (stock solution)	0.01
Very simple wet operations	0.1
Normal chemical operations	1
Complex wet operations with risk of spills	10
Simple dry operations	10
Handling of volatile compounds	100
Dry and dusty operations	100

TABLE V. PROTECTION SAFETY FACTORS

Protection measure	Protection safety factor f_{ps}
Open bench operations	1
Fume hood	0.1
Glove box	0.01

I.6. A specific radionuclide 'decision factor' d_j for a specific practice can be defined as:

$$d_j = \frac{A_j e(g)_{j,inh} f_{fs} f_{hs} f_{ps}}{0.001}$$

where A_j is the cumulative activity of radionuclide j present in the workplace over the course of the year, $e(g)_{j,inh}$ is the dose coefficient (Sv/Bq) for inhalation of radionuclide j (from BSS Table II–III [2], with the AMAD normally taken to be 5 μm for the workplace), and the 0.001 is a conversion factor from Sv to mSv. If f_{fs} has the default value of 0.01, the above equation may be simplified to:

$$d_j = 10\, A_j\, e(g)_{j,inh} f_{hs} f_{ps}$$

The decision factor for all radionuclides in the workplace D is given by:

$$D = \sum_j d_j$$

Then, if D is 1 or more, a need for individual monitoring would be indicated, and if D is less than 1, individual monitoring may not be necessary.

I.7. For a single radionuclide being handled on the open bench ($f_{hs} = 1$) with normal chemical operations ($f_{ps} = 1$), with the default value of f_{fs} of 0.01, the activity needed to give a value of $d_j = 1$ would be 5 times the relevant limit on intake (Section 2).

I.8. Where more than one radionuclide is present in the workplace, decisions to conduct individual monitoring for the separate radionuclides may be based on the following criteria:

(i) All radionuclides for which $d_j \geq 1$ shall be monitored;
(ii) When $D \geq 1$, radionuclides for which $d_j \geq 0.3$ should be monitored;
(iii) Monitoring of radionuclides for which d_j is much less than 0.1 is unnecessary.

I.9. Consider as an example a workplace in which ^{239}Pu as an insoluble oxide is handled during normal chemical operations in a fume hood. The default AMAD for workplaces of 5 μm is assumed. The values of f_{fs}, f_{hs} and f_{ps} are taken to be 0.01, 1.0 and 0.1, respectively. The above equation then becomes:

$$d_{\mathrm{Pu239}} = 10\, A_{\mathrm{Pu239}} \times 8.3 \times 10^{-6} \times 1 \times 0.1$$

$$= \; 8.3 \times 10^{-6}\, A_{\mathrm{Pu239}}$$

From this, individual monitoring would be required if A_{Pu239}, the activity of ^{239}Pu, is greater than:

$$\frac{1}{8.3 \times 10^{-6}} = 1.2 \times 10^{5}\ \mathrm{Bq}$$

Otherwise, individual monitoring would not be required.

I.10. If ^{137}Cs is also being handled in the same workplace, the decision factor for ^{239}Pu (d_{Pu239}) would be as before, and the decision factor for ^{137}Cs would be given by:

$$d_{\mathrm{Cs137}} = 6.7 \times 10^{-9}\, A_{\mathrm{Cs137}}$$

where A_{Cs137} is the activity of ^{137}Cs present in the workplace. If:

$$D = 8.3 \times 10^{-6} A_{Pu239} + 6.7 \times 10^{-9} A_{Cs137} \geq 1$$

then individual monitoring should be performed for any nuclide for which $d_j \geq 0.3$, i.e. for ^{239}Pu if A_{Pu239} is greater than 36 kBq, and for ^{137}Cs if A_{Cs137} is greater than 45 MBq. If $D \geq 1$, individual monitoring would be unnecessary for ^{239}Pu if A_{Pu239} is much less than 12 kBq and for ^{137}Cs if A_{Cs137} is much less than 15 MBq.

I.11. For operations that meet the above criteria, individual workers should be monitored, either as part of a routine monitoring programme if the work is performed continuously, or as part of a task related monitoring programme if that operation is performed only occasionally. In addition, if an incident occurs which may cause an intake of radioactive material (e.g. the failure of a fume hood or the breakage of a container), special monitoring of the persons involved in the incident should take place, whether or not they are part of the routine monitoring programme.

I.12. In addition, some workers may be assigned multiple duties in different workplaces. The decision whether such a worker requires individual monitoring should be based on a careful review of all of the worker's duties.

I.13. More detailed guidelines on workplace categorization and monitoring requirements are given in Ref. [52].

Appendix II

DETECTION LIMITS FOR MEASUREMENT METHODS

II.1. All measurement methods have limits of detection. These arise from a number of factors, such as natural radioactivity, statistical fluctuations in counting rates and factors related to sample preparation and analysis. Consequently, the detection of intakes is also limited. The dose resulting from an intake of less than the detection limit of the measurement method will be missed.

II.2. In the reporting and interpretation of analytical measurements, a sound understanding of the fundamental concepts of statistical detectability is critical. This understanding is needed to make decisions concerning the acceptability of different methods of direct and indirect measurement for the assessment of intakes of radionuclides. A summary of the concepts is given below; greater detail is given in reviews by Currie [53] and by Altshuler and Pasternack [54]. A comprehensive analysis of the statistical methods appropriate for application to direct and indirect monitoring is given in Health Physics Society Standard N13.30 [44].

II.3. The minimum significant activity (MSA), often termed the decision limit or critical level (L_C), corresponds to the smallest signal significantly in excess of the background response for the specific measurement method. It corresponds to the level of a randomly fluctuating background response which, in the absence of a radioactive sample, will be exceeded only with some low probability α. Conventionally, α is taken to be 0.05, so that a net signal corresponding to deposited activity at the MSA level may be taken to indicate the presence of radioactive substance with 95% probability. If, as will usually be the case, random fluctuations in the net counts follow a normal distribution, the MSA will correspond to 1.65σ, where σ is the standard deviation of the distribution. Reports of assessments which lie below this level may legitimately state that radioactive material was ‘not detected’ or that any activity was ‘below the limit of detection’. Radioactive material present at the MSA level will not necessarily be detected: in such a situation, there would be a 50% probability that a net recorded signal corresponding to a deposit at or below the MSA would be obtained.

II.4. The minimum detectable activity (MDA), often termed the detection limit (L_D), corresponds to the level of activity which is needed to ensure, with some chosen level of confidence β, that the net signal will be detected, according to the criterion that it exceed the MSA. The mathematical treatment is simplified, as in the following section, if $\beta = \alpha$, and by common convention 0.05 is adopted for both.

EVALUATION OF MSA AND MDA

II.5. The following formulae may be derived as in Refs [53, 54] or by analogous treatments, and relate to simple comparisons of sample and background counting rates. In the following, only those variations associated with counting statistics are considered. If n_b is the background count rate, t_s and t_b are, respectively, the count times for the sample and for an associated measurement of the background, F is a calibration factor (the count rate per unit activity in the sample) and 95% confidence intervals are assumed to apply, i.e. $\alpha = \beta = 0.05$, then:

$$\mathrm{MSA} = \frac{1.56}{F}\sqrt{\frac{n_b}{t_s}\left(1+\frac{t_s}{t_b}\right)}$$

For sample and background measurements with equal counting times, i.e. $t_s = t_b$, this simplifies to:

$$\mathrm{MSA} = \frac{2.33\sigma_b}{F}$$

where σ_b is the standard deviation of the background count rate, given by:

$$\sigma_b = \sqrt{\frac{n_b}{t_b}}$$

The measurement MDA is given by:

$$\mathrm{MSA} = \frac{3}{Ft_s} + 2\mathrm{MSA}$$

The first term in this expression represents a correction for the non-normality of Poisson counting statistics at low total counts [44] and can be neglected if, as generally applies:

$$n_b t_s >> \frac{0.7}{1+\frac{t_s}{t_b}}$$

II.6. These equations for MSA and MDA take into account only variation in the detection process. They may be related to the overall MSA and MDA for the method by multiplying by appropriate conversion factors, for example, for fractional

radiochemical yield and counting efficiency. However, the resultant values will not take into account variation in these additional factors, and may therefore underestimate the method MDA. Health Physics Standard N13.30 provides a thorough description of the analysis that can be used when this additional variation is important [44].

II.7. However, in many situations the variability in background and sample counts is greater than that to be expected from Poisson variation in counting statistics alone. This extra variation may be due, for example, to the presence of variable natural levels of activity in the sample. In such circumstances, the distribution of measured background (and sample) counts may be approximately normal, in which case a straightforward correction can be derived [55] using the measured standard deviation σ_m. In other cases the distribution may not be normal, for example it may be lognormal, in which case an appropriate — but not simple — correction may still be derived. Such an example could be the presence of ^{137}Cs in the population following atmospheric tests of nuclear weapons or the Chernobyl accident, from which internal contamination from occupational sources has to be distinguished.

II.8. It should be emphasized that when determinations of detection limits or calculations of significance do not take into account all sources of variability, the detection capability of the method will be overestimated.

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Annex

BASIC DATA

TABLE A–I. DOSE COEFFICIENTS FOR SELECTED RADIONUCLIDES (from [A–1])

Radionuclide	Inhalation			Ingestion	
	Type/form[a]	$e(g)_{inh}$ (Sv/Bq)		f_1	$e(g)_{ing}$ (Sv/Bq)
		AMAD = 1 μm	AMAD = 5 μm		
H-3	HTO[c]	1.8×10^{-11}[b]		1	1.8×10^{-11}
	OBT	4.1×10^{-11}[b]		1	4.2×10^{-11}
	Gas	1.8×10^{-15}[b]			
C-14	Vapour	5.8×10^{-10}[b]		1	5.8×10^{-10}
	CO_2	6.2×10^{-12}[b]			
	CO	8.0×10^{-13}[b]			
P-32	F	8.0×10^{-10}	1.1×10^{-9}	0.8	2.3×10^{-10}
	M	3.2×10^{-9}	2.9×10^{-9}		
Fe-55	F	7.7×10^{-10}	9.2×10^{-10}	0.1	3.3×10^{-10}
	M	3.7×10^{-10}	3.3×10^{-10}		
Fe-59	F	2.2×10^{-9}	3.0×10^{-9}	0.1	1.8×10^{-9}
	M	3.5×10^{-9}	3.2×10^{-9}		
Co-60	M	9.6×10^{-9}	7.1×10^{-9}	0.1	3.4×10^{-9}
	S	2.9×10^{-8}	1.7×10^{-8}	0.05	2.5×10^{-9}
Sr-85	F	3.9×10^{-10}	5.6×10^{-10}	0.3	5.6×10^{-10}
	S	7.7×10^{-10}	6.4×10^{-10}	0.01	3.3×10^{-10}
Sr-89	F	1.0×10^{-9}	1.4×10^{-9}	0.3	2.6×10^{-9}
	S	7.5×10^{-9}	5.6×10^{-9}	0.01	2.3×10^{-9}
Sr-90	F	2.4×10^{-8}	3.0×10^{-8}	0.3	2.8×10^{-8}
	S	1.5×10^{-7}	7.7×10^{-8}	0.01	2.7×10^{-9}
Zr-95	F	2.5×10^{-9}	3.0×10^{-9}	0.002	8.8×10^{-10}
	M	4.5×10^{-9}	3.6×10^{-9}		
	S	5.5×10^{-9}	4.2×10^{-9}		
Nb-95	M	1.4×10^{-9}	1.3×10^{-9}	0.01	5.8×10^{-10}
	S	1.6×10^{-9}	1.3×10^{-9}		
Ru-106	F	8.0×10^{-9}	9.8×10^{-9}	0.05	7.0×10^{-9}
	M	2.6×10^{-8}	1.7×10^{-8}		
	S	6.2×10^{-8}	3.5×10^{-8}		

TABLE A–I. (cont.)

Radionuclide	Inhalation			Ingestion	
	Type/form[a]	$e(g)_{inh}$ (Sv/Bq)		f_1	$e(g)_{ing}$ (Sv/Bq)
		AMAD = 1 µm	AMAD = 5 µm		
Sb-125	F	1.4×10^{-9}	1.7×10^{-9}	0.1	1.1×10^{-9}
	M	4.5×10^{-9}	3.3×10^{-9}		
I-125	F	5.3×10^{-9}	7.3×10^{-9}	1.0	1.5×10^{-8}
	V	1.4×10^{-8}[b]			
I-131	F	7.6×10^{-9}	1.1×10^{-8}	1.0	2.2×10^{-8}
	V	2.0×10^{-8}[b]			
Cs-134	F	6.8×10^{-9}	9.6×10^{-9}	1.0	1.9×10^{-8}
Cs-137	F	4.8×10^{-9}	6.7×10^{-9}	1.0	1.3×10^{-8}
Ce-144	M	3.4×10^{-8}	2.3×10^{-8}	5×10^{-4}	5.2×10^{-9}
	S	4.9×10^{-8}	2.9×10^{-8}		
Po-210	F	6.0×10^{-7}	7.1×10^{-7}	0.1	2.4×10^{-7}
	M	3.0×10^{-6}	2.2×10^{-6}		
Pb-210	F	8.9×10^{-7}	1.1×10^{-6}	0.2	6.8×10^{-7}
Ra-226	M	3.2×10^{-6}	2.2×10^{-6}	0.2	2.8×10^{-7}
Ra-228	M	2.6×10^{-6}	1.7×10^{-6}	0.2	6.7×10^{-7}
Th-228	M	3.1×10^{-5}	2.3×10^{-5}	5×10^{-4}	7.0×10^{-8}
	S	3.9×10^{-5}	3.2×10^{-5}	2×10^{-4}	3.5×10^{-8}
Th-232	M	4.2×10^{-5}	2.9×10^{-5}	5×10^{-4}	2.2×10^{-7}
	S	2.3×10^{-5}	1.2×10^{-5}	2×10^{-4}	9.2×10^{-8}
U-234	F	5.5×10^{-7}	6.4×10^{-7}	0.02	4.9×10^{-8}
	M	3.1×10^{-6}	2.1×10^{-6}	0.002	8.3×10^{-9}
	S	8.5×10^{-6}	6.8×10^{-6}		
U-235	F	5.1×10^{-7}	6.0×10^{-7}	0.02	4.6×10^{-8}
	M	2.8×10^{-6}	1.8×10^{-6}	0.002	8.3×10^{-9}
	S	7.7×10^{-6}	6.1×10^{-6}		
U-238	F	4.9×10^{-7}	5.8×10^{-7}	0.02	4.4×10^{-8}
	M	2.6×10^{-6}	1.6×10^{-6}	0.002	7.6×10^{-9}
	S	7.3×10^{-6}	5.7×10^{-6}		
Np-237	M	2.1×10^{-5}	1.5×10^{-5}	5×10^{-4}	1.1×10^{-7}
Np-239	M	9.0×10^{-10}	1.1×10^{-9}	5×10^{-4}	8.0×10^{-10}
Pu-238	M	4.3×10^{-5}	3.0×10^{-5}	5×10^{-4}	2.3×10^{-7}
	S	1.5×10^{-5}	1.1×10^{-5}	1×10^{-5}	8.8×10^{-9}
				1×10^{-4}	4.9×10^{-8}

TABLE A–I. (cont.)

Radionuclide	Type/form[a]	Inhalation		Ingestion	
		$e(g)_{inh}$ (Sv/Bq)		f_1	$e(g)_{ing}$ (Sv/Bq)
		AMAD = 1 µm	AMAD = 5 µm		
Pu-239	M	4.7×10^{-5}	3.2×10^{-5}	5×10^{-4}	2.5×10^{-7}
	S	1.5×10^{-5}	8.3×10^{-6}	1×10^{-5}	9.0×10^{-9}
				1×10^{-4}	5.3×10^{-8}
Pu-240	M	4.7×10^{-5}	3.2×10^{-5}	5×10^{-4}	2.5×10^{-7}
	S	1.5×10^{-5}	8.3×10^{-6}	1×10^{-5}	9.0×10^{-9}
				1×10^{-4}	5.3×10^{-8}
Pu-241	M	8.5×10^{-7}	5.8×10^{-7}	5×10^{-4}	4.7×10^{-9}
	S	1.6×10^{-7}	8.4×10^{-8}	1×10^{-5}	1.1×10^{-10}
				1×10^{-4}	9.6×10^{-10}
Am-241	M	3.9×10^{-5}	2.7×10^{-5}	5×10^{-4}	2.0×10^{-7}
Cm-242	M	4.8×10^{-6}	3.7×10^{-6}	5×10^{-4}	1.2×10^{-8}
Cm-244	M	2.5×10^{-5}	1.7×10^{-5}	5×10^{-4}	1.2×10^{-7}

[a] For lung absorption types see para. 6.13.
[b] For inhalation of gases and vapours, the AMAD does not apply for this form.
[c] HTO — tritiated water; OBT — organically bound tritium.

TABLE A–II. DERIVED AIR CONCENTRATIONS (DACs) FOR SELECTED RADIONUCLIDES

Radionuclide	Type/form[a]	DAC (Bq/m^3)		
		AMAD = 1 μm	AMAD = 5 μm	Gas/vapour
H-3	HTO[b]			5×10^5
	OBT			2×10^5
	Gas			5×10^9
C-14	Vapour			1×10^4
	CO_2			1×10^6
	CO			1×10^7
P-32	F	1×10^4	8×10^3	
	M	3×10^3	3×10^3	
Fe-55	F	1×10^4	9×10^3	
	M	2×10^4	3×10^4	
Fe-59	F	4×10^3	3×10^3	
	M	2×10^3	3×10^3	
Co-60	M	9×10^2	1×10^3	
	S	3×10^2	5×10^2	
Sr-85	F	2×10^4	1×10^4	
	S	1×10^4	1×10^4	
Sr-89	F	8×10^3	6×10^3	
	S	1×10^3	1×10^3	
Sr-90	F	3×10^2	3×10^2	
	S	6×10^1	1×10^2	
Zr-95	F	3×10^3	3×10^3	
	M	2×10^3	2×10^3	
	S	2×10^3	2×10^3	
Nb-95	M	6×10^3	6×10^3	
	S	5×10^3	6×10^3	
Ru-106	F	1×10^3	9×10^2	
	M	3×10^2	5×10^2	
	S	1×10^2	2×10^2	
Sb-125	F	6×10^3	5×10^3	
	M	2×10^3	3×10^3	
I-125	F	2×10^3	1×10^3	
	V			6×10^2

TABLE A–II. (cont.)

Radionuclide	Type/form[a]	DAC (Bq/m^3)		
		AMAD = 1 μm	AMAD = 5 μm	Gas/vapour
I-131	F	1×10^3	8×10^2	
	V			4×10^2
Cs-134	F	1×10^3	9×10^2	
Cs-137	F	2×10^3	1×10^3	
Ce-144	M	2×10^2	4×10^2	
	S	2×10^2	3×10^2	
Po-210	F	1×10^1	1×10^1	
	M	3×10^0	4×10^0	
Pb-210	F	9×10^0	8×10^0	
Ra-226	M	3×10^0	4×10^0	
Ra-228	M	3×10^0	5×10^0	
Th-228	M	3×10^{-1}	4×10^{-1}	
	S	2×10^{-1}	3×10^{-1}	
Th-232	M	2×10^{-1}	3×10^{-1}	
	S	4×10^{-1}	7×10^{-1}	
U-234	F	2×10^1	1×10^1	
	M	3×10^0	4×10^0	
	S	1×10^0	1×10^0	
U-235	F	2×10^1	1×10^1	
	M	3×10^0	5×10^0	
	S	1×10^0	1×10^0	
U-238	F	2×10^1	1×10^1	
	M	3×10^0	5×10^0	
	S	1×10^0	1×10^0	
Np-237	M	4×10^{-1}	6×10^{-1}	
Np-239	M	9×10^3	8×10^3	
Pu-238	M	2×10^{-1}	3×10^{-1}	
	S	6×10^{-1}	8×10^{-1}	
Pu-239	M	2×10^{-1}	3×10^{-1}	
	S	6×10^{-1}	1×10^0	
Pu-240	M	2×10^{-1}	3×10^{-1}	
	S	6×10^{-1}	1×10^0	

TABLE A–II. (cont.)

Radionuclide	Type/form[a]	DAC (Bq/m^3)		
		AMAD = 1 μm	AMAD = 5 μm	Gas/vapour
Pu-241	M	1×10^{1}	1×10^{1}	
	S	5×10^{1}	1×10^{2}	
Am-241	M	2×10^{-1}	3×10^{-1}	
Cm-242	M	2×10^{0}	1×10^{0}	
Cm-244	M	3×10^{-1}	5×10^{-1}	

[a] For lung absorption types see para. 6.13.

[b] The DAC does not allow for absorption through the intact skin.

TABLE A–III. TISSUE WEIGHTING FACTORS [FROM A–1]

Tissue or organ	Tissue weighting factor[a,b] (w_T)
Gonads	0.20
Bone marrow (red)	0.12
Colon[c]	0.12
Lung[d]	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Oesophagus	0.05
Thyroid	0.05
Skin	0.01
Bone surface	0.01
Remainder[e]	0.05

[a] Values of w_T originally from ICRP Publication 60 [A–2].

[b] The values have been developed for a reference population of equal numbers of both sexes and a wide range of ages. In the definition of effective dose they apply to workers, to the whole population and to either sex [A–2].

[c] Doses calculated as mass-weighted average to upper and lower large intestine: $H_{\text{colon}} = 0.57\ H_{\text{ULI}} + 0.43\ H_{\text{LLI}}$ [A–3].

[d] Thoracic region of the respiratory tract.

[e] For the purposes of calculation, the remainder is composed of adrenal glands, brain, extrathoracic region of the respiratory tract, small intestine, kidney, muscle, pancreas, spleen, thymus and uterus. In those cases in which the most exposed remainder tissue receives the highest committed equivalent dose of all organs, a weighting factor of 0.025 shall be applied to that tissue or organ and a weighting factor of 0.025 to the mass-weighted average dose in the rest of the remainder as defined here [A–4].

REFERENCES TO ANNEX

[A–1] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANISATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).

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[A–3] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Age-dependent Doses to Members of the Public from Intake of Radionuclides: Part 2, Ingestion Dose Coefficients, Publication No. 67, Elsevier Science, Oxford and New York (1993).

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DEFINITIONS

The following definitions apply for the purposes of the present publication. Unless otherwise stated, terms listed in the glossary of the BSS have the definitions given in that glossary.

acute intake. A single intake that is assumed to occur instantaneously.

bioassay. Any procedure used to determine the kind, quantity, location and/or retention of radionuclides in the body by direct (in vivo) measurements or by in vitro analysis of material excreted or removed from the body.

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

biological half-life. The time taken for a biological system, such as a tissue compartment or the whole body, to eliminate, by natural processes other than by radioactive decay, 50% of the amount of a radionuclide that has entered it.

chronic intake. An intake over an extended period of time that cannot be assumed to have been instantaneous.

derived air concentration (DAC). A derived limit on the activity concentration in air of a specified radionuclide, calculated such that a typical worker, breathing air with constant contamination at the DAC while performing light physical activity for a working year, would receive the annual limit on intake for the radionuclide in question. Calculated as the annual limit on effective dose divided by the dose coefficient, $e(50)_{inh}$, and the volume of air inhaled by the reference adult worker in a working year (2.4×10^3 m^3). The unit of DAC is Bq/m^3.

dose coefficient. The committed equivalent dose to tissue per unit intake at age t_o, $h_T(\tau)$, or committed effective dose per unit intake, $e(\tau)$, where τ is the time period in years over which the dose is calculated, i.e. 50 years for adults and $(70 - t_o)$ years for children. The unit of $h_T(\tau)$ or $e(\tau)$ is Sv/Bq.

effective half-life (T_e). The time taken for the amount of a radionuclide deposited in a living organism to be reduced by 50% as a result of the combined action of radioactive decay and biological elimination, i.e.:

$$T_e = \frac{T_b T_p}{T_b + T_p}$$

where T_b is the biological half-life and T_p is the physical half-life.

intake. The act or process of taking radionuclides into the body by inhalation or ingestion or through the skin, or the activity taken into the body by that act or process.

minimum detectable activity (MDA). That activity in a sample which produces a counting rate that will be detected (i.e. considered to be above background) with a certain level of confidence. In a sample containing activity equal to the MDA, random fluctuations will produce a counting rate less than the decision level with a certain probability β (normally taken to be 5%), thereby resulting in a false negative (Type II error). This amount is also referred to as the detection limit or lower limit of detection and the counting rate at this amount is also referred to as the determination level. A sample containing exactly the MDA will be taken to be free of activity only 5% of the time. (See Appendix II).

minimum significant activity (MSA). That activity in a sample which produces a counting rate that may be reliably distinguished from background with a certain level of confidence. Random fluctuations in measurements of a suitable blank (including all sources of variation) will produce a counting rate at or above this level with a certain probability α (normally taken to be 5%). Thus, observation of a counting rate equal to that at the MSA level results in a false positive (Type I error) in only 5% of the cases. This amount is also referred to as the decision limit and the counting rate at this amount is also referred to as the critical level. A sample containing exactly the MSA will be taken to be free of activity 50% of the time (i.e. the counting rate will be less than that corresponding to the decision level), but a true background sample will be taken to contain activity only 5% of the time. (See Appendix II).

physical half-life (T_p). The time taken for the activity of a radionuclide to decrease by 50% as a result of radioactive decay.

uptake. The processes by which radionuclides enter the body fluids from the respiratory tract or gastrointestinal tract or through the skin, or the fraction of an intake that enters the body fluids by these processes.

Human Respiratory Tract Model

alveolar–interstitial (AI) region. The respiratory bronchioles, alveolar ducts and sacs with their alveoli, and the interstitial connective tissue.

AMAD (activity median aerodynamic diameter). The value of aerodynamic diameter[2] 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

[2] The aerodynamic diameter of an airborne particle is the diameter that a sphere of unit density would need to have in order to have the same terminal velocity when settling in air as the particle of interest. The thermodynamic diameter is the diameter that a sphere of unit density would need to have in order to have the same diffusion coefficient in air as the particle of interest.

associated with particles larger than the AMAD. Used when deposition depends principally on inertial impaction and sedimentation, typically when the AMAD is greater than about 0.5 μm. For smaller particles, deposition typically depends primarily on diffusion, and the AMTD (activity median thermodynamic diameter) — defined in an analogous way but with reference to the thermodynamic diameter[2] of the particles — is used.

bronchial (BB) region. The trachea and bronchi.

bronchiolar (bb) region. The bronchioles and terminal bronchioles.

clearance. The removal of material from the respiratory tract by particle transport and by uptake.

deposition. The initial processes determining how much of a material in inhaled air remains in the respiratory tract after exhalation. Deposition of material may occur during both inhalation and exhalation.

extrathoracic (ET) airways. The anterior nose (ET_1) and the posterior nasal passages, mouth, pharynx and larynx (ET_2).

particle transport. Processes that clear material from the respiratory tract to the gastrointestinal tract and to the lymph nodes and move material from one part of the respiratory tract to another.

thoracic (TH) airways. The bronchial (BB), bronchiolar (bb) and alveolar–interstitial (AI) regions.

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