Australian Government
Australian Radiation Protection and Nuclear Safety Agency

SAFETY GUIDE

Monitoring, Assessing and Recording Occupational Radiation Doses in Mining and Mineral Processing

RADIATION PROTECTION SERIES No. 9.1

Radiation Protection Series

The ***Radiation Protection Series*** is published by the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) to promote practices which protect human health and the environment from the possible harmful effects of radiation. ARPANSA is assisted in this task by the Radiation Health and Safety Advisory Council, which reviews the publication program for the ***Series*** and endorses documents for publication, and by the Radiation Health Committee, which oversees the preparation of draft documents and recommends publication.

There are four categories of publication in the ***Series***:

Radiation Protection Standards set fundamental requirements for safety. They are prescriptive in style and may be referenced by regulatory instruments in State, Territory or Commonwealth jurisdictions. They may contain key procedural requirements regarded as essential for best international practice in radiation protection, and fundamental quantitative requirements, such as exposure limits.

Codes of Practice are also prescriptive in style and may be referenced by regulations or conditions of licence. They contain practice-specific requirements that must be satisfied to ensure an acceptable level of safety and security in dealings involving exposure to radiation. Requirements are expressed in 'must' statements.

Recommendations provide guidance on fundamental principles for radiation protection. They are written in an explanatory and non-regulatory style and describe the basic concepts and objectives of best international practice. Where there are related **Radiation Protection Standards** and **Codes of Practice,** they are based on the fundamental principles in the **Recommendations**.

Safety Guides provide practice-specific guidance on achieving the requirements set out in **Radiation Protection Standards** and **Codes of Practice**. They are non-prescriptive in style, but may recommend good practices. Guidance is expressed in 'should' statements, indicating that the measures recommended, or equivalent alternatives, are normally necessary in order to comply with the requirements of the **Radiation Protection Standards** and **Codes of Practice**.

In many cases, for practical convenience, prescriptive and guidance documents which are related to each other may be published together. A **Code of Practice** and a corresponding **Safety Guide** may be published within a single set of covers.

All publications in the ***Radiation Protection Series*** are informed by public comment during drafting, and **Radiation Protection Standards** and **Codes of Practice**, which may serve a regulatory function, are subject to a process of regulatory review. Further information on these consultation processes may be obtained by contacting ARPANSA.

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The mission of ARPANSA is to protect the health and safety of people, and to protect the environment, from the harmful effects of radiation.

Published by the Chief Executive Officer of ARPANSA in June 2011

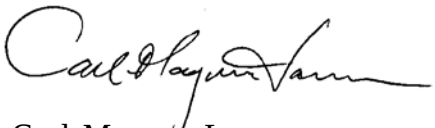
Foreword

The purpose of this Safety Guide is to promote a nationally consistent approach to monitoring, assessing and recording occupational exposures to radiation for mining and mineral processing operations.

This Safety Guide is a companion volume to the *Code of Practice and Safety Guide for Radiation Protection and Radioactive Waste Management in Mining and Mineral Processing* (the '*Mining Code*'), which was published in 2005. The Mining Code provides a uniform framework for radiation protection and radioactive waste management, including information and guidance to assist in development of a radiation management plan and a radioactive waste management plan.

This Safety Guide provides further, more detailed guidance on the specific topic of radiation monitoring, dose assessment and recording of doses for employees. It is intended to provide a practical and technically valid resource for industry and government. In particular, it should be a useful aid in the development and implementation of a radiation management plan for a mining and processing operation. Although the details of monitoring data collection and assessment are necessarily site and equipment specific, it is important that reported results are comparable despite technical specificities. The recent development of the Australian National Radiation Dose Register (ANRDR) has underlined the importance of such comparability, and this Safety Guide is expected to be an important supporting document to the ANRDR.

The draft Safety Guide was released for public comment from 29 March to 21 May 2010. The final version of the Safety Guide was approved by the Radiation Health Committee at its meeting of 16-17 March 2011. The Radiation Health and Safety Advisory Council advised me to adopt the Safety Guide on 15 April 2011.



Carl-Magnus Larsson
CEO of ARPANSA

30 June 2011

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Note: Technical terms which are described in the Glossary appear in **bold type** on their first occurrence in the text.

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1. Introduction

1.1 CITATION

This Safety Guide may be cited as the *Safety Guide for Monitoring, Assessing and Recording Occupational Radiation Doses in Mining and Mineral Processing (2011)*.

1.2 BACKGROUND

In mining and mineral processing **operations employees** may be exposed to naturally-occurring sources of **radiation**[1] in the work environment, including radiation from mined or processed materials. If **occupational exposures** are below regulatory concern, the practice may be exempt from regulatory control of radiation protection[2], but otherwise such exposure should be monitored, assessed and recorded in accordance with the *Recommendations for Limiting Exposure to Ionizing Radiation and National Standard for Limiting Occupational Exposure to Ionizing Radiation* (ARPANSA 2002) (hereafter called RPS 1) and, in particular, with the *Code of Practice and Safety Guide for Radiation Protection and Radioactive Waste Management in Mining and Mineral Processing* (hereafter called the Mining Code) (ARPANSA 2005). Normally, the assessed **dose** to employees will need to be reported to the **relevant regulatory authority** at quarterly and/or annual intervals.

In the past few years, there have been several developments in international recommendations for radiation protection and dose assessment, in particular with *The 2007 Recommendations of the International Commission on Radiological Protection* (ICRP 2007)[3], and with the *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources* (IAEA 1996a)[4]. It is anticipated that this Safety Guide will be updated, as necessary, as and when recommended Australian standards adopt new international recommendations[5].

In 2008, an Australian Government initiative, in cooperation with and partially funded by industry bodies, launched the development of the Australian National Radiation Dose Register (**ANRDR**). This followed a recommendation from the Steering Group of the Uranium Industry Framework (UIF) established by the Commonwealth Minister for Industry, Tourism and Resources in 2005, which involved representatives of the

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1 'Radiation' as used in the context of this Safety Guide means ionizing radiation.

2 A decision on this matter will be made by the relevant regulatory authority. General guidance is given in Section 3.5 of the Mining Code and in Section 4 of Radiation Protection Series No. 15 (ARPANSA 2008).

3 This Safety Guide has been made consistent with RPS 1 which is based on the earlier recommendations of ICRP Publication 60 (ICRP 1991).

4 At the time of preparation of this Safety Guide, a revision of the International Basic Safety Standards was nearing completion, taking into account the 2007 Recommendations of the ICRP.

5 At the time of preparation of this Safety Guide, ARPANSA had begun the process of revision of the Australian standards set out in RPS 1. Further, the ICRP was in the process of developing new guidance on occupational dose assessment calculations for intakes of radionuclides.

uranium industry; the Australian, South Australian and Northern Territory governments; and the Northern Land Council:

> **Recommendation 12**
> The Australian Government should work with relevant state and territory governments to establish cooperative arrangements with industry to ensure that permanent records of the radiological dose history of uranium industry workers are collected, maintained and retrievable. (Uranium Industry Framework 2006)

The purpose of the ANRDR is to implement this recommendation, with the potential to extend the coverage of the national register to include all mining and mineral processing operations which give rise to occupational exposure to radiation. It is intended that this Safety Guide should facilitate the implementation of the national register by promoting nationally consistent approaches to radiation monitoring, dose assessment and recording for mining and mineral processing operations.

1.3 OBJECTIVE

The objectives of this Safety Guide are to:

- provide a practical and technically valid resource for industry and government on methods for radiation monitoring, dose assessment and recording for employees in the mining and mineral processing industries
- promote a nationally consistent approach to assessing and recording occupational exposures to radiation for mining and mineral processing operations.

1.4 SCOPE

This Safety Guide is applicable to the exposure to radiation of employees in the mining and minerals processing industries. It includes guidance on the routine monitoring[6] of exposure, assessment of radiation dose and recording of assessed doses. It covers exposure from naturally-occurring **radioactive materials** mined and processed in the workplace, but does not address in detail exposures from other, artificial sources that may also be present, for which there is existing guidance[7]. The guidance is directed to operations involving mining and processing of uranium and of mineral sands but, in general, it is adaptable to all forms of mining and processing of naturally-occurring radioactive materials (NORM) in cases when occupational dose assessment is required by the relevant regulatory authority. This Safety Guide does not deal with the assessment of exposure of members of the public.

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6 Routine monitoring is undertaken during day-to-day operations primarily for the purpose of demonstrating that working conditions, including worker doses received, remain acceptable and meet regulatory requirements, and to support the process of optimisation of protection. Other forms of monitoring include task-related monitoring (to support decisions on the management of a specific task or to support optimisation of protection), and special monitoring (which is investigative in nature and intended to provide information on specific issues).

7 For guidance on dealing with such exposures, see for example: (ARPANSA 2004a, 2007) and (NHMRC 1989a, 1989b).

1.5 STRUCTURE

This Safety Guide is structured as follows:

Section 2 deals with the design of a monitoring and dose assessment program suitable for use in mining and mineral processing operations.

The next three sections describe typical processes for routine assessment of doses from the three major pathways of exposure. Section 3 covers exposure to radiation from external sources; Section 4 deals with exposure to **radon** and **radon progeny**; and Section 5 covers exposure from intakes of **radionuclides**, both by inhalation and by ingestion. All of these make assumptions about the characteristics of the exposure situation that allow a straightforward analysis to be carried out. However, since the circumstances of exposure in mining and mineral processing environments can be quite variable, more complex assessments may sometimes be required, and examples are given in Annex D.

Section 6 discusses the assessment of performance of a radiation protection program against dose criteria such as **dose constraints** and the occupational annual **effective dose** limit.

Section 7 covers record keeping by the **operator** or **employer** and reporting of dose assessments to the relevant regulatory authority.

Several annexes are included providing specific information on particular topics, such as quality management and the handling of uncertainties.

1.6 RELATIONSHIP WITH OTHER RADIATION PROTECTION SERIES PUBLICATIONS

This Safety Guide supports Radiation Protection Series No. 9, or RPS 9, *Code of Practice and Safety Guide for Radiation Protection and Radioactive Waste Management in Mining and Mineral Processing* (ARPANSA 2005).

Readers of this Safety Guide may find it useful to examine other related publications in the Radiation Protection Series, including:

RPS 1 (ARPANSA 2002) *Recommendations for Limiting Exposure to Ionizing Radiation and National Standard for Limiting Occupational Exposure to Ionizing Radiation*, which presents the overall radiation protection system and occupational and public dose limits.

RPS 6 (ARPANSA 2004b), *National Directory for Radiation Protection*, which provides an overall framework for uniformity including specific regulatory elements.

RPS 15 (ARPANSA 2008), *Safety Guide for the Management of Naturally Occurring Radioactive Material (NORM)*, which assists regulators and industries in which radionuclide concentrations in NORM may be enhanced, in managing NORM and assessing the need for radiation protection measures, including regulation.

2. Design of a Monitoring and Dose Assessment Program

Monitoring and assessment of occupational exposure to radiation should be undertaken within the context of a radiation management plan (RMP), as required by the Code of Practice (Section 2.7) and as described in the Safety Guide (Section 3.8) of the Mining Code. Operational mines also have management systems in place for non-radiological hazards, and the design of a radiation monitoring and assessment program should take this fact into account.

The monitoring and dose assessment part of the RMP should include the following components.

Firstly, a thorough evaluation of the characteristics of the workplace environment and the normal and potential occupational exposures should be undertaken. The radionuclides and radioactive materials present, their chemical and physical form, their location relative to occupied areas and workstations, the possible pathways for exposure, and the possible variability in any of the parameters relevant to dose assessment should all be established.

> The RMP should contain sufficient information to allow all significant exposure sources and pathways to be identified. This should include plans of the mine or processing plant, descriptions of the equipment to be used and processes involved, and estimates of the radionuclide concentrations in process streams. (Mining Code, 3.8.1(a))

Secondly, a program of regular monitoring, dose assessment and reporting should be implemented, with a frequency and accuracy appropriate for the circumstances:

> There are three main aims for monitoring, which should be addressed:
>
> - demonstration of compliance with regulatory limits, etc;
> - determination of doses received by individuals or groups; and
> - provision of information on the effectiveness of engineering and procedural control measures.
>
> Different monitoring techniques may be required to achieve these aims.
>
> The plan needs to provide information on the monitoring techniques to be used, and schedules of monitoring frequencies. Monitoring plans need to be flexible enough to respond to changing circumstances. (Mining Code, 3.8.1(c))

The type and frequency of monitoring should be suitable for the planned or normal exposures. It is common practice to 'designate' employees who are likely to receive significant doses. Such designated employees are then monitored more intensively (including, where appropriate, by personal monitoring) and their doses are assessed individually. Non-designated employees will be monitored less intensively; their doses might be assessed as a pro-rated average of their relevant work groups. When assessments are

based on pooled or averaged measurements, monitoring should be carefully planned to yield results that are representative of the work group.

The monitoring program may use periodic survey measurements ('walk-throughs' and area surveys), installed (fixed location) monitors, and personal monitoring, and will typically involve a combination of these.

Thirdly, and in particular where a default approach has been implemented using set values of parameters, periodic confirmation of the continuing validity of the dose assessment methodology should be carried out.

2.1 UNDERSTANDING THE EXPOSURE ENVIRONMENT

The characteristics of the workplace environment and the pathways for exposure should be carefully assessed in order to implement an appropriate monitoring program.

In some cases, the nature of the process will involve only physical treatment of the radioactive material. Thus exposure can arise from material at the mine face, in stockpiles, in the process stream and at waste repositories, from spillages, and from radon and airborne dust that is transported to other parts of the workplace. Use of appropriate shielding, ventilation, dust control measures and good housekeeping in cleaning up spills and residues within the workplace will help to restrict exposure.

In other cases, chemical processing and thermal treatments may affect the composition of the radioactive material, leading to additional possible pathways of exposure and to different parameters used in the assessment of exposures. Although there are exceptions, most mined raw materials are relatively insoluble and chemically stable. If taken into the body, for example through inhalation or ingestion (see Section 5), subsequent radionuclide uptake into blood and tissues is relatively slow. If chemically processed, however, they may be transformed into more soluble and reactive forms that are absorbed more rapidly in the body. This can have a significant effect on the radiation dose delivered. Processes involving high temperatures, such as calcining and smelting, also lead to material changes. They may create another exposure pathway through the generation of fumes.

In addition, the processing of the mined material may disturb the **secular equilibrium** between radionuclides in the uranium and thorium decay series, and may lead to different radionuclides presenting in different parts of the processing stream, including stockpiles and waste streams. Clearly, this can also have an effect on radiation dose as different radionuclides have different decay and radiation characteristics, environmental behaviour, and intake-to-dose conversion factors. When inhalation pathways are important, for example in underground mining, the ventilation conditions of the workplace are closely related to intake of airborne dust and radon progeny.

At some sites and for some employees, exposures may also arise from equipment containing artificial radioactive sources, such as thickness and density gauges, and from radiation generators, such as X-radiography equipment.

Once the nature of the radioactive materials and the pathways and circumstances of exposure have been established, consideration should be given to variations over time. This may arise from operational schedules for material processing, for example, or in some cases from seasonal climatic change. It is important that the monitoring program is able to determine a reliable long-term average exposure, as regulatory requirements typically relate to periods of one year or one quarter.

A sound analysis of the overall circumstances of exposure will allow the operational monitoring program to be properly designed and will support the validity of the dose assessment process. The evaluation should include some form of sensitivity analysis to establish which factors are the most critical for dose assessment. Such factors relate to the type of measurements made, choice of sampling location, frequency of measurement or measurement errors such as those of calibration. An evaluation will facilitate the subsequent reviews (see Section 2.3) of the adequacy of the operational monitoring program by drawing attention to the factors in the assessment which are likely to require the closest scrutiny.

2.2 ESTABLISHING AN OPERATIONAL MONITORING PROGRAM

An operational monitoring program should include a schedule of measurements to be made, their locations, dates and times, and the equipment and techniques to be employed in each case. Care should be taken to ensure that area monitoring results can be readily linked with employee occupancy records. The program documentation should also include the methods of calculation to be used to derive dose assessments from the monitoring results.

In some cases, assessments of dose to employees within work groups are made using personal monitoring data from a subset of individuals within the work group. In such cases, an effort should be made to establish that the work group is approximately homogeneous. One related approach is to utilise randomisation of choice of employees for personal monitoring within the work group.

As the three primary exposure pathways – external gamma radiation, inhalation of radon and its progeny, and intakes of radionuclides – have distinct requirements for monitoring, each of these should be dealt with separately in the monitoring program.

2.2.1 External gamma radiation

The monitoring program should specify that employees who are potentially exposed to significant gamma radiation and work in a **controlled area** are to be issued with personal dosemeters (normally on an individual basis), such as Thermoluminscent Dosemeters (TLDs) or Electronic Personal Dosemeters (EPDs) (see Section 3.1). The system for issuing the dosemeters and the database for recording results should ensure that dose records are accurately assigned to wearers and that there are no continuity gaps in any individual's monitoring record. Procedures for issuing and collecting dosemeters, for updating dose records, and for cases when dosemeters are lost or damaged, should be documented and followed.

In some cases, where cost-effective, employees in **supervised areas** may also be issued with personal dosemeters, but otherwise doses should be assessed through area monitoring and occupancy records. The monitoring program should specify the monitoring locations, the equipment to be used, and the frequency with which monitoring is undertaken. There should be a robust mechanism to link monitoring results to employee occupancy data for the monitored areas.

2.2.2 Inhalation of radon and radon progeny

At the time of preparation of this Safety Guide, workplace area monitoring combined with time-in-location data was the most common industry approach to monitoring for inhalation of radon progeny. In principle, personal monitoring for radon progeny is the most accurate method of monitoring. However, at the present state of development it requires careful compliance with procedures if the results are to be reliable, is quite labour-intensive, and is integrating rather than giving direct feedback to assist in dose control. Monitoring for radon gas is usually not suitable for dose estimation where there is significant variation in the **equilibrium factor**.

Careful consideration should be given to the monitoring and sample handling techniques to be employed. When using commercially available personal air samplers, the operating instructions should be closely adhered to. Where counting of **activity** on filter papers is required, correct handling of filters and adherence to prescribed counting regimes is crucial for reliable results.

In some cases, attention may need to be paid to the presence of **thoron** (Rn-220) and its progeny, because a proportion of any **thoron progeny** which are present may contribute to the count rate (ARL 1990). This proportion will depend upon the equipment and counting approach used. If significant thoron progeny may be present, then the suitability of the equipment and counting approach for the situation should be assessed. For some mining and milling operations, thoron progeny may be a significant contributor to the dose, in which case the **potential alpha energy concentration** (PAEC) for thoron progeny will need to be measured.

The monitoring program should specify the equipment to be used, including calibration documentation and procedures, size-selective performance of the sampling head, collection media (filter papers), flow rate, and the period for which any personal air sampler is to be worn during a working shift (the full-shift exposure being estimated pro-rata). For installed equipment, it should specify the locations, and the times, dates and periods of sampling.

2.2.3 Intakes of radionuclides

Inhalation

The monitoring program should include frequent personal monitoring of employees who are potentially exposed to significant dose from inhalation of long-lived **alpha particle** activity. While personal air samplers usually have a low flow rate (~0.1-0.2 $m^3\ h^{-1}$) and collect only small quantities of material for analysis, they have the advantage over positional samplers of collecting air close to an employee's breathing zone wherever the employee goes in the workplace. The allocation of personal monitors should be carefully planned

to yield representative results that can be used to derive full-shift and work group estimates. The program should specify the sampling equipment used, calibration documentation, collection media, flow rates, the periods for which the monitor is worn, the equipment and methodology for measuring the activity on the collection medium, and the computations to be used to estimate intake.

For employees working in areas where inhalation of long-lived alpha particle activity is a low contributor to employee dose, area monitoring using positional air samplers with an appropriate flow rate should be undertaken. High-volume samplers (~60-80 $m^3 h^{-1}$) are generally preferred, especially when airborne dust concentrations are low, but may not be suitable in all situations due to their size. Medium-volume samplers (~1 $m^3 h^{-1}$) are useful in small workspaces or locations that are difficult to access as it may be possible to use battery power.

In the case of underground mining operations, an assessment should be made of the potential for dose from the output of surface ventilation fans from long-lived alpha particle activity and radon progeny.

Ingestion

Good housekeeping to keep the working environment as clean as practicable, and good personal hygiene that keeps the risk of ingestion as low as practicable, should ensure that the ingestion pathway is only a minor contributor to total dose. Surface contamination monitoring may be used to support the implementation of ingestion control measures within the radiation management plan. Examples of this are the use of surface contamination monitoring during checking of machinery which is to leave a supervised site for repair, and the use of properly placed stations for employees to check for contamination on their hands during or following a work shift.

Where there are grounds for suspecting significant intakes may occur, assessment of received doses using analysis of biological samples, such as urine, may be considered (see Section 5.1.3).

2.2.4 Occupational exposure and natural background radiation

Regulatory requirements for occupational radiation protection, particularly dose limits, refer to 'occupational exposure'. This is defined (see Glossary) as: 'exposure of a person to radiation which occurs in the course of that person's work and which is not **excluded exposure**'. Excluded exposure means: 'in the context of occupational exposure, the component of exposure which arises from natural background radiation, provided that any relevant action level, or levels, for the workplace are not exceeded and that the relevant regulatory authority does not prohibit its exclusion'. These definitions follow the direction of international recommendations that it is the exposures caused by the operation (the business undertaking) that are controllable and subject to regulation, and that natural background radiation is not normally included.

Separating occupational exposure from excluded exposure is handled differently for each of the above primary pathways (2.2.1 to 2.2.3). For external gamma radiation assessed by integrating dosemeters such as TLDs, a control badge[8] issued with the worn badges allows the component of dose received by a badge outside work duties to be subtracted from the worn badge record. The control badge is kept with worn badges when they are stored between work duties. In some mining or mineral processing operations it may be necessary for workplace efficiency to store off-shift worn badges at a location where the gamma dose rate is regarded as not representative of natural background exposure. In such cases, a second control badge stored at a location representative of natural background may be used to adjust reported doses.

For inhalation of radon and radon progeny, no subtraction of background exposure is made in areas for which individual dose records are required to be kept. This approach reflects a compromise between strict interpretations of 'occupational exposure' and 'excluded exposure' and the impracticality of assessing what the background exposure would have been if an employee had not engaged in the work duties undertaken. In areas for which individual dose records are not required, radon and **radon progeny exposure** is regarded as excluded exposure unless the relevant action level or reference level is exceeded.

For intakes of radionuclides other than radon and radon progeny, the natural background level of exposure may be determined using samples taken at an appropriate point which is not affected by dust from the minesite. Alternatively the natural background level of exposure may be taken to be zero. While this is a compromise in the interests of practicality, it is generally considered that intakes would be small in the absence of the work duties undertaken, and that any consequent overestimation of occupational exposure would not be of concern for regulatory purposes.

2.3 CONFIRMATORY MONITORING AND QUALITY MANAGEMENT

While the following Sections 3, 4 and 5 describe monitoring and dose assessment methods that are likely to be adequate in most cases, it should be borne in mind that exposure conditions in mining and mineral processing environments can be variable or have particular characteristics that need to be addressed. For example, it may be necessary to measure additional parameters, such as the particle size (**AMAD**) of airborne dust, the **unattached fraction** of radon progeny and the particle size of the attached fraction. When radon concentration measurements are made as a surrogate for measurement of radon progeny exposure, the equilibrium factor between radon and its progeny is needed. Further parameters may relate to the chemical composition and solubility of radioactive material or the use of personal protective equipment. Confirmatory monitoring and analysis should be carried out as necessary to validate the continued use of the routine

[8] The control badge is used to subtract the dose accumulated by a worker's badge due to excluded exposure (see Section 2.2.4).

methodologies implemented. If found to be inadequate, more complex procedures and analyses may be required, as indicated in Annex D. This is part of the process of quality management for ensuring that the monitoring and dose assessment practices are fit for purpose.

Quality management principles (Standards Australia 2008) should be applied to the processes involved in carrying out the monitoring and dose assessment tasks (see Annex C). This will not only result in confidence in the validity of the monitoring and assessment results, but also provide a robust documentary record, and help improve comparability of dose assessments between different mining operations.

3. Doses from external sources

Naturally-occurring radioactive materials emit **ionizing radiation** in the form of **gamma rays**, alpha particles and **beta particles** which can irradiate the tissues and organs of workers who may be exposed. When the source of the exposure – the radioactive material – is outside the body, it is known as an external source of radiation. Monitoring of occupational exposure to gamma rays in this context is usually straightforward, since personal dosemeters for detecting and quantifying exposure are readily available and relatively inexpensive to use. They are worn on the body and integrate exposure over a given wearing period. Depending on the dosemeter or dosimetry service, readouts and results are generally reported as **personal dose equivalent**, $H_p(0.07)$ or $H_p(10)$. However, effective dose, E, is needed for comparison with regulatory standards. For the radiations encountered in mining and mineral processing of raw materials, and for the normal situation in such operations where external irradiation is reasonably uniform rather than targeting particular areas of the body, $H_p(10)$ may be considered an acceptable estimator of effective dose for the purpose of demonstrating regulatory compliance.

When handling some processed materials or residues, or when such materials are allowed to remain on the skin or clothing, it may be necessary to make an assessment of personal dose equivalent, $H_p(0.07)$, to the hands or skin, taking beta radiation into account.

3.1 MONITORING AND MEASUREMENT OF EXPOSURE

Employees who are potentially exposed to significant dose from gamma radiation should be issued with personal integrating dosemeters. These may be of a passive type, typically a TLD, or an active type, such as an EPD. TLDs have the benefit of a smaller capital outlay and usually the dosemeters and results are provided by a commercial supplier. However, there can be a substantial period (some weeks) between the end of a monitoring period and the return of the dose results from the monitoring service. EPDs can be read out immediately following a shift, and some EPDs have an additional advantage that they can serve as alarming monitors. EPDs require regular (normally annual) calibration and require labour to record the doses or, when readouts are coupled directly to a computer database, to ensure the quality management of the data. Experience shows that there is a small but significant loss rate of personal monitors in the rugged operational environments encountered in mining and mineral processing (possibly as much as 4%-5% of issued monitors). These latter factors militate against the more expensive EPDs for routine use.

The dosemeters should be capable of measuring personal dose equivalent, $H_p(0.07)$ or $H_p(10)$, with an adequate accuracy for the energies of the radiation concerned. The dosemeters should have a lower limit of measurement of about 0.07 mSv or less. Personal dosemeters should normally be worn on the trunk at chest or waist height. The wearing period is typically three months, though in some higher dose rate locations a period of one month may be more appropriate. If monitoring of extremities is required – such as monitoring of fingers when handling highly active material, advice should be sought from the Radiation Protection Officer.

For employees employed in supervised areas, workplace monitoring coupled with occupancy records may be adequate for estimating external exposure, or alternatively, and where approved by the relevant regulatory authority, the pooled data from a monitored representative subset of employees may be used to estimate doses to non-monitored employees from the same work area.

3.2 ASSESSMENT OF DOSE

When personal dosemeters are worn, assessment of dose from external exposure is straightforward, as the dose reported for the dosemeter by the monitoring service may be taken as the effective dose absorbed by the wearer. When area monitoring is employed, it may be carried out using fixed TLD or gamma monitors. The reported results for the dosemeter may then be combined with occupancy records to assess individual doses for employees who work in the area. In cases where short-term dose rate measurements are made in situ, these need to be time-averaged over the assessment period and combined with occupancy records.

Example A

An underground uranium mine worker is issued with a personal TLD monitor worn as a chest badge. The badge is collected by the worker at the start of each shift from a holding location where a control badge is also kept, and returned to the holding location at the end of each shift. Following a wearing period of 3 months, the dosemeter is sent to a personal monitoring service for analysis. It is returned with a report that the badge recorded a personal dose equivalent of 0.7 mSv, following subtraction of the control badge reading. The worker's dose record is then updated to show that he received an effective dose of 0.7 mSv for the quarter.

Example B

Workers in a particular supervised area are not all badged, but personal dosemeters are issued to a carefully chosen cross-section of the work group[9]. From their pooled reported doses the average effective dose for the quarter is 0.3 mSv for the work area. All non-badged workers in the area are assigned a quarterly effective dose of 0.3 mSv (pro-rated if they have not worked a full quarter), while the badged workers are assigned their individual reported doses.

Example C

The dose received by an office worker in the administration building at a mineral sands processing plant is checked by using a fixed TLD monitor in the office area to provide a long-term average dose rate. A nominal working period of 500 hours per quarter is used to check that the worker's quarterly dose remains below the relevant recording level. An individual dose record is not maintained[10].

[9] Some rotation of badging among individual workers quarter by quarter may be appropriate.

Example D

A technician works mostly in the office building of Example C but she has occasional duties in the supervised area of Example B testing equipment performance. The external exposure within the office building is considered to be background radiation (excluded exposure) and is not recorded[10]. When working in the supervised area, she is not issued with a personal dosemeter, but a record of the times she spends in the supervised area is maintained: a total of 70 hours for the quarter. From the pooled reported doses in the supervised area (see Example B) an average effective dose of 0.3 mSv for the quarter is obtained. The quarterly effective dose, E, to the technician is then estimated from:

$$E = 0.3 \times \frac{70}{500} = 0.04 \text{ mSv}$$

where 500 hours is the nominal working period for the quarter.

[10] The exposure is considered to be indistinguishable from natural background exposure and is excluded from assessment of occupational exposure. Note that the dose rate in office areas at some locations may not lie below the recording level, in which case doses should be estimated.

4. Doses from inhalation of radon/thoron and radon/thoron progeny

In undisturbed raw materials containing uranium or thorium the radionuclides of each decay series are typically present in approximate secular equilibrium. This may not be the case when geochemical processes have preferentially removed particular elements, such as radium, or when the material is at the ground surface and radon is released to the atmosphere. Some radon and thoron will be present above ground where uranium and thorium deposits occur, through diffusion through pore spaces and emanation through the ground surface. When mining takes place, pathways for release to air are opened leading to increased concentrations of radon and thoron and of radon progeny and thoron progeny.

Radon and radon progeny

Monitoring of radon (Rn-222) and its progeny in a mining or mineral processing environment focuses on measurement of the short-lived progeny. This is because assessment of the radon progeny in terms of PAEC more directly correlates with dose to lung tissue and risk of lung cancer, especially in conditions of secular disequilibrium. There are several active air sampling techniques of varying complexity, reliability and cost, including manual sampling and analysis (see for example Chapter 5 of ARL (1990) and Appendix A of the guideline NORM-3.4 (DMPWA 2008)) and microprocessor-controlled automated systems.

Where radon progeny concentrations are low and reasonably stable, measurement of radon-222 may be adequate for dose assessment purposes. A knowledge of the equilibrium factor between radon and its progeny is required, but measurement of radon allows passive integrating monitors to be used for long-term average results.

Thoron and thoron progeny

Exposure to thoron (Rn-220) and thoron progeny is usually only of concern in mining and mineral processing of raw materials containing significant concentrations of thorium-232. Monitoring and dose assessment for thoron and thoron progeny may be required for such operations. The monitoring techniques are similar to those for radon (Rn-222) progeny, but with sample activities counted after the short-lived Rn-222 progeny have largely decayed (ARL 1990, DMPWA 2008).

4.1 MONITORING AND MEASUREMENT OF EXPOSURE

Monitoring of radon progeny intake may be undertaken through area monitoring (in which a dosemeter or sampler is located within the workplace and the employee's occupancy of the area is recorded), or through personal monitoring (in which a passive dosemeter or an air sampler is worn on the body). In addition to estimating employee doses, measurements may be carried out to derive reference levels of radon progeny concentration to assist in deciding which type of sampling to employ, or to determine the boundary of a controlled area.

Area monitoring may be either by grab sampling or by continuous monitoring, depending upon the particular situation. In either case, the method used should be capable of measuring radon progeny PAEC (ARL 1990).

Active personal sampling has the advantage in principle of focussing on individual intake from the breathing zone. Where technically feasible personal air sampling is recommended for employees working in areas where the radon progeny concentrations could potentially cause exposures to exceed the relevant dose constraint.

4.2 ASSESSMENT OF DOSE

4.2.1 Intake-to-dose conversion for radon-222 progeny

The **dose conversion convention** for radon progeny is given by:

$$E = e_{RnD}\, I_{RnD} \quad \ldots (1)$$

where e_{RnD} is the effective dose per unit exposure to radon progeny, and

I_{RnD} is the potential alpha energy exposure in mJ h m^{-3}.

The conversion coefficient e_{RnD} has the value[11] 1.4 mSv per mJ h m^{-3} (ICRP 1995; IAEA 1996a).

Example E

A uranium mine employee works partly in the open-pit mine and partly in a workshop adjacent to the pit. Continuous radon daughter monitors are used in the pit and in the workshop to record radon progeny concentrations in PAEC. Over a work shift of 8 hours, the pit monitor records a PAEC of 0.3 μJ m^{-3} and the workshop monitor records 0.04 μJ m^{-3}. During the shift, the worker spent 2 hours in the pit and 6 hours in the workshop. The employee's radon progeny exposure for the shift is calculated as:

$$I_{RnD} = 0.3 \times 10^{-3} \times 2 + 0.04 \times 10^{-3} \times 6 = 8.4 \times 10^{-4} \text{ mJ h m}^{-3}$$

From Equation (1), the effective dose from radon progeny for the shift is:

$$E_{RnD} = 1.4 \times 8.4 \times 10^{-4} = 1.2 \times 10^{-3} \text{ mSv or } 1.2 \text{ μSv}$$

4.2.2 Intake-to-dose conversion for radon-222

Monitoring of radon, whether by personal sampling or area monitoring, will normally provide results in terms of the average concentration of radon-222 over a given period. Conventionally, assessment of dose from inhalation of radon is made in terms of radon concentration or **radon exposure** – the product of radon concentration in air and the period of time exposed – rather than intake. The International Commission on Radiological Protection has recommended a dose conversion convention to estimate effective dose from

[11] At the time of preparation of this Safety Guide, an updated value was expected to be soon recommended by ICRP.

radon exposure. Historically, this has been expressed in several different ways and in different units. For the purpose of this Safety Guide, the dose conversion convention for radon can be expressed as:

$$E = e_{Rn} I_{Rn} \quad \text{... (2)}$$

where E is the effective dose in mSv that corresponds to exposure[12] I_{Rn} in Bq h m^{-3}, and

e_{Rn} is the conversion coefficient that gives the effective dose per unit exposure to radon at work, which has the value[13], assuming an equilibrium factor of 0.4 and a nominal breathing rate of 1.2 m^3 h^{-1}, of 3.1 × 10^{-6} mSv per Bq h m^{-3} (ICRP 1995; IAEA 1996a).

Example F

An employee at a uranium mine works in an above-ground maintenance area some distance from the open pit and from ore processing operations. Radon concentration is measured using track etch detectors at fixed locations in the workplace. The results for the quarter yield an average radon concentration of 135 Bq m^{-3}. Having worked for 480 hours the employee's quarterly effective dose from exposure to radon is assessed as:

$$135 \times 480 \times 3.1 \times 10^{-6} = 0.2 \text{ mSv}$$

In many areas on an active mine site the equilibrium factor may differ considerably from the nominal value of 0.4. Further, most of the dose to lung tissue is delivered by radon progeny present with the radon, rather than by radon-222 itself (see Annex D). Consequently, measurement of radon progeny PAEC is preferred for dose assessment purposes, although radon measurements may be useful for other (non-dose calculation) applications.

4.2.3 Intake-to-dose conversion for radon-220 and progeny

The dose conversion convention for thoron can be expressed as:

$$E = e_{Tn} I_{Tn} \quad \text{... (3)}$$

where E is the effective dose in mSv that corresponds to exposure I_{Tn} in Bq h m^{-3}, and

e_{Tn} is the conversion coefficient that gives the effective dose per unit exposure to thoron at work, which has the value, assuming an equilibrium factor of 1[14] and a nominal breathing rate of 1.2 m^3 h^{-1}, of 3.6 × 10^{-5} mSv per Bq h m^{-3} (ICRP 1995; IAEA 1996a).

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12 In the case of exposure to radon and radon progeny, it is conventional to work in terms of exposure rather than intake. However, the symbol, I, is used, as for intake of radionuclides (see Section 5.1).

13 Gamma emitting radionuclides on the skin also irradiate tissues at depth in the body, but this is treated as external exposure.

14 The equilibrium factor is likely to be variable and different from 1. However, except for cases of extreme disequilibrium between Rn-220 and Bi-212, the conversion coefficient is relatively insensitive to the value of the equilibrium factor.

The dose conversion convention for thoron progeny is given by:

$$E = e_{TnD} I_{TnD} \quad \text{... (4)}$$

where e_{TnD} is the effective dose per unit exposure to thoron progeny. The conversion coefficient e_{TnD} has the value 0.48 mSv per mJ h m^{-3} (ICRP 1995; IAEA 1996a), and

I_{TnD} is the potential alpha energy exposure in mJ h m^{-3}.

Example G

An employee at a mineral sands secondary separation plant works near to a set of air separation tables. A program of manual sampling and assessment of the air using the Rock method (see NORM-3.4 (DMPWA 2008)) indicates an average thoron progeny concentration in PAEC of 42 nJ m^{-3}. The employee's thoron progeny exposure for a quarter (500 hours) is calculated as:

$$I_{TnD} = 4.2 \times 10^{-5} \times 500 = 2.1 \times 10^{-2} \text{ mJ h m}^{-3}$$

From Equation (4), the effective dose from thoron progeny for the quarter is:

$$E_{TnD} = 0.48 \times 2.1 \times 10^{-2} = 1.0 \times 10^{-2} \text{ mSv or } 10 \text{ µSv}$$

5. Doses from intakes of radionuclides

When working with unsealed radioactive material, intakes of radionuclides may occur through inhalation or ingestion. In addition, some chemical forms of radioactive material may be absorbed through the skin[13]. Skin absorption may also occur if employee injuries such as cuts and abrasions have not been properly protected. In **accident** situations, there may be an intake through wounds[15]. If an intake of radionuclides occurs, internal organs and tissues of the body are exposed to radiation, and there is a need to assess the doses delivered.

The usual purpose of occupational dose assessment is to estimate the effective dose[16] received by an employee during a defined period of time. This can then be compared with any applicable dose constraints and, over a twelve-month period, with annual dose limits. Assessed doses can also be used to check the effectiveness of the radiation protection program in optimising protection. In some circumstances, when the radionuclides ingested or inhaled preferentially irradiate specific parts of the body, it may be necessary to estimate **equivalent doses**[16] to organs and tissues.

The dose estimates depend on monitoring and measurement data. In most cases, provided doses are comfortably below relevant limits, measurements of radionuclide concentration within the workplace, whether by area monitoring or by personal monitoring, are adequate for dose assessment purposes. In some cases, for example if a suspected overexposure has occurred, direct measurement of radionuclides within the body or indirect measurement through analysis of excreta or other biological samples may be necessary. However, the difficulty of interpretation of results from such biological samples mean that this approach is not normally used for routine monitoring. Consequently, for the ingestion pathway the best approach is prevention, such as through use of hand surface contamination monitors at mine exit points and before eating areas.

The standard method of assessment is first to estimate intake from monitoring results, and then to convert intake to dose through published conversion coefficients obtained from biokinetic and dosimetric modelling. In many cases, where the inventory of radionuclides, their chemical form and their physical properties are known and stable, a single conversion coefficient appropriate for the circumstances may be adequate. Where this is not the case, individual radionuclides may need to be assessed separately, and additional factors such as chemical form and particle size may need to be taken into account.

5.1 MONITORING AND MEASUREMENT OF EXPOSURE

Operational monitoring and dose assessment of employees should be undertaken within a radiation monitoring and dose assessment program that has been designed for the purpose. The first step of the design process is to

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15 The skin and wound pathways are beyond the scope of this Safety Guide. If needed, further information should be sought from specialist providers of dosimetry services.

16 The terms ‘effective dose’ and ‘equivalent dose’ may be assumed to mean ‘committed effective dose’ and ‘committed equivalent dose’, respectively, when used in the context of dose from the intake of radionuclides.

identify the possible pathways of exposure. For intakes of radionuclides, this involves both inhalation of airborne dust containing radioactive material and ingestion of radioactive material taken into the mouth. The pathway evaluation should therefore examine possible sites for creation or resuspension of dust and possible scenarios for contamination of food, drink and the skin. Normally, consumption of food and drink will not be permitted in work areas prone to radioactive dusts, and smoking will be prohibited. Where it is necessary for rehydration drinks to be taken within the workplace, due regard should be given to the cleanliness of the hands and drink containers, in conformity with the working rules of the site.

Once possible pathways of exposure have been identified, each step within that pathway should be analysed to establish the characteristics that bear upon dose assessment. These include the radionuclide composition of process materials and dust, the characteristic particle size of airborne dust (AMAD), the chemical form of the radioactive material, and the typical concentrations of airborne dust and their variation over time and location. The analysis should be sufficient to allow the workplace to be 'mapped' for the purposes of implementing an operational monitoring program and to allow appropriate monitoring equipment and sampling regimes to be selected. Attention should be paid to any seasonal variations over the year, and to possible malfunction conditions of the operation. Often, it will be found that a small number of dominant exposure pathways contribute most of the dose received by employees. Minor pathways, contributing less than 5% to 10% of total dose, may not need to be routinely monitored.

5.1.1 Monitoring of intakes of radionuclides

Monitoring of radionuclide intake by inhalation may be undertaken through personal air sampling (in which the sampler is worn on the body) or through area monitoring (in which a sampler is located within the workplace and the employee's occupancy of the area is recorded). Personal sampling has the advantage in principle of focussing on individual intake. However, the sampling rate is lower than for an area sampler, and carrying the sampler and battery supply may have its drawbacks for the wearer. In general, personal sampling is recommended for employees who work in controlled areas where the dust inhalation pathway contributes significantly to total dose or where the nature of the work involves variable occupancy of multiple dust-exposure locations, making it difficult to obtain reliable occupancy rates in each area. It may be useful to derive reference levels of radionuclide concentration in air – based on the consequent implied dose – to assist in deciding which type of sampling to employ and, for area sampling, in establishing sampling criteria.

Both types of sampling involve drawing air through a filter paper at a known rate for a known period and subsequently counting the activity collected on the filter. Personal samplers operate at low flow rates – typically about 2 litres per minute, which would draw a volume of about 1 m^3 if worn for a full 8-hour shift. Area samplers operate at much higher flow rates and can collect a greater quantity of dust. This can make it easier to quantify the radioactive material on the filter, but it may also introduce technical difficulties related to the larger filter diameter and the possibility of self-absorption of alpha particles by the dust when counting. The design of the sampling head that holds the filter paper may have an effect on the efficiency

of collection of the dust, and may cause the efficiency to vary significantly with the size of the dust particles. Techniques for reliable sampling are quite complex and are not discussed in this Safety Guide. The standard method for conducting workplace sampling of inhalable dust is detailed in Australian Standard 3640 (Standards Australia 2004). Some further information is available in the guideline NORM-3.4 (DMPWA 2008) on the application of AS 3640 to sampling of radioactive dust in mining operations.

For a flow rate through the filter of $\dot{v}_f$ litres per minute, a sampling time of t minutes, and a measured activity on the filter paper of a_j in Bq for radionuclide j, the average activity concentration in air c_j of radionuclide j in Bq m^{-3} during sampling may be found from:

$$c_j = \frac{a_j \times 1000}{\dot{v}_f \times t} \quad \ldots(5)$$

Identifying individual radionuclides requires a spectroscopic analysis of the filter, which can be complex and costly. For dusts encountered in the mining and mineral processing industries, the most significant radionuclides contributing to dose (~90-95% of dose) are alpha emitters. Further, the mixture of radionuclides, including the state of secular equilibrium in the uranium and thorium series, is often well known and stable[17]. Consequently, dose calculations are typically worked in terms of the total alpha particle activity[18]. Measurement of gross alpha particle activity on a filter is relatively simple. In this case, Equation 5 becomes:

$$c_\alpha = \frac{a_\alpha \times 1000}{\dot{v}_f \times t} \quad \ldots(6)$$

where a_α is the measured gross alpha particle activity in Bq on the filter, and

c_α is the calculated average alpha activity concentration in Bq m^{-3} during the sampling period.

The Mining Code uses a non-standard unit of alpha decays per second – αdps – for the purpose of distinguishing gross alpha activity from total activity measured in Bq. In this Safety Guide, αdps has been replaced by Bq$_\alpha$.

Assuming that there are i samplings during an assessment period, the average alpha activity concentration over all sampling periods, C_α, is:

$$C_\alpha = \frac{\sum_i c_{\alpha,i} \times t_i}{\sum_i t_i} \quad \ldots(7)$$

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17 This is not always the case. For example, changes in ventilation, particularly underground, may cause significant changes to the relative concentrations of radionuclides. In such cases, assessment of individual radionuclides and use of Equation (5) may be required.

18 A conversion coefficient from alpha activity to dose that is appropriate for the mixture needs to be used: see Section 5.2.1 and Equation 12.

Note that this is a time-weighted average concentration, which ensures that the subsequently calculated intake is a valid cumulative estimate over all of the sampling periods. It is assumed that this average is representative of the whole assessment period, *T*, including the periods during which samples were not taken.

The intake I_T during the assessment period is then:

$$I_T = C_\alpha \times \dot{v}_b \times T_w \quad \ldots(8)$$

where $\dot{v}_b$ is the assigned breathing rate in $m^3\ h^{-1}$ for the work activity, and

T_W is the total time worked by the individual during the assessment period *T*.

When the exposure of an employee occurs in more than one location, with significantly different activity concentrations, Equation 8 should be replaced by:

$$I_T = \sum_k C_{\alpha,k} \times \dot{v}_{b,k} \times T_{w,k} \quad \ldots(9)$$

where subscript *k* refers to each of the separately identified locations.

When it is necessary to assess the intake of individual radionuclides, *j*, Equations 6, 7, 8 and 9 may be used by substituting subscript *j* for subscript α.

Example H

An employee at a mineral sands operation works in the secondary separation plant controlling the air tables. They are issued with a personal air sampler, which yields a gross alpha activity concentration of thorium dust of 0.2 $Bq_\alpha\ m^{-3}$ when time-averaged over a working week of 40 hours. The employee's intake of thorium dust for the week, at a nominal breathing rate of 1.2 $m^3\ h^{-1}$, is calculated (Equation 8) as:

$$I_T = 0.2 \times 1.2 \times 40 = 9.6\ Bq_\alpha$$

5.1.2 Direct methods of measurement of radionuclides in the body

Radionuclides within the body may be detected through the characteristic gamma radiation they emit during **radioactive decay**. Highly sensitive gamma-ray detectors may be employed, in either a partial-body or a whole-body geometry around the monitored individual. Whole-body monitoring within a well-shielded enclosure is generally more sensitive, but is complex and costly and needs to take place at a specialised facility. Very few such facilities are available, and they are likely to be employed only when exceptional circumstances arise, such as ingestion or inhalation of large quantities of radioactive material in an accident. Partial-body monitoring is possible with relocatable equipment, but is best suited to investigation of specific organs for which appropriate shielding can be arranged. Since much of the gamma-ray information comes from the decay of radionuclides one or several steps along the decay chain from the parent uranium or thorium, assumptions need to be made about the secular equilibrium of the ingested or

inhaled material in order to estimate the activity within the body. This is a specialised technique (IAEA 1996b; ICRU 2003) that is not commonly applied in routine monitoring and is not dealt with further in this Safety Guide.

5.1.3 Indirect methods of measurement of radionuclides in the body

Biological samples taken from an individual who has inhaled or ingested radioactive material may be analysed for their radionuclide content – a process known as bioassay. The most frequently-used technique is bioassay of urine. This can be useful for soluble forms of uranium and thorium and their progeny that are absorbed into the blood or gut and are excreted through urine, but the technique needs to be applied with care in order to distinguish a large intake of insoluble material from a small intake of soluble material. Such uncertainties, and the range of assumptions that need to be made concerning the metabolism of the material within the body, mean that the technique is often unsuited to regular use in estimating dose for routine monitoring purposes, unless there are grounds for suspecting significant intakes such as through accidental exposure. It may be best suited to situations where the chemical form of ingested material is well characterised, or to detection and investigation of failures in the system of protection in the workplace, for which accurate dosimetry may be less important than the observation of elevated levels of radionuclides in the body. There is a broad literature of sampling and analytical techniques (see, for example, IAEA 2000), the details of which are beyond the scope of this Safety Guide.

Other bioassay techniques include faecal sampling, which may be useful for detecting insoluble forms of uranium and thorium materials, and measurement of radon and thoron in breath, which can give an indication of the content of Ra-226 and Ra-224 in the lung. They are not normally used in routine dose estimation, but may provide an indicator of long term integrated intake.

5.2 ASSESSMENT OF DOSE

Once an intake of radioactive material has been estimated for an employee, whether as gross alpha activity or as individual radionuclide activities, the corresponding dose is obtained by applying the appropriate intake-to-dose conversion factor.

In general, the **committed effective dose** E from intakes of radionuclides can be calculated from the following formula:

$$E = \sum_j e_{j,ing} \; I_{j,ing} + \sum_j e_{j,inh} \; I_{j,inh} \qquad \ldots(10)$$

where $e_{j,ing}$ and $e_{j,inh}$ are the committed effective doses per unit intake by ingestion and inhalation for radionuclide j, and

$I_{j,ing}$ and $I_{j,inh}$ are the intakes via ingestion and inhalation of radionuclide j during the specified period.

Values of dose conversion factors are given in the International Basic Safety Standards (IAEA 1996a). For practical reasons, the ingestion and inhalation pathways are treated separately below.

5.2.1 Intake-to-dose conversion for inhalation of radionuclides

For inhalation alone, Equation 10 reduces to:

$$E = \sum_j e_{j,inh} \; I_{j,inh} \qquad ...(11)$$

Tables of intake-to-dose conversion factors, $I_{j,inh}$, for radionuclides of significance in the mining and mineral processing industries are given in the International Basic Safety Standards (IAEA 1996a).

In cases where the mixture of radionuclides is known and stable, this formula may be simplified to:

$$E = e_{inh} \; I_{inh} \qquad ...(12)$$

where e_{inh} is the committed effective dose per unit intake by inhalation for the mixture of radionuclides, and

I_{inh} is the intake.

It is important that the conversion coefficient, e_{inh}, is appropriate for the characteristics of the mixture, including particle size and chemical form.

For the inhalation pathway in some mining environments, as noted above, it is convenient to measure the long-lived alpha activity (LLAA) of the prevailing mixture of uranium-series and thorium-series radionuclides in terms of the total number of alpha particle decays per second (Bq_α). Default values of intake-to-dose conversion factors for mining and mineral processing environments can be taken from the Mining Code.

Example I

Continuing Example H from above, a mineral sands worker is found to have an inhalation intake of 9.6 Bq_α of thorium dust for a working week. It has been established that the intake-to-dose conversion factor for an AMAD of 5 µm given in the Mining Code is appropriate. The worker's effective dose from inhalation for the week is calculated (Equation 12 and the Mining Code) as:

$$E_{inh} = 8.0 \times 10^{-3} \times 9.6 = 0.08 \text{ mSv}$$

In cases where Annex A values do not apply – for example, when the mixture of radionuclides is different from the assumptions in the Mining Code – they should be replaced by values appropriate for a particular site, as necessary, from the International Basic Safety Standards (IAEA 1996a) or from other primary sources such as ICRP Publication 68 (ICRP 1994b) (see also Annex D).

Example J

A worker in the smelter area of a copper/uranium processing operation is exposed to a fine airborne particulate containing Po-210. Personal dust sampling and alpha spectroscopy of the collected dust shows that their Po-210 intake over a 3-month period is 310 Bq. Prior analyses have determined that the particulate should be treated as lung absorption type M, with an aerodynamic median thermodynamic diameter (AMTD) of 0.3 µm. From the ICRP database of dose coefficients for adult workers the conversion factor from intake to dose is 3.9×10^{-6} Sv Bq^{-1}. From these data, the worker's inhalation dose for the quarter for this pathway is calculated as:

$$E_{inh} = 3.9 \times 10^{-6} \times 310 \text{ Sv} = 1.2 \text{ mSv}$$

Allowance for personal protective equipment

In some workplace environments, personal protective equipment is issued to employees. If a respirator or face mask is worn in a dusty environment, for example, the quantity of dust inhaled will be less than would be estimated from a free air measurement. Provided that such protective equipment is properly used and that a reliable protection factor, *PF*, can be established, the assessed intake from free air measurements, I_f, should be corrected to allow for the protection factor:

$$I = \frac{I_f}{PF} \qquad \ldots(13)$$

This will avoid an unnecessary overestimate of the dose received and recorded.

It is difficult to establish with a high degree of confidence a protection factor that will be reliably maintained in a working environment. However, for dusts of typical aerodynamic diameter in mining and mineral processing environments (~5 µm), a protection factor in the range 2 to 5 may be afforded by paper masks correctly worn, while tight-fitting half-face respirators may provide a protection factor of up to 10. In areas where airborne particle sizes are much smaller, for example where smelting is carried out, simple paper masks do not provide a reliable reduction in intake. Close-fitting half-face respirators can be effective in reducing intake from fume, provided that they are not continually removed and refitted by employees. Great care needs to be taken in assigning a protection factor to respirators worn in these circumstances. The requirements of the Australian Standard for use and maintenance of respiratory protective devices (Standards Australia 1994) should be followed.

Example K

Continuing Example I from above, the mineral sands worker wears a face mask which reduces inhalation intake of thorium dust by a factor of 3. Their actual inhalation of thorium dust for the week is calculated as:

$$I = \frac{9.6}{3} = 3.2 \text{ Bq}_\alpha$$

Their effective dose from inhalation for the week is then calculated as:

$$E_{inh} = 8.0 \times 10^{-3} \times 3.2 = 0.03 \text{ mSv}$$

5.2.2 Intake-to-dose conversion for ingestion of radionuclides

For ingestion alone, equation 10 reduces to:

$$E = \sum_j e_{j,ing} \; I_{j,ing} \qquad \ldots(14)$$

Tables of intake-to-dose conversion factors, $I_{j,ing}$, for radionuclides of significance in the mining and mineral processing industries are given in the International Basic Safety Standards (IAEA 1996a).

In cases where the mixture of radionuclides is known and stable, this formula may be simplified to:

$$E = e_{ing} \; I_{ing} \qquad \ldots(15)$$

where e_{ing} is the committed effective dose per unit intake by ingestion for the mixture of radionuclides, and

I_{ing} is the intake.

It is important that the conversion coefficient, e_{ing}, is appropriate for the characteristics of the mixture, including the chemical and physical form.

Intake-to-dose conversion calculations for the ingestion pathway are complex. A bioassay will yield the activity concentration of radioactive material in the sample, from which the systemic activity in the body or the activity in relevant organs at the time of sampling, M, needs to be estimated, using biokinetic models appropriate for the chemical form and physical properties of the material. From this, the intake at the time of ingestion prior to sampling needs to be estimated, again using biokinetic modelling, which may be summarised in the equation:

$$I_{j,ing} = \frac{M}{m(t)} \qquad \ldots(16)$$

where $m(t)$ is a retention fraction (for direct measurement methods) or an excretion fraction (for bioassay measurements) over time since intake. The ICRP has proposed values of $m(t)$ for uranium, radium and thorium compounds, together with biokinetic models and parameters (ICRP 1997).

6. Assessment of performance against dose criteria

One of the main objectives of a radiation monitoring and dose assessment program is to show that doses received comply with applicable dose criteria, as part of the demonstration of adequacy of the radiation protection program. The primary regulatory requirement is to show that doses received by employees do not exceed the occupational annual effective dose limit.

6.1 TOTAL EFFECTIVE DOSE

For each employee for whom individual dose records are kept, the total effective dose, E, should be calculated and entered into the employee's dose record for the relevant recording period. The general calculation is:

$$E = \sum_j E_j \qquad \ldots(17)$$

where E_j is component j of the total effective dose.

In most cases, in a mining and mineral processing environment, application of the following formula will be sufficient[19]:

$$E = E_{ext} + E_{RnD} + E_{inh} \qquad \ldots(18)$$

where E_{ext} is the dose reported for external radiation (see Section 3.2)

E_{RnD} is the dose assessed from exposure to radon progeny (see Section 4.2.2), and

E_{inh} is the dose estimated from inhalation of radionuclides (see Section 5.2.1).

Often, only one or two of these three exposure pathways will be present or contribute significantly to an individual's total effective dose. In some cases, other exposure pathways may need to be included, such as doses from ingestion of radionuclides, E_{ing}, (see Section 5.2.2) or from exposure to thoron progeny, E_{TnD}, (see Section 4.2.3).

Example L

The underground uranium mine worker of Example A is exposed from three pathways: external gamma radiation (effective dose for the quarter: 0.7 mSv), inhalation of radon progeny, and inhalation of uranium ore dust. Their radon progeny dose for the quarter is assessed by summing their measured personal monitoring doses for each shift and the estimated dose from area monitoring and occupancy records for shifts when they were not individually monitored (effective quarterly dose: 0.5 mSv). The dose from inhalation of dust is calculated from periodical area monitoring to obtain a time-averaged concentration of ore dust, combined with the worker's occupancy records (effective quarterly dose: 0.2 mSv). The worker's total quarterly dose is reported as:

$$E = 0.7 + 0.5 + 0.2 = 1.4 \text{ mSv}$$

[19] The ingestion pathway does not normally contribute significant dose.

The total effective dose, *E*, can be compared with any applicable dose constraint for the assessment period, or with any other criteria established for the process of **optimisation** of protection, allowing a judgement to be made about the efficacy of the radiation protection program and the need for any further investigation or protective action. The total effective dose for the year can be compared with the occupational annual effective dose limit to assess compliance with the limit. When the assessed total dose is well below the dose limit, it may be concluded that compliance has been demonstrated to an adequate degree of confidence. If the assessed dose is within 10-20% of the annual dose limit (or any other criterion specified by the relevant regulatory authority), then an evaluation of the uncertainty in the dose assessment should be undertaken (see Annex B). The evaluation should also examine whether any particular pathway of exposure contributes more than others to the total dose, which may allow protective measures to be taken to reduce exposure from that pathway in the future.

6.2 EQUIVALENT DOSES TO ORGANS AND TISSUES FOLLOWING AN INCIDENT OR ACCIDENT

Assessment of equivalent dose to individual organs and tissues of employees is unlikely to be required during normal operation in mining and mineral processing environments. Assessment of effective dose, provided that it demonstrates compliance with the occupational effective dose limit and the principle of optimisation of protection, will usually ensure an adequate degree of protection from exposure to radiation.

An exception might occur in the case of an accident, such as a major spill of uranium product (uranium oxide concentrates) that causes significant inhalation, ingestion and skin contamination, or in the mishandling of process filters or equipment in which a high concentration of radium has built up, causing high external doses and possible intake of radium in a soluble form. Personal dosemeters for external radiation should be assessed promptly following such an event, but may not properly reflect the complexity of the exposure and TLD monitors, for example, will not provide immediate information. Personal air samplers may be saturated with process material in accidents of this kind and may also not provide useful information. Consequently, dose assessment may need to rely on reconstruction of the accident scenario and modelling of exposure. Such assessments are beyond the scope of this Safety Guide.

In addition to an accident reconstruction analysis, direct and indirect assessments of intake of radionuclides may be appropriate. If there has been significant inhalation of radioactive material and suspected retention in the lung, for example, then chest or whole-body gamma counting may provide useful information. If there has been significant ingestion of radioactive material, then urine or faecal analyses may provide data from which to estimate doses. The specialist expertise and equipment needed for these analyses will often have to be drawn from outside providers, and details are not discussed here (ICRP 1997)[20].

[20] ARPANSA and the Australian Nuclear Science and Technology Organisation maintain the capability to perform these measurements, as do a number of commercial laboratories.

When organ or tissue doses have been determined, the corresponding effective dose should be calculated using:

$$E = \sum_T w_T \, H_T \qquad \ldots(19)$$

and the appropriate **tissue weighting factors**, w_T, (see RPS 1) and this should be recorded in the employee's dose record.

Doses received by an employee while voluntarily taking part in emergency response to save lives or to bring an accident under control should be recorded separately in the employee's dose record, as required by the National Standard (ARPANSA 2002, Paragraph 13.4 of the National Standard). Separate recording is needed because such exposure, which might for example exceed the annual dose limit, should not necessarily preclude an employee from future work. Medical advice should be obtained in such circumstances as to the employee's fitness for continuing to work with exposure to radiation.

6.3 USE OF DOSE CONSTRAINTS AND OPTIMISATION OF PROTECTION

Monitoring and assessment of occupational exposure not only serves to record exposure histories and to demonstrate that dose limits are observed but can also provide information to be fed back into implementation of the radiation management plan in order to optimise protection. One of the key tools to facilitate the optimisation process is the dose constraint. A dose constraint is not a dose limit – there are no regulatory penalties for exceeding it, but it is intended to provide an upper bound for expected doses in a particular situation. It constrains the optimisation process in that efforts should be made to keep doses below the constraint value.

The history of assessed doses for a particular work area or work task may be used to derive a dose constraint for that area or task in the future. The vast majority of assessed doses are, pro rata, well below the occupational dose limit, leading to effective dose constraints that typically fall in the range of 1 to 10 mSv per year. A constraint, or 'soft' boundary, not far above typical doses, is more useful for the process of controlling exposure than a regulatory limit that far exceeds the doses that would be regarded as consistent with good practice for a given area or task.

7. Record keeping and reporting

Keeping proper records and providing appropriate reports relating to radiation monitoring and dose assessment are important aspects of a radiation protection program. Records retained by an employer should preserve sufficient information that foreseeable enquiries concerning an employee's exposure history may be readily answered. Reports to the relevant regulatory authority should contain the information required by the authority, including as a minimum the annual effective doses received by employees for whom a dose assessment is made. It is desirable that doses also be reported by calendar quarter to facilitate the maintenance of national records within the Australian National Radiation Dose Register.

7.1 RECORD KEEPING

RPS 1 requires the employer to keep records that include:

- specifications of the plans for radiation monitoring and dose assessment;
- doses assessed to have been received by employees who work directly with radiation and by other employees as required by the appropriate authority, including details of monitoring results and of dose calculation methods, as required by the appropriate authority; and
- details of **incidents** and accidents involving exposure to radiation and of corrective measures taken.
  (ARPANSA 2002, Paragraph 14.1 of the National Standard)

7.1.1 Radiation monitoring and dose assessment program

The radiation monitoring and dose assessment program should be fully documented. The specification of the program should be a controlled document within the quality management program for the operation. The description of the program should include, for each monitored area, as appropriate:

- identification of the area and its boundary
- the sources and types of radiation exposure present
- the categorisation of the area as a controlled area or supervised area
- the work groups that will occupy the area
- the types of monitoring to be undertaken
- the equipment to be used
- the planned frequency of measurement
- pointers or references to records of the calibration and performance data for the measurement equipment
- actions to be taken in response to a lost dosemeter.

The documentation should also include, for each area, work group or exposure situation, as necessary, the methods of analysis to be used to estimate dose from the monitoring results.

7.1.2 Occupational exposure dose assessment records

The record for each employee for whom individual dose assessments are made should include the following items[21], as appropriate:

- a unique identifier for the individual
- the full name, gender and date of birth of the individual
- the date on which the individual was first entered into the database
- if not included in the current database, a pointer or reference to where earlier dose records for the individual may be found
- the period (dates) for which the following items apply[22]
- the categorisation of the individual's work area (as a controlled area or supervised area) and, where appropriate, designation of the employee
- measurements of external dose and the methods of assessment, including
  - effective dose, E, or personal dose equivalent, $H_p(10)$
  - if appropriate, personal dose equivalent $H_p(0.07)$ (e.g. in the case of significant exposure to low energy X-rays or beta radiation)
- assessments of committed effective dose from radon exposure and radon progeny exposure and the methods of assessment
- assessments of internal dose and the methods of assessment, including
  - committed effective dose from intake of radionuclides
  - if appropriate, **committed equivalent dose** to organs and tissues (e.g. in the case of an accidental overexposure)
- evaluations of any anomalous results, such as unexpectedly high doses
- an explanation, when a dose has been allocated other than from the intended monitoring results, of how the dose was assessed (e.g. in the case of a lost dosemeter or a corrupted measurement)
- records of any formal declaration of pregnancy, any revocations of such declarations, and notifications of the conclusion of a pregnancy
- any doses received during an emergency or as a consequence of an accident
- the cumulative effective dose for the year to date.

To conform with the requirements of RPS 1 the records for an individual employee should be kept:

> during the working life of the employee and afterwards for not less than 30 years after the last dose assessment and at least until the employee reaches, or would have reached, the age of 75 years.
> (ARPANSA 2002, Paragraph 14.2 of the National Standard)

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[21] The record, or database, should be designed to allow multiple entries where necessary: for example, if an individual works in different areas during the assessment period.

[22] For national uniformity and collation of national records, it is desirable that assessments are made by calendar quarter.

This requirement may create difficulties for some employers, and it may be possible by agreement with the relevant authority and with the individual to pass records of former employees who are unlikely to return to the authority for future retention. In any case, the same paragraph of RPS 1 requires certain records to be passed on if an operation terminates:

> When an operation terminates, the employer shall pass to the appropriate authority the retained records of doses assessed to have been received by employees and any other records specified by the appropriate authority.

Records should be made available for inspection by duly authorised officers of the relevant regulatory authority (the 'appropriate authority' in RPS 1). Each employee should be advised in a timely manner of the assessed effective dose received during each assessment period, and an employee's exposure history should be made available to the individual on request.

7.2 REPORTING

Retention of records is important for the long-term security of individual workers' dose records and to:

- demonstrate compliance with dose limits
- provide data for analysis of dose distributions, which may lead to improvements in protection
- evaluate exposure trends, which may also take into account collective dose[23]
- facilitate optimisation of the effectiveness of monitoring procedures and programs.

Records are also often needed for litigation purposes or for workers' compensation cases, which may arise years after the actual or claimed exposure.

The relevant regulatory authority will specify the reporting requirements for each authorised operation within its jurisdiction, according to the applicable legislation. In addition, it is recommended that certain occupational dose records are provided to the Australian National Radiation Dose Register (ANRDR)[24]. The ANRDR has been established to enable workers' dose records to be tracked and recorded throughout their career, and to make accessible to workers summaries of their individual quarterly, annual and cumulative exposures. Detailed reporting procedures are available from ARPANSA. The confidentiality of personal information contained in the Register is ensured under the *Commonwealth Privacy Act, 1988* (Commonwealth of Australia 1988).

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[23] Collective dose is a measure of the total radiation exposure of a group of people, obtained by estimating the sum of their individual doses.

[24] At the time of preparation of this Safety Guide, the ANRDR was available for upload of data by uranium mining and milling operators only.

Annex A – Tables

Table A1 Dose conversion convention for radon/thoron and progeny at work[a]

[derived from ICRP 65 (ICRP 1995) and the International Basic Safety Standards (IAEA 1996a)][b]

Conversion	Conversion factor	
	Radon-222	Radon-220
from radon progeny exposure to effective dose	1.4 mSv per mJ h m^{-3}	0.48 mSv per mJ h m^{-3}
from radon exposure to effective dose[c]	3.1×10^{-6} mSv per Bq h m^{-3}	3.6×10^{-5} mSv per Bq h m^{-3}
from radon concentration to annual effective dose[d]	0.6 mSv per 100 Bq m^{-3}	7 mSv per 100 Bq m^{-3}

[a] The value of the dose conversion convention from exposure to radon-222 progeny to effective dose has been developed by the ICRP by comparison of risk from radon-222 progeny on the one hand and risk from exposure to external gamma radiation on the other. The other conversion factors are derived from this value.

[b] These values are consistent with the Mining Code.

[c] Assuming an equilibrium factor F of 0.4 for Rn-222 and 1 for Rn-220.

[d] Assuming an equilibrium factor F of 0.4 for Rn-222 and 1 for Rn-220 and a working year of 2000 hours.

Table A2 Intake-to-dose conversion factors for workers for commonly encountered radionuclides in mining and mineral processing

[from the International Basic Safety Standards (IAEA 1996a)]

Radionuclide	Type [a]	Inhalation e_{inh} (1 µm) Sv/Bq	Inhalation e_{inh} (5 µm) Sv/Bq	Ingestion f_1 [b]	Ingestion e_{ing} Sv/Bq
Uranium-238 series					
U-238	S	7.3×10^{-6}	5.7×10^{-6}	0.002[c]	7.6×10^{-9}
U-234	S	8.5×10^{-6}	6.8×10^{-6}	0.002[c]	8.3×10^{-9}
Th-234	S	7.3×10^{-9}	5.8×10^{-9}	2.0×10^{-4}	3.4×10^{-9}
Th-230	S	1.3×10^{-5}	7.2×10^{-6}	2.0×10^{-4}	8.7×10^{-8}
Ra-226	M	3.2×10^{-6}	2.2×10^{-6}	0.200	2.8×10^{-7}
Bi-214	M	1.4×10^{-8}	2.1×10^{-8}	0.050[d]	1.1×10^{-10}
Bi-210	M	8.4×10^{-8}	6.0×10^{-8}	0.050[d]	1.3×10^{-9}
Pb-214	F	2.9×10^{-9}	4.8×10^{-9}	0.200	1.4×10^{-10}
Pb-210	F	8.9×10^{-7}	1.1×10^{-6}	0.200	6.8×10^{-7}
Uranium-235 series					
U-235	S	7.7×10^{-6}	6.1×10^{-6}	0.002[c]	8.3×10^{-9}
Th-231	S	3.2×10^{-10}	4.0×10^{-10}	2.0×10^{-4}	3.4×10^{-10}
Pa-231	S	3.2×10^{-5}	1.7×10^{-5}	5.0×10^{-4} [c]	7.1×10^{-7}
Th-227	S	9.6×10^{-6}	7.6×10^{-6}	2.0×10^{-4}	8.4×10^{-9}
Ac-227	S	6.6×10^{-5}	4.7×10^{-5}	5.0×10^{-4} [d]	1.1×10^{-6}
Ra-223	M	6.9×10^{-6}	5.7×10^{-6}	0.200	1.0×10^{-7}
Pb-211	F	3.9×10^{-9}	5.6×10^{-9}	0.200	1.8×10^{-10}
Thorium-232 series					
Th-232	S	2.3×10^{-5}	1.2×10^{-5}	2.0×10^{-4}	9.2×10^{-8}
Th-228	S	3.9×10^{-5}	3.2×10^{-5}	2.0×10^{-4}	3.5×10^{-8}
Ac-228	S	1.4×10^{-8}	1.2×10^{-8}	5.0×10^{-4} [d]	4.3×10^{-10}
Ra-228	M	2.6×10^{-6}	1.7×10^{-6}	0.200	6.7×10^{-7}
Ra-224	M	2.9×10^{-6}	2.4×10^{-6}	0.200	6.5×10^{-8}
Bi-212	M	3.0×10^{-8}	3.9×10^{-8}	0.050[d]	2.6×10^{-10}
Pb-212	F	1.9×10^{-8}	3.3×10^{-8}	0.200	5.9×10^{-9}

[a] **Lung absorption type**

[b] **Gut transfer factor**

[c] Type M

[d] Type F

Annex B – Methods for Handling Uncertainties

All dose assessments will involve some degree of uncertainty. Typically, an assessment will require a calculation that combines a number of measured quantities and established or recommended parameter values. Variation in parameter values introduces systematic error[25], while measurement results carry a random or statistical uncertainty.

Non-default parameter values may be handled by 'unpacking' the assumptions made in selecting the default value and taking a step back in the level of simplification. For example, when airborne dust cannot be adequately characterised as one of the four types given in the Mining Code, individual radionuclides may be taken into account using Equation 11 and the intake-to-dose conversion factors given in the International Basic Safety Standards (IAEA 1996a).

Measurement error can be handled using standard statistical techniques. In most cases, it will be adequate to use a simple error propagation calculation. Where a quantity, Y, is derived from measured quantities, X_i, through a functional relationship f:

$$Y = f(X_1, X_2, X_3 \ldots X_n) \qquad \ldots B1$$

the measurements, x_i, (treated statistically as estimates of the X_i) lead to an estimate y of Y. Provided there are no significant correlations between the X_i, the variances of y and the x_i can be related by:

$$\sigma^2(y) = \sum_{i=1}^{n} \left(\frac{\partial f}{\partial x_i}\right)^2 \sigma^2(x_i) \qquad \ldots B2$$

Equivalently, again assuming insignificant correlations, uncertainties may be 'added in quadrature'. For quantities that are added or subtracted ($Y = \sum_i X_i$), the variances (squares of the uncertainties) are summed:

$$\sigma^2(y) = \sum_{i=1}^{n} \sigma^2(x_i) \qquad \ldots B3$$

For quantities that are combined by product or division: ($Y = \prod_i x_i$), the *relative* uncertainties are added in quadrature:

$$\frac{\sigma^2(y)}{y^2} = \sum_{i=1}^{n} \frac{\sigma^2(x_i)}{{x_i}^2} \qquad \ldots B4$$

[25] Systematic error should be avoided if possible. For example, an accurate record of hours spent in areas of known exposure would eliminate uncertainty in this parameter.

Example

A gross alpha activity concentration in air is to be estimated from measurements obtained from a personal sampler, using Equation 6 (Section 5.1.1). The alpha particle activity on the sample filter is obtained using a solid state detector and counter:

$$a_\alpha = \frac{1}{F_d}\left(\frac{N_\alpha}{t_\alpha} - \frac{N_b}{t_b}\right) \quad \text{...B5}$$

where F_d is the efficiency of the detector, N_α is the filter count in time t_α, and N_b is the blank count in time t_b.

Rewriting Equation 6 in SI units, the required concentration is:

$$c_\alpha = \frac{a_\alpha}{\dot{v}_f \times t} \quad \text{...B6}$$

Applying Equation B2 to Equation B6 then yields:

$$\sigma_{c_\alpha}^2 = \frac{1}{\left(\dot{v}_f\, t\, F_d\right)^2}\left[\frac{N_\alpha}{t_\alpha^2} + \frac{N_b}{t_b^2} + \frac{1}{\dot{v}_f^2}\left(\frac{N_\alpha}{t_\alpha} - \frac{N_b}{t_b}\right)^2 \sigma_{\dot{v}_f}^2\right] \quad \text{...B7}$$

where $\sigma_{N_\alpha}^2$ is taken to be N_α, and

$\sigma_{N_b}^2$ is taken to be N_b, and where it is assumed that the uncertainties in F_d and in all timings are negligible by comparison with those for $\dot{v}_f$, N_α and N_b.

For the purposes of this illustration, the following values are assumed:

$\dot{v}_f$ = 2 L/min, with an uncertainty of 10%
t = 4 hours (half of one shift)
F_d = 40%
N_b = 60 counts in 60 minutes (background or blank count)
N_α = 120 counts in 20 minutes (filter count).

Substituting these values into Equations B5, B6 and B7 yields:

a_α = 0.21 Bq_α, c_α = 0.43 Bq_α m^{-3}, and σ_{c_α} = 0.06 Bq_α m^{-3}.

The gross alpha activity concentration result may then be given as 0.43 ± 0.06 Bq_α m^{-3}.

Using the alternative method of summing uncertainties in quadrature:

$$\sigma_{a_\alpha}^2 = \frac{1}{F_d^2}\left[\frac{N_\alpha}{t_\alpha^2} + \frac{N_b}{t_b^2}\right] = 0.00055 \quad \text{...B8}$$

and

$$\frac{\sigma_{c_\alpha}^2}{c_\alpha^2} = \frac{\sigma_{a_\alpha}^2}{a_\alpha^2} + \frac{\sigma_{\dot{v}_b}^2}{\dot{v}_b^2} = 0.012 + 0.010 = 0.022 \quad \text{...B9}$$

from which σ_{c_α} = 0.06 Bq_α m^{-3}, as above.

Annex C – Quality Management

Part of the quality management program for radiation protection in a mining or mineral processing environment should cover the assessment of doses received by the workforce. Standard operating procedures should be developed that ensure continuing confirmation that exposures are being assessed correctly. The key stakeholders – management, employees and regulatory authorities – all require assurance that the dose assessments are valid.

Responsibility for applying the quality management system for particular assessments should be assigned to and accepted by the person conducting the relevant assessment. Quality management training should be incorporated in the training and certification programs for all such employees.

Quality management in measurement includes:

- competency of the person making the measurement
- selection of appropriate equipment and techniques for the assessment concerned
- regular calibration of equipment, including background measurements when relevant
- clear labelling of any samples taken or dosemeters collected, including chain of custody considerations to preclude loss, contamination and tampering
- prompt recording of results in a secure form
- full recording of methods and assumptions used in calculations
- backing up of documentation – paper or digital – in a safe location.

The quality management system should be consistent with the Australian Standard (Standards Australia 2008).

Annex D – Complex Assessments

The main text of this Safety Guide provides advice that is likely to be adequate for dose assessment purposes most of the time. Several simplifying assumptions have been made in order to provide default values of parameters used in dose calculations that may be used in a straightforward way in the majority of cases. However, there may be some situations in which the simplifying assumptions do not apply. In such cases, calculations should be performed at a greater level of detail. The following sections describe the most common circumstances in which this may be required.

Inhalation of radon - disequilibrium corrections

Although dose assessment for inhaled air often involves measurements of radon gas concentration, it is primarily the radon progeny in the air that deliver radiation dose to lung tissue. Radon is an inert gas and most inhaled radon atoms are exhaled again. This is not the case for radon progeny, and for air in which radon is in equilibrium with its progeny, the progeny contribute about 99% of the lung dose. From the point of view of dose assessment, radon concentration measurement serves as a surrogate for assessment of the radon progeny, which works well when the equilibrium conditions between radon and its progeny are known and stable. If there is a variable degree of disequilibrium, or if the assumed disequilibrium conditions do not apply, then the dose delivered may differ from the dose computed from radon concentration measurement.

The equilibrium condition between radon and its progeny is characterised by the 'equilibrium factor', F, defined as:

$$F = \frac{C_{EERn}}{C_{Rn}} \qquad \text{...D1}$$

where C_{Rn} is the concentration of radon in a volume of air, and

C_{EERn} is a quantity called the 'equilibrium equivalent concentration' of radon calculated from the concentration of radon progeny in the same air.

Whatever the individual concentrations of each of the radon progeny, together they can be assessed in terms of their total potential alpha energy concentration, PAEC. The equilibrium equivalent concentration of radon is the concentration that, in equilibrium with its progeny, would correspond to the same value of PAEC as that actually present. In the usual units, a PAEC of 1 J m^{-3} is 'equivalent' dosimetrically to a radon concentration of 1.8×10^8 Bq m^{-3} in equilibrium with its progeny.

Thus, the equilibrium factor may be expressed as:

$$F = 1.8 \times 10^8 \times \frac{PAEC \text{ (in J m}^{-3})}{C_{Rn} \text{ (in Bq m}^{-3})} \qquad \text{...D2}$$

where $PAEC$ is the potential alpha energy concentration of the radon progeny in a volume of air, measured in J m^{-3}, and

C_{Rn} is the radon concentration in the same air, measured in Bq m^{-3}.

For old, undisturbed air close to equilibrium, the equilibrium factor would approach unity, but this situation is seldom encountered in practice. An equilibrium factor of around 0.7-0.8 is often observed in outdoor air in non-mining environments. A factor in the region of 0.4 is reasonably representative of most indoor air environments and is taken as the default value for the purposes of dose estimation. A factor of around 0.1-0.2 might be observed in very young air, for example in highly

ventilated areas. An underground mine is a good example of a location where the equilibrium factor can be highly variable, and dependent on the ventilation. For this reason the dose assessment at minesites is based on measurements of radon progeny PAEC rather than use of radon measurements.

The International Commission on Radiological Protection recommends a dose conversion convention for exposure to radon progeny of 1.4 Sv per (J h m^{-3}) at work, assuming a nominal breathing rate of 1.2 m^3 h^{-1} (ICRP 1995). For an equilibrium factor of 0.4, this translates to:

$$e_{Rn} = 0.4 \times \frac{1.4}{1.8 \times 10^8} = 3.1 \times 10^{-9} \text{ Sv per (Bq h m}^{-3}\text{)} \qquad \text{...D3}$$

in terms of measured radon, more commonly written as 3.1 × 10^{-6} mSv per (Bq h m^{-3}), where e_{Rn} is as in Equation 2 (Section 4.2.2). In situations where F is known and is significantly different from 0.4, the corrected dose conversion is then:

$$e_{Rn} = \frac{F}{0.4} \times 3.1 \times 10^{-6} \text{ mSv per (Bq h m}^{-3}\text{)} \qquad \text{...D4}$$

The value of the equilibrium factor, F, may be assessed by making simultaneous measurements of radon concentration and PAEC. Normally, this would not be necessary as a routine assessment, but would be carried out for the initial characterisation of the working environment, followed by periodical confirmatory measurements. For measured values of F between 0.3 and 0.6, it is not considered necessary to make a correction to assessed occupational dose.

Inhalation of radon progeny – attached and unattached fractions, particle size

As radon is an inert gas, radon exists as single atoms in air. On decay, the immediate progeny (Po-218 for Rn-222, and Po-216 and almost immediately Pb-212 for Rn-220) is also created as a single atom, typically as a positive ion and therefore chemically active. The majority of polonium and lead atoms rapidly become attached to aerosol particles, and subsequent progeny exhibit similar behaviour. The unattached fraction, denoted by f_p, is the fraction of the potential alpha energy concentration of short-lived radon progeny that is not attached to the ambient aerosol. Because the deposition of particles in the lung is affected by their aerodynamic size, the 'attached fraction' of radon progeny deposits differently from the 'unattached fraction', leading to differences in lung dose.

The unattached fraction is typically of the order of 5%, while variation between 1% and 20% could lead to a decrease or increase in dose of a factor of two, respectively. The characteristic size of the particles in the attached fraction influences deposition in the lung and therefore affects lung dose, and this needs to be taken into account. Yet further complexity arises from differences related to nose breathing vs. mouth breathing.

For most mining environments, these confounding factors are considered not to require major correction to dose estimates. In exceptional cases, advice should be sought from specialist providers, as measurements of the unattached fraction and of the aerodynamic diameter or thermodynamic diameter of the attached fraction require equipment and techniques not normally available to mine-site radiation safety staff.

Inhalation of dust – non-default parameters

Estimation of effective dose from inhaled dust is based on the human respiratory tract model of the ICRP (ICRP 1994a). The model provides a mathematical representation of the respiratory tract based on anatomical, physiological and biokinetic data from Reference Man (ICRP 1975). It also provides methods for performing the complex calculations describing the deposition of inhaled radionuclides in the lung, their metabolism through body tissues, and the dose they deliver to tissues for each radioactive decay scheme and each type of radiation produced. The functional relationship between intake and effective dose depends on several parameters, including the radionuclide composition of the inhaled dust, its characteristic particle size, and its chemical form and solubility in body fluids. The default intake-to-dose conversion factors given in the Mining Code involve assumptions about all of the above factors. In cases where these assumptions do not hold, it may be necessary to refine the dose estimation by explicitly taking account of individual factors.

One quite common requirement is to estimate dose for inhaled dusts that do not adequately fit the categorisation of the Mining Code. There may be secular disequilibrium, for example, or a mixture of uranium and thorium dusts. These cases can generally be dealt with by applying Equation 9 and using the intake-to-dose conversion factors given in the International Basic Safety Standards (IAEA 1996a).

In some situations, suspended dusts may be particularly fine, such that a particle size (AMAD) of 1μm is more appropriate than the default value of 5 μm. The Mining Code and the International Basic Safety Standards (IAEA 1996a) provide intake-to-dose conversion factors that may be used in this case, but for extreme variation in AMAD values, or for nuclides not listed in the International Basic Safety Standards (IAEA 1996a), information should be sought from primary sources (ICRP 1994b).

Inhalation of fine particulates

Some industrial processes generate fine particulates when processed material is heated to a very high temperature, such as in a smelter. Airborne particles then generally have a much smaller characteristic size than is the case for mechanically generated dust, and the thermodynamic size (**AMTD**) is a more appropriate parameter than aerodynamic size (AMAD). Consequently, if inhaled, their deposition pattern in the lung is different from larger dust particles, leading to a different intake-to-dose conversion factor. Neither the Mining Code nor the International Basic Safety Standards (IAEA 1996a) is suitable for dose calculations in this case, and conversion factors should be derived from primary sources (ICRP 1994b) (see Example I).

Measurement of airborne concentrations is typically by personal or area samplers and alpha activity counting of filters. In some cases, alpha spectroscopic methods may be necessary, especially when the mixture of radionuclides is not well known or when the activity of a particular nuclide is sought. Knowledge of the particle size distribution, AMTD, is needed to obtain the correct intake-to-dose conversion factor.

Annex E – Regulatory Authorities

Where advice or assistance is required from the relevant regulatory authority for radiation protection, it may be obtained from the following officers:

COMMONWEALTH, STATE/TERRITORY	CONTACT
Commonwealth	Chief Executive Officer ARPANSA PO Box 655 Tel: (02) 95418333 Miranda NSW 1490 Fax: (02) 9541 8314 Email: info@arpansa.gov.au
Australian Capital Territory	Director Health Protection Service ACT Health Locked Bag 5005 Tel: (02) 6205 1700 Weston Creek ACT 2611 Fax: (02) 6205 1705 Email: hps@act.gov.au
New South Wales	Manager Hazardous Materials and Radiation Section Office of Environment and Heritage Department of Premier and Cabinet PO Box A290 Tel: (02) 9995 5000 Sydney South NSW 1232 Fax: (02) 9995 6603 Email: radiation@environment.nsw.gov.au
Northern Territory	Manager Radiation Protection Radiation Protection Section Department of Health GPO Box 40596 Tel: (08) 8922 7152 Casuarina NT 0811 Fax: (08) 8922 7334 Email: envirohealth@nt.gov.au
Queensland	Director, Radiation Health Unit Queensland Health PO Box 2368 Tel: (07) 3328 9987 FORTITUDE VALLEY BC QLD 4006 Fax: (07) 3328 9622 Email: radiation_health@health.qld.gov.au
South Australia	Manager Radiation Protection Environment Protection Authority GPO Box 2607 Tel: (08) 8204 2000 Adelaide SA 5001 Fax: (08) 8124 4671 Email: radiationprotection@epa.sa.gov.au
Tasmania	Senior Health Physicist Radiation Protection Unit Department of Health & Human Services GPO Box 125B Tel: (03) 6222 7256 Hobart TAS 7001 Fax: (03) 6222 7257 Email: radiation.protection@dhhs.tas.gov.au
Victoria	Team Leader, Radiation Safety Department of Health GPO Box 4541 Tel: 1300 767 469 Melbourne VIC 3001 Fax: 1300 769 274 Email: radiation.safety@health.vic.gov.au
Western Australia	Secretary Radiological Council Locked Bag 2006 PO Tel: (08) 9346 2260 Nedlands WA 6009 Fax: (08) 9381 1423 Email: radiation.health@health.wa.gov.au

Please note: This table was correct at the time of printing but is subject to change from time to time. For the most up-to-date list, the reader is advised to consult the ARPANSA web site (www.arpansa.gov.au).

For after hours emergencies only, the police will provide the appropriate emergency contact number.

Annex F – ARPANSA Radiation Protection Series Publications

ARPANSA has taken over responsibility for the administration of the former NHMRC Radiation Health Series of publications and for the codes developed under the *Environment Protection (Nuclear Codes) Act 1978*. The publications are being progressively reviewed and republished as part of the *Radiation Protection Series*. All of the Nuclear Codes have now been republished in the *Radiation Protection Series*.

All publications listed below are available in electronic format, and can be downloaded free of charge by visiting ARPANSA's website at www.arpansa.gov.au/Publications/codes/index.cfm.

Radiation Protection Series publications are available for purchase directly from ARPANSA. Further information can be obtained by telephoning ARPANSA on 1800 022 333 (freecall within Australia) or (03) 9433 2211.

RPS 1	Recommendations for Limiting Exposure to Ionizing Radiation (1995) and National Standard for Limiting Occupational Exposure to Ionizing Radiation (republished 2002)
RPS 2	Code of Practice for the Safe Transport of Radioactive Material (2008)
RPS 2.1	Safety Guide for the Safe Transport of Radioactive Material (2008)
RPS 3	Radiation Protection Standard for Maximum Exposure Levels to Radiofrequency Fields – 3 kHz to 300 GHz (2002)
RPS 4	Recommendations for the Discharge of Patients Undergoing Treatment with Radioactive Substances (2002)
RPS 5	Code of Practice and Safety Guide for Portable Density/Moisture Gauges Containing Radioactive Sources (2004)
RPS 6	National Directory for Radiation Protection, July 2011
RPS 7	Recommendations for Intervention in Emergency Situations Involving Radiation Exposure (2004)
RPS 8	Code of Practice for the Exposure of Humans to Ionizing Radiation for Medical Research Purposes (2005)
RPS 9	Code of Practice and Safety Guide for Radiation Protection and Radioactive Waste Management in Mining and Mineral Processing (2005)
RPS 9.1	Safety Guide for Monitoring, Assessing and Recording Occupational Radiation Doses in Mining and Mineral Processing (2011)
RPS 10	Code of Practice and Safety Guide for Radiation Protection in Dentistry (2005)
RPS 11	Code of Practice for the Security of Radioactive Sources (2007)
RPS 12	Radiation Protection Standard for Occupational Exposure to Ultraviolet Radiation (2006)
RPS 13	Code of Practice and Safety Guide for Safe Use of Fixed Radiation Gauges (2007)
RPS 14	Code of Practice for Radiation Protection in the Medical Applications of Ionizing Radiation (2008)
RPS 14.1	Safety Guide for Radiation Protection in Diagnostic and Interventional Radiology (2008)

RPS 14.2 Safety Guide for Radiation Protection in Nuclear Medicine (2008)

RPS 14.3 Safety Guide for Radiation Protection in Radiotherapy (2008)

RPS 15 Safety Guide for Management of Naturally Occurring Radioactive Material (NORM) (2008)

RPS 16 Safety Guide for the Predisposal Management of Radioactive Waste (2008)

RPS 17 Code of Practice and Safety Guide for Radiation Protection in Veterinary Medicine (2009)

RPS 18 Safety Guide for the Use of Radiation in Schools Part 1: Ionizing Radiation (2009)

RPS 19 Code of Practice for Radiation Protection in the Application of Ionizing Radiation by Chiropractors (2009)

RPS 20 Safety Guide for Classification of Radioactive Waste (2010)

Those publications from the NHMRC ***Radiation Health Series*** that are still current are:

RHS 9 Code of practice for protection against ionizing radiation emitted from X-ray analysis equipment (1984)

RHS 13 Code of practice for the disposal of radioactive wastes by the user (1985)

RHS 15 Code of practice for the safe use of microwave diathermy units (1985)

RHS 16 Code of practice for the safe use of short wave (radiofrequency) diathermy units (1985)

RHS 18 Code of practice for the safe handling of corpses containing radioactive materials (1986)

RHS 21 Revised statement on cabinet X-ray equipment for examination of letters, packages, baggage, freight and other articles for security, quality control and other purposes (1987)

RHS 22 Statement on enclosed X-ray equipment for special applications (1987)

RHS 24 Code of practice for the design and safe operation of non-medical irradiation facilities (1988)

RHS 25 Recommendations for ionization chamber smoke detectors for commercial and industrial fire protection systems (1988)

RHS 28 Code of practice for the safe use of sealed radioactive sources in borehole logging (1989)

RHS 30 Interim guidelines on limits of exposure to 50/60Hz electric and magnetic fields (1989)

RHS 31 Code of practice for the safe use of industrial radiography equipment (1989)

RHS 34 Safety guidelines for magnetic resonance diagnostic facilities (1991)

RHS 35 Code of practice for the near-surface disposal of radioactive waste in Australia (1992)

RHS 38 Recommended limits on radioactive contamination on surfaces in laboratories (1995)

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Glossary[26]

Absorbed dose

the energy absorbed per unit mass by matter from ionizing radiation which impinges upon it. Absorbed dose, *D*, is defined by the expression:

$$D = \frac{dE}{dm}$$

where *dE* is the mean energy imparted by ionizing radiation to matter of mass *dm*. The unit of absorbed dose is joule per kilogram ($J\ kg^{-1}$) with the special name gray (Gy).

Accident

an unintended event which causes, or has the potential to cause, employees or members of the public to be exposed to radiation from which the individual doses or collective doses received do not lie within the range of variation which is acceptable for normal operation. An accident may result from human error, equipment failure or other mishap; it may require emergency action to save life or to safeguard health, property or the environment; it requires investigation of its causes and consequences and, possibly, corrective action within the program for control of radiation; and it may require remedial action to mitigate its consequences.

Activity

Activity, *A*, is a measure of the amount of a radioactive material given by:

$$A = \frac{dN}{dt}$$

where *dN* is the expectation value of the number of spontaneous nuclear transitions which take place in the time interval *dt*. The unit of activity is s^{-1} with the special name Becquerel (Bq).

Activity median aerodynamic diameter (AMAD)

a measure of the characteristic particle size of airborne dust. Activity median aerodynamic diameter, AMAD, is the diameter of a unit density sphere with the same terminal velocity in air as that of an aerosol particle whose activity is the median for the entire aerosol. [Typically, in the dosimetry of inhaled radioactive particles, AMAD is an appropriate particle size parameter above about 0.5 µm.]

Activity median thermodynamic diameter (AMTD)

a measure of the characteristic particle size of very fine particulates. Activity median thermodynamic diameter, AMTD, is the diameter of a unit density sphere with the same thermodynamic diffusion properties as that of an aerosol particle whose activity is the median for the entire aerosol. [Typically, in the dosimetry of inhaled radioactive particles, AMTD is an appropriate particle size parameter below about 0.5 µm.]

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[26] Where available, these definitions are taken from RPS 1; otherwise they are based on other publications in the Radiation Protection Series, the International Basic Safety Standards or publications of the International Commission on Radiological Protection, as appropriate.

Alpha particle

a charged particle, consisting of two protons and two neutrons, emitted by the nucleus of a radionuclide during radioactive decay (α-decay).

ANRDR

the Australian National Radiation Dose Register.

Beta particle

an electron or positron emitted by the nucleus of a radionuclide during radioactive decay (β-decay).

Committed effective dose

Committed effective dose, E(τ), is the effective dose which an individual is committed to receive from an intake of radioactive material over the period subsequent to that intake and is given by the expression:

$$E(\tau) = \sum_T w_T H_T(\tau)$$

where τ is the period over which the integral of the equivalent dose rate for organ or tissue T is made to obtain the committed equivalent dose $H_T(\tau)$ (see below).

For adults, an integration period of 50 years is assumed; for children, the integration period is taken to age 70.

Committed equivalent dose

Committed equivalent dose, $H_T(\tau)$, is the equivalent dose which would be received by an organ or tissue from an intake of radioactive material over the period subsequent to that intake and is given by the expression:

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

where $\dot{H}_T(t)$ is the *relevant* equivalent dose rate in organ or tissue T at time t, and

τ is the period over *which* the integration is made.

For adults, an integration period of 50 years is assumed; for children, the integration period is taken to age 70.

Controlled area

an area to which access is subject to control and in which employees are required to follow specific procedures aimed at controlling exposure to radiation.

Dose

a generic term which may mean *inter alia* **absorbed dose**, equivalent dose or effective dose depending on context.

Dose constraint

a prospective restriction on anticipated dose, primarily intended to be used to discard undesirable options in an optimisation calculation; in occupational exposure, a dose constraint may be used to restrict the options considered in the design of the working environment for a particular category of employee.

Dose conversion convention

a conversion factor between exposure to radon or to radon progeny and effective dose; it allows these exposures to be assessed using the quantity effective dose and thereby added to effective dose from other exposure pathways for the purposes of estimating total effective dose and comparison with the occupational effective dose limit (see Table A1).

Effective dose

Effective dose, E, is the sum of weighted equivalent doses in all organs and tissues of the body. It is given by the expression:

$$E = \sum_T w_T H_T$$

where H_T is the equivalent dose in organ or tissue T and w_T is the weighting factor for that organ or tissue (see RPS 1).

The unit of effective dose is the same as for equivalent dose, J kg^{-1}, with the special name sievert (Sv).

Employee

a person who works for an employer within an operation.

Employer

an operator who or which engages people to work within an operation; the term employer includes a self-employed person.

Equilibrium factor

The equilibrium factor, F, for radon in air is the ratio of the equilibrium equivalent concentration of radon to the actual radon concentration, where the equilibrium equivalent concentration is the activity concentration of radon in equilibrium with its short-lived progeny having the same potential alpha energy concentration as the actual non-equilibrium mixture.

Equivalent dose

Equivalent dose, H, is a weighted dose in an organ or tissue, with the **radiation weighting factor(s)** determined by the type and energy of the radiation to which the organ or tissue is exposed. The equivalent dose H_T in organ or tissue T is given by the expression:

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

where $D_{T,R}$ is the absorbed dose averaged over the organ or tissue T due to radiation R and

w_R is the radiation weighting factor for that radiation (see RPS 1).

The unit of equivalent dose is the same as for absorbed dose, J kg^{-1}, with the special name sievert (Sv).

Excluded exposure

in the context of occupational exposure, the component of exposure which arises from natural background radiation, provided that any relevant action level, or levels, for the workplace are not exceeded and that the relevant regulatory authority does not prohibit its exclusion.

Exposure

either: the circumstance of being exposed to radiation,

or: a defined dosimetric quantity now largely obsolete except for radon exposure and radon progeny exposure.

Gamma ray

ionizing electromagnetic radiation emitted by a radionuclide during radioactive decay or during a nuclear (isomeric) transition.

Gut transfer factor

The gut transfer factor, f_1, is the proportion of an ingested radionuclide that is transferred to body fluids in the gut.

Incident

an event which causes, or has the potential to cause, abnormal exposure of employees or of members of the public, and which requires investigation of its causes and consequences and may require corrective action within the program for control of radiation, but which is not of such scale as to be classified as an accident.

Ionizing radiation

radiation which is capable of causing ionization, either directly (for example: for radiation in the form of gamma rays and charged particles) or, indirectly (for example: for radiation in the form of neutrons).

Lung absorption type

The lung absorption type (F, M or S) of a radioactive material is an index that characterises the speed of absorption of the material (fast, medium or slow) in the ICRP modelling of the respiratory tract (ICRP 1994a; ICRP 1994b).

Occupational exposure

exposure of a person to radiation which occurs in the course of that person's work and which is not excluded exposure.

Operation

an instance of a practice; a particular human activity which may result in exposure to ionizing radiation and to which a **program of radiation protection** applies.

Operator

any person or entity responsible for an operation which may lead to exposure to ionizing radiation.

Optimisation

the process of maximising the net benefit arising from human activities which lead to exposure to radiation.

Personal dose equivalent

a measure of dose to the body used for the purposes of metrology, and the quantity typically reported by providers of personal dosimetry services; in many situations, personal dose equivalent may be taken to be numerically equal to equivalent dose for the purposes of demonstrating compliance with radiation protection standards.

Personal dose equivalent, $H_p(d)$, is the dose equivalent[27] in soft tissue below a specified point on the body at an appropriate depth, d. The relevant depths are generally d = 10mm for strongly penetrating radiation, such as X -rays and gamma rays, and d = 0.07mm for weakly penetrating radiation, such as beta radiation.

Potential alpha energy

Potential alpha energy is the total alpha energy ultimately emitted during the decay of radon-222 progeny through its decay chain, up to but not including lead-210, and of radon-220 progeny through its decay chain up to but not including lead-208.

Potential alpha energy concentration

Potential alpha energy concentration, PAEC, is the concentration of potential alpha energy in air containing radon progeny; it is usually measured in µJ m^{-3}.

Program of radiation protection

an instance of a system of radiation protection, designed for a particular operation.

Radiation

electromagnetic waves or quanta, and atomic or sub-atomic particles, propagated through space or through a material medium.

Radiation weighting factor

a radiation weighting factor, w_R, is a modifying factor which is applied to an organ or tissue absorbed dose to yield equivalent dose and which depends o n the type and energy of the radiation to which the organ or tissue is exposed (see RPS 1).

Radioactive decay

the spontaneous transformation of the nucleus of an atom into another state, accompanied by the emission of radiation; for a quantity of such atoms, the expectation value of the number of atoms present decreases exponentially with time.

Radioactive material

material which spontaneously emits ionizing radiation as a consequence of radioactive decay.

Radionuclide

a species of atomic nucleus which undergoes radioactive decay; a radionuclide is identified either by its elemental symbol with the mass number of the nuclide as a superscript (eg: ^{235}U) or, as in this Safety Guide, by its elemental name followed by the mass number (eg: uranium-235).

Radon

used generically, all isotopes of the element radon, having atomic number 86, but typically used to refer to the radioactive gas radon-222.

Radon exposure

a measure of exposure to radon-222 expressed as the cumulative product of radon concentration in inhaled air and time, in units of Bq h m^{-3}.

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[27] 'Dose equivalent' is a quantity used by the International Commission on Radiation Units and Measurements (ICRU) in defining operational dosimetry quantities such as personal dose equivalent. It has been superseded for radiation protection purposes by 'equivalent dose'. These terms are explained in ICRU Report 51 (ICRU 1993).

Radon progeny

the short-lived products of the radioactive decay of radon, namely polonium-218, lead-214, bismuth-214, and polonium-214.

Radon progeny exposure

a measure of exposure to radon progeny expressed as the cumulative product of potential alpha energy concentration in inhaled air and time, in units of mJ h m^{-3}. (The historical unit of radon progeny exposure, working level month (WLM), is equal to 3.54 mJ h m^{-3}.)

Relevant regulatory authority

the radiation protection authority or authorities designated, or otherwise recognised, for regulatory purposes in connection with protection and safety in mining and mineral processing. A list of radiation protection authorities in Australia is included as Annex E of this Safety Guide.

Secular equilibrium

the condition in which successive members of a decay chain have the same activity.

Supervised area

an area in which working conditions are kept under review but in which special procedures to control exposure to radiation are not normally necessary.

Thoron

the radioactive gas radon-220.

Thoron progeny

the short-lived products of the radioactive decay of thoron, namely polonium-216, lead-212, bismuth-212, polonium-212, and thallium-208.

Tissue weighting factor

a tissue weighting factor, w_T, is a modifying factor which is applied to an organ or tissue equivalent dose to yield a component of effective dose and which depends on the organ or tissue irradiated (see RPS 1).

Unattached fraction

the fraction, f_p, of the potential alpha energy concentration of short-lived radon progeny that is not attached to the ambient aerosol.

X-ray

ionizing electromagnetic radiation emitted during the transition of an atomic electron to a lower energy state or during the rapid deceleration of a charged particle.

Contributors to Drafting and Review

Mr Cameron Jeffries	Scientist, Radiation Protection Division, Environment Protection Authority, South Australia
Mr Ivan Fetwadjieff	Senior Scientific Officer, Radiation Department of Mines and Petroleum Resources Safety Division, Western Australia
Mr Frank Harris	Manager Radiation Governance, BHP Billiton Uranium CSG, South Australia
Ms Sharon Paulka	Paulka Radiation & Environment Pty Ltd, South Australia
Dr Ches Mason	Consultant, Victoria
Dr Rick Tinker	Environmental & Radiation Health Branch, ARPANSA, Victoria
Dr Paul Martin	Environmental & Radiation Health Branch, ARPANSA, Victoria

Note: Organisational relationships listed reflect the status at time document was prepared.

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