Government of **Western Australia**
Department of **Mines, Industry Regulation and Safety**

WorkSafe

Managing naturally occurring radioactive material (NORM) in mining and mineral processing – Guide

NORM-V Dose assessment

February 2023

Contents

1. Purpose

This document provides guidance on the methods for the assessment of radiation doses received by mine workers and members of the general public from exposure to naturally occurring radioactive materials (NORM).

The first edition of this Guide updated previous guidance (NORM-5) by incorporating the revised dose coefficients (DC) for members of the ^{238}U, ^{235}U and ^{232}Th decay series as published in the International Commission for Radiological Protection publications *Occupational intakes of radionuclides: Part 3 ICRP publication 137* (ICRP-137) and *Part 4 ICRP publication 141* (ICRP-141).

This second edition aligns the Guide with the Work Health and Safety (Mines) Regulations 2022 (WHS Mines Regulations); specifically Part 10.2, Division 3, Subdivision 3B – Radiation in Mines.

2. Scope

The WHS Mines Regulations embrace a risk-based approach to regulating the potential radiation doses of Western Australian (WA) mine workers arising from their exposure to radionuclides of natural origin (NOR) and pursue, where practicable, national uniformity of regulation of occupational radiation exposures by adoption of the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) *Code of practice and safety guide for radiation protection and radioactive waste management in mining and mineral processing (2005): Radiation protection series No. 9* (RPS 9).

The WorkSafe Commissioner is the regulator under the *Work Health and Safety Act 2020* (WHS Act). As defined in regulation 641I(3), a reference to the regulatory authority in RPS-9 is a reference to the WorkSafe Commissioner as the regulator.

Regulations 641K and 641L set out the scope of the WHS Mines Regulations in relation to radioactive material

Regulation 641K defines material as **radioactive material** if the activity concentration in the material exceeds 1 $Bq.g^{-1}$, and the material either:

- exhibits radioactivity
- emits ionising radiation or particles, or
- contains radionuclides of natural origin.

The most significant NORs to the WA mining industry are:

- members of the decay series of:
  - ^{232}Th
  - ^{238}U
  - ^{235}U
- ^{40}K
- ^{87}Rb.

The activity concentration criteria can be met by a single NOR or a combination of the contributions from two or more NORs, and:

- if secular equilibrium is assumed among all members of the ^{232}Th, ^{238}U or ^{235}U decay series, only the activity concentration of the head-of-chain is considered
- if secular equilibrium has been disturbed, the contribution by individual NORs in a decay series requires evaluation against the 1 $Bq.g^{-1}$ criteria

Section 7.8 of this Guide contains information on secular equilibrium.

Regulation 641L defines the parameters that must be met in order for a mine to be considered a relevant mine**,** namely:

- the activity concentration criteria of $1 Bq.g^{-1}$ must be met or exceeded
- mining operations that encounter, process, use, handle, store or dispose of radioactive minerals or materials, and where annual doses to:

– workers exceed 1 millisievert (mSv) (r. 641L(b)(i)

– members of the public exceed 0.5 mSv (r. 641L(b)(ii).

Division 3, Subdivision 3B is not applicable to mining operations that do not encounter radioactive minerals or materials, and are able to demonstrate, with verifiable data, that annual doses to workers are consistently less than 1 mSv, and annual doses to members of the public are consistently less than 0.5 mSv.

Note that the onus is on the mine operator to demonstrate that Division 3, Subdivision 3B does not apply to their mining operation(s).

3. Legislative status

In accordance with regulation 641Q(2) the regulator has approved this Guide as an "approved procedure".

Until otherwise directed by the regulator, the dose coefficients, method to determine dose conversion factors and the protocols for calculation of doses in this Guid**e** are to be applied to the assessment of doses of radiation from naturally occurring radioactive materials in the Western Australian mining industry as stipulated under Regulation 641Q(1)(c).

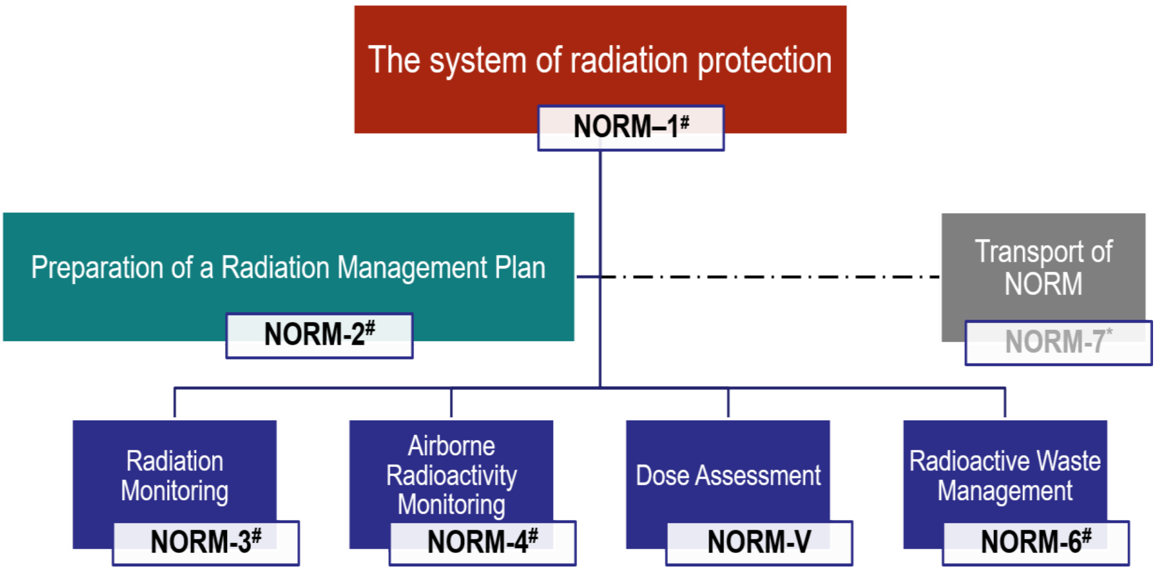
For the purpose of the assessment of doses of radiation, this Guide has precedence over RPS 9.

Further, it is an expectation of the regulator that **relevant mines** will make commitments in their radiation management plan to apply relevant sections of this Guide for the evaluation and control of exposures to NORM.

4. Relationship to other NORM guides

The diagram below illustrates the relationship of NORM-V with the suite of NORM Guides and Guidelines.

Note: Other Guides will be re-numbered using a Roman numeral format as they are published.



under review at time of publication of NORM-V

* under consideration for development

5. Introduction

This Guide details the dose assessment procedures to be used when assessing radiation exposure from NORM in WA **relevant mines**, and is based on the recommendations in relevant documents produced by the International Commission on Radiological Protection (ICRP), International Atomic Energy Association (IAEA) and ARPANSA.

All exposure pathways must be considered when assessing potential radiation doses arising from exposure to NORM, including:

- external radiation exposure
- inhalation of dust containing NORs
- inhalation of radon (both ^{222}Rn and ^{220}Rn) and its decay products
- ingestion of drinking water
- ingestion of food, dust and soil.

5.1. Justification for not considering a pathway

If a particular pathway is considered as making an insignificant contribution to dose, or is thought to be not relevant to the operations conducted by a **relevant mine**, justification must be provided for it to be not considered in the dose assessment process.

5.2. Pre-operational monitoring program

In accordance with regulation 641M, the regulator is to be provided with a proposed program for assessing the radiological characteristics of a site under consideration for the location of a mining operation.

The monitoring program is to be approved by the regulator prior to the commencement of mining operations. The approved program is referenced as a **pre-operational monitoring program**.

The results of the approved pre-operational monitoring program (referred to as baseline data) are to be provided to the regulator before the commencement of a mining operation.

Guidance as to the collection of baseline data is provided in *Guideline: NORM-3.1 Pre-operational monitoring requirements* (NORM-3.1).

The results of the baseline data are used to establish the natural background levels for each pathway.

In the absence of baseline data, the values proposed to be used as natural background are to be established from appropriate practically available information, and are to be approved by the regulator on a case-by-case basis.

5.3. Statistical derivation of natural background

To determine representative natural background radiation levels, sufficient sample locations shall be selected to obtain a statistically significant data base (refer to NORM-3.1 for guidance).

Unless sufficient data is collected to empirically prove otherwise, the distribution of data points to determine natural background is assumed to be log-normal.

Accordingly, the value to be used in calculating the contribution from natural background will be the geometric mean (GM) of the collected data points.

When a statement is made about the GM of the data it shall be accompanied by the:

- geometric standard deviation (GSD), and
- 95 percent confidence interval (GM x GSD $^{1.645}$).

5.4. Contribution from natural background sources

As per Note 2 to Schedule 1 in RPS 9, the contributions to dose due to natural background sources of radiation are not to be considered when assessing the dose received by a worker or a member of the public.

The Note also reflects that exceptions to the exclusion of contributions from natural sources can be made when "specifically identified by the appropriate authority as requiring control through the implementation of a program of radiation protection".

The regulator has determined that the dose arising from exposure via inhalation of ^{222}Rn or ^{220}Rn and their decay products is to include the contribution from natural background sources.

The natural variation in concentrations of these sources of exposure is such that it is difficult to determine, with any precision, a "typical" background contribution to dose.

As is discussed in section 8.4 of this Guide, dose calculations from exposure to ^{222}Rn and ^{220}Rn and their decay products shall include background, and be accompanied by a note indicating an estimated concentration of the applicable source of exposure (^{222}Rn or ^{220}Rn or their decay products) and any assumptions made in calculating the dose estimate.

Where the contribution from natural background to committed effective dose is known and has been deducted from the dose arising from any exposure pathway:

- the contribution from natural background should be stated for each exposure pathway
- if the natural background differs between similar exposure groups (SEGs) or critical groups, the value attributed to natural background for each group is required to be stated
- a statement to the effect that natural background has been deducted from worker and member of the public dose assessments shall accompany records of dose assessments.

Note. The IAEA Basic Safety Standards have replaced the term 'critical group' with 'representative person'. However, at time of publication, Australia has not yet adopted the revised IAEA nomenclature.

6. Assessing external radiation exposure

External radiation exposures are determined from either personal monitoring devices (as approved by the WA Radiological Council), gamma radiation survey results or a combination of both. The methods used are dependent on the person or group that is being assessed. **Designated workers (DWs)**

Regulation 641S(1)(a) defines a **designated worker** as a person who works under conditions that result in an effective dose greater than 5 mSv per year.

Designated workers are monitored more intensively than non-designated workers (including, where appropriate, by personal monitoring) and their doses are assessed individually

Refer to ARPANSA *Safety guide for monitoring, assessing and recording occupational radiation doses in mining and mineral processing*: *Radiation protection series No. 9.1*, 2011 (RPS 9.1).

All designated workers must be included in a radiation dose monitoring program.

6.1. Pregnant workers

In accordance with regulation 641T, once a worker learns that they are pregnant, they are not permitted to be a designated worker and a dose limit not exceeding 1 mSv shall apply to the remainder of the pregnancy.

6.2. Designated workers: monitoring of external dose

Monitoring may be conducted via passive devices such as an optically stimulated luminescence (OSL) dosimeter or thermo-luminescent dosimeter (TLD), or an active type, such as a personal electronic dosimeter (PED).

An external supplier typically provides personal passive devices (OSLs or TLDs) and conducts the analysis of the devices to determine the external dose. A list of personal monitoring services approved for use in WA is available from the WA Radiological Council.

A substantial time delay (up to three months) may occur between the end of a monitoring period and the return of the dose results from the monitoring service. As such, PEDs can be used in conjunction with passive monitors (not as a substitute):

- PEDs can be read out immediately following a shift or a specific task, and some PEDs have an additional advantage of being able to sound an alarm in the event that a preset threshold dose is measured.
- PEDs require regular calibration, in accordance with protocols established in the approved radiation management plan applicable to the operation.

A comparison to the worker effective radiation dose limit should be provided. Regulation 641P(1) defers to the occupational dose limits published in Schedule 1 of RPS 9, namely an effective dose of 20 mSv per year, averaged over a period of five consecutive years, with the further provision that the effective dose must not exceed 50 mSv in any single year.

Both the annual result and the previous five years cumulative result should be presented against the applicable dose limits.

The results of external gamma radiation survey results can be used to verify the assessment of likely exposures.

6.3. Non-designated workers

As per RPS 9.1, “non-designated workers will be monitored less intensively than designated workers; 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” or similar exposure group (SEG).

Therefore, a cross-section of non-designated workers shall be included in a monitoring program, with monitoring conducted on a frequency determined by the radiation exposure risk.

The cross section should include workers from different SEGs with sufficient data collected for each SEG to be representative of the general exposure scenarios, and:

- all members of a SEG must have similar exposure to NORMs, but may have very different roles or occupations
- the validity of the dose assessment methodology should be periodically confirmed.

6.4. Non-designated workers: monitoring of external dose

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

The monitors may be of a passive type, typically an OSL or TLD or an active type, such as an appropriately calibrated survey instrument or PED.

The results (above natural background) from each SEG are averaged for each monitoring period and then added to give an annual dose for that SEG. All members of that SEG are then deemed to have that exposure. Refer to Example 2 in Appendix 3 for calculations of dose to a SEG.

A comparison to the designated worker effective radiation dose threshold level of 5 mSv per year should be provided. As per section 6.6 below, the minimum detectable limit (MDL) should be provided and substituted into the dose assessment calculation when the results of personal monitoring is reported as “<MDL”, and a notation provided that the MDL dose was not necessarily received, and that it represents a potential maximum value.

6.5. Other workers, critical groups and members of the general public

Regulation 641P(2) requires that the site senior executive of the mining operation must ensure that a member of the public does not receive a dose of radiation as a consequence of the operation of the mine (i.e. levels above background levels) exceeding the public effective dose limit.

In accordance with regulation 641O(2) the regulator may, as part of the approval process of a radioactive waste management plan, establish an authorised limit for radiation exposure, contamination level, or airborne or waterborne discharge for a mining operation. This is an important control to be considered if critical groups or members of the public are potentially exposed to sources of radiation from more than one mining operation.

Compliance with any discharge limits that may be imposed under the *Radiation Safety Act 1975* is also required.

The external gamma radiation exposures for other workers and the members of the general public are estimated from the results of regular radiation surveys of specific areas and the likely hours that a person will be in those areas.

A comparison to the annual dose limit for members of the public, or a regulator imposed authorised limit (if applicable) should be provided.

6.6. Calculation of external dose

The calculation of external dose is dependent upon the monitoring method, and assumptions in relation to membership of a SEG and of the hours of exposure.

6.7. Designated workers

The external radiation dose component is obtained directly from the results of the personal monitoring program. Further information is available in the Guide *NORM-3.2 Monitoring NORM: Operational monitoring requirements* (NORM 3.2).

The dose results (above background) for each monitoring period are added together to give an annual dose for that worker and no further adjustments are required.

The results of external gamma radiation survey results can be used to verify the assessment.

In the absence of sufficient personal monitoring data to establish a statistically valid assessment of dose, the assessment may be supplemented by time and motion studies as applied to non-designated workers (see Section 6.4.2).

6.8. Non-designated workers, critical groups and members of the public

To calculate the external radiation exposures, the average gamma dose rate (above natural background levels) from the last survey is multiplied by the actual hours worked by the worker, if the data is available.

If the actual working hours are not available, an estimate can be made from the applicable shift roster. If an estimate is made, it should be noted that:

6.9. if a worker is at work for 40 hours per week, a default value of 2000 hours exposure per year can be applied

6.10. calculations of exposure hours must be made for workers that have rosters significantly different to 8 hours per day for 5 days per week. For example, a person on a 2 week on 1 week off roster with 12hr shifts will work approximately 2912 hours per year. When calculating work hours:

- 6.10.1. meal breaks and shower and change time should be included as work time
- 6.10.2. an allowance shall be made for periods of leave, where no exposure is expected to occur as a result of employment. The deduction for leave should be specified in the dose assessment
- 6.10.3. provide shift rosters as part of the assessment.

6.11. the default hours for members of the public is 8760 hours per year, based on 24 hour exposure 365 days a year. If a member of the public is at an area less than continuously then justification and the calculation must be provided.

The GM gamma dose rate from the most recent survey in mSv per hour (mSv/h) is to be used. The calculation methods are illustrated in Examples 3 and 4 in Appendix 3.

6.9. External gamma radiation readings

As detailed in NORM-3.1 (Appendix B about survey instruments and data interpretation), there is a wide range of radiation monitoring instruments used to measure the gamma dose rate with levels expressed differently e.g. μSv/h, micrograys per hour (μGy/h) and millirems per hour mR/h).

Typically, the following approximate interpretation should be used for external gamma readings arising from NORM:

100 roentgens = 100 roentgen equivalent man (rem) = 1 gray = 1 sievert

There are several equivalencies that could theoretically be used for the conversion. The use of any of those conversion factors is not approved and should be avoided in the assessment of external exposures to radiation of workers and members of the general public.

6.10. Minimum detection limit (MDL)

Similar to all scientific analytical techniques, personal monitors will have a level below which the uncertainty of the analytical result exceeds a required level of accuracy. This value is called the minimum detection limit (MDL).

When a personal badge result returns a value of less than the MDL (<MDL) this should not be interpreted as the wearer was not exposed to gamma radiation as:

- the <MDL result indicates that the exposure was below the threshold at which the dose could be reported with statistical certainty
- the resultant dose is *not* zero.

Whenever a personal badge result returns a reading of less than the MDL, the MDL shall be substituted into the dose assessment calculation.

The value of the MDL and the uncertainty (error) of the result should be provided when presenting the results of a dose assessment (e.g. MDL=0.02 mSv quarterly dose 1.50 ± 0.02mSv; annual dose 5.22 ± 0.08mSv)

7. Assessing internal radiation exposure via inhalation of dusts

The effective radiation dose received following inhalation of particulates is dependent on their:

6.11. size, expressed as activity median aerodynamic diameter (AMAD) in microns (µm)

6.12. activity

6.13. solubility.

All three parameters must be considered when assessing the radiation dose from inhaled particles.

7.1. Impact of particle size

Airborne particulate matter present on mining operations can have aerodynamic equivalent diameters (AEDs) that exceed 100 µm to ultrafine particles in the sub-micron range.

The effects that a particle will have on the body is highly dependent on its chemical properties and where it is deposited in the respiratory system, or whether it can pass into blood and be preferentially deposited in target organs. For this reason the aerodynamic behaviour of the inhaled particles must be evaluated or, in the absence of a valid sampling regime, estimated.

The aerodynamic behaviour depends chiefly on inertial impaction and sedimentation of particles and is represented by the median aerodynamic diameter (MAD). The MAD is the value of aerodynamic diameter for which:

- 50% of some quantity in a given aerosol is associated with particles smaller than the MAD
- 50% of the quantity is associated with particles larger than the MAD.

In radiological applications, MAD is measured in µm.

7.2. Mass median aerodynamic diameter (MMAD)

MMAD is indicative of the distribution of particles by their mass, where 50% of the mass of the particles is greater than the MMAD and 50% is less than the MMAD.

In radiological applications, MMAD can be a useful reference point for determining the activity concentration of the median particle size, by dividing the AMAD (see Section 7.3) by the MMAD.

7.3. AMAD, dose coefficients and derivation of dose conversion factors for dusts

The activity median aerodynamic diameter (AMAD) defines the distribution of the activity of particles, where 50% of the activity of the particles is greater than the AMAD and 50% is less than the AMAD.

The AMAD is then applied to determine the dose coefficient (DC) for a specific NOR. Internationally accepted DC's are published by the ICRP and IAEA.

The latest publications of the ICRP provide the DCs for inhalation of NORs (see Appendix 5: References). Unless otherwise specified, the ICRP Occupational Intake of Radionuclides *Data Viewer* has been used as the source of DCs used in this Guide.

This data is summarised in Tables 1-10, for the NORs in the ^{238}U, ^{235}U and ^{232}Th decay chains.

7.3.1. ICRP-130 confirms that a default AMAD of 5 µm is to be used for occupational exposures of mechanically generated dusts

- however, once operations are stabilised then the AMAD must be determined and the closest applicable AMAD applied.

7.3.2. ICRP-66 confirms that an AMAD of 1 µm is to be used for environmental exposures to members of the general public from dusts.

When calculating the impact of a mixture of NORs, such as occurs in mineral dusts, where the NORs are in secular equilibrium, the DCs are treated in combination, to derive a dose conversion factor (DCF):

- the calculation of the applicable DCF for each decay chain, by particle size is provided at the end of each of Tables 1 to 10
- the DCFs for mineral dusts with AMAD sizes of 1, 3, 5, 10 and 20 μm containing thorium and uranium in different mass ratios are provided in Table 11.

For convenience, when conducting a preliminary assessment, the DCFs for typical minerals and materials encountered in the WA mining and mineral processing industry are provided for the default AMAD of 5 μm in Table 12.

Please note that data in this Table is for preliminary assessment only, and regular analysis of the thorium/uranium ratio (Th:U) (by mass) and the AMAD value is required to ensure that a site specific DCF is established.

Note that is recognised that AMAD and Th:U ratio may vary across a mining operation, and reflect the mineral being processed, and the method by which processing occurs. Therefore it is likely that several different Th:U ratios and AMAD's will be applicable across a complex mining operation.

Where the values differ from ones provided in Table 12, the DCs values in Table 11 shall be used.

7.4. AMAD and dose coefficients for fumes

The DCs and DCFs for mechanically-generated dusts, such as those liberated from crushing and screening operations are not applicable when ultra-fine particles less than 1 μm are generated, such as fumes. Examples can include collected particles in electrostatic precipitators or furnace exhaust stacks.

Certain NORs such as ^{210}Pb or ^{210}Po may be present in these very fine dusts in the range 0.03 to 3.0 μm in these situations. Table 13 contains the DCs for dusts containing these NORs with sizes smaller than 1 μm.

7.5. Additional AMAD dose coefficients for lithium operations

Potassium-40 (^{40}K) and Rubidium-87 (^{87}Rb) are present in spodumene and lepidolite ores common in the production of lithium concentrate.

In addition to the dose received from uranium/thorium an assessment of these two NORs may be required.

The values from the IAEA *Radiation protection and safety of radiation sources: International basic safety standards* (IAEA 2014) are provided in Table 14.

Note that the latest ICRP Publications do not contain the DCs for the inhalation of these NORs.

7.6. Additional AMAD DCs for rare earths operations

Minerals such as monazite contain lanthanides including samarium. If downstream processing for rare earth minerals occurs then an assessment of potential internal dose due to the inhalation of dust containing a natural isotope of samarium, ^{147}Sm (which makes up 15% of samarium) may be required.

The DCs for ^{147}Sm extracted from IAEA 2014 are provided in Table 14.

7.7. Particle sizing and AMAD determination

Further information and techniques for determining particle size distribution, MMAD and AMAD are covered in the revised version of the Guide *NORM-3.5: Monitoring NORM - measurement of particle size*.

7.8. Estimation of dust activity concentrations and secular equilibrium

Secular equilibrium occurs when a radioactive nuclide is decaying at the same rate at which it is being produced. Therefore the activity of each progeny will be the same as the parent and the activity of each NOR will not need to be measured to determine the activity of the entire material. Secular equilibrium can only occur when the parent has a long half-life compared to the progeny and when no factors influence this relationship.

Generally when only gravimetric or magnetic separation occurs, for example when concentrating zircon, ilmenite or tantalum, then secular equilibrium is maintained.

Secular equilibrium is unlikely when minerals are subject to:

7.8.1. any chemical processing of the material, such as leaching or adding flotation agents to the process

7.8.2. any thermal processing of the material. Due to the variety of different materials and methods used in their treatment, it is impossible to establish a universal 'cut-off' point for the temperature at which some NORs can volatilise and disrupt the equilibrium; 250-300°C is suggested as a general guide at which additional analysis of the material may be required

7.8.3. when mineralogy and processing methods result in certain minerals being greater or less prevalent. For example in the heavy mineral sands industry the mineral monazite is softer and finer than the other processed minerals, which typically results in the concentration of this mineral in airborne dust being up to 30 times higher than in the bulk concentrate that is being processed

7.8.4. when scale or sludge precipitates or deposits, for example:

7.8.4.1. cleaning processing vessels in titanium pigment plants (^{228}Ra and ^{226}Ra scales) and in zirconia production facilities (^{210}Pb films and ^{210}Po dust)

7.8.4.2. processing and storage of natural gas and replacement of stack filters at different smelters (^{210}Pb and ^{210}Po films)

7.8.4.3. oil exploration and production (^{226}Ra and ^{228}Ra scales)

7.8.4.4. geothermal energy generation and hydraulic fracturing (^{226}Ra and ^{228}Ra in scales and process water).

7.8.5. Tables 1 – 10 list the values for individual NORs .

In cases where the estimation of potential exposure of both workers and members of the general public needs to be carried out it is essential to estimate the potential dust activity concentrations in the workplace and in the environment based on the content of NORs in the ore, mineral or concentrate.

The correct conversion coefficients for such calculations are provided in Table 15.

7.9. Dose assessment

To assess the amount of radioactive material inhaled by an individual, an appropriate monitoring program that is representative of the exposure to dust throughout the year is required.

The monitoring program needs to be reflective of the activities undertaken by the typical representative worker in a SEG, and be sufficient in number to provide statistical certainty.

Further information is provided in the Guide NORM-4 dealing with the different aspects of airborne radioactivity monitoring.

In accordance with regulation 641Q(1)(b) of the WHS Mines Regulations, no allowance is to be made for any respiratory protection factors in the assessment of the internal doses due to the inhalation of dust, unless:

- a comprehensive and an auditable respiratory protection program has been established at the site in accordance with AS/NZS 1716:2012 *Respiratory protective devices* and AS/NZS 1715:2009 *Selection, use and maintenance of respiratory protective equipment*
- the respiratory protection program has been fully assessed by DMIRS and a written authorisation has been issued by the regulator to use a specific respiratory protection factor for specific respiratory protection devices in all or some areas of the site
- in the event that an approved respiratory protection factor is applied to the estimation of doses, the dose prior to application of the protection factor shall be included in reports to the regulator, as well as the dose after the protection factor has been taken into account.

To determine the annual dose, the annual intake (in Bq) is multiplied by the appropriate DC (or DCF if secular equilibrium has been maintained) to calculate the annual dose (mSv).

7.10. Worker dose assessment

Internal dose assessment for the workforce is determined by:

1. Grouping personnel with similar exposure profiles into SEGs
   - note that SEGs may contain workers from different occupation groups if their exposure profiles are similar.
2. Using the results of the personal air monitoring program to calculate the GM of the gross alpha activity concentration (Bq/m^3) for each task performed by the SEG during the specified monitoring period (AV_i).
3. Applying data from shift rosters or time sheets to estimate the working hours that the average gross alpha activity concentration results apply (HW_i).
4. Determining whether the task is light to moderate activity or heavy physical work, and the appropriate breathing rate (BR). The BR is:
   - 1.2 m^3 per hour for light to moderate work (used as the default value) or
   - 1.6 m^3 per hour for heavy physical work.
5. Assuming the dusts are insoluble (Class S), selecting the appropriate DCF in units of mSv/Bq based on the AMAD and Th:U ratio of the dust in the air samples, and other applicable NORs.
6. Calculating the internal dose by adding all the task intakes and multiplying by the applicable DC as follows:

$$\boldsymbol{Individual\ internal\ dose} = \sum_{\boldsymbol{i}}(\boldsymbol{AV_i} \times \boldsymbol{HW_i} \times \boldsymbol{BR}) \times \boldsymbol{DCF_i}$$

where:

- *AV_i* – average of gross alpha-activity concentration for the work task *i* (Bq/m^3)
- *HW_i* – hours worked by a worker for the work task *i*
- *BR* – assumed breathing rate of a worker (1.2 m^3/hour, may be up to 1.6 m^3/h in cases of heavy work)
- *($AV_i \times HW_i \times BR$)* – personal intake for the period of assessment for the work category *i* (Bq)
- *DCF* – dose conversion factor in mSv/Bq for different AMAD sizes, different uranium and thorium weight ratios and other NORs (assuming secular equilibrium has been maintained). The derivation of DCFS for varying particle sizes and thorium to uranium ratios is provided in Appendix 1).

7.11. Critical group dose assessment

A similar equation can be used for the assessment of internal exposures for the members of the general public with adjustment to BR and DC.

This assumes that the sampling results are of sufficient duration and frequency to be representative of annual exposures.

$$\boldsymbol{Individual\ internal\ dose} = \boldsymbol{AV} \times \boldsymbol{H} \times \boldsymbol{BR} \times \boldsymbol{DCF}$$

where:

- *AV* – average of gross alpha-activity concentration measured for the group of the members of the public (Bq/m^3)
- *H* – realistic estimate of hours per year that a member of the public spends in the specific area (8,760 hours may not be correct in many situations)
- *BR* – assumed breathing rate of a member of the public (0.96 m^3/hour)

- *(AV × H × BR)* – personal intake for the period of assessment for the member of the public (Bq)
- *DCF* – dose conversion factor in mSv/Bq for different uranium and thorium weight ratios. AMAD is always taken as 1 μm for members of the public. The DCF data is provided in Appendix 1.

If there are significant variations likely, such as prevailing winds at certain times of the year, then the summation method provided in Section 7.9.1 may be required to be applied.

7.12. Correction factors for use in assessment of gross alpha activity concentration

The determination of gross alpha concentrations is conducted by collecting a representative sample of airborne dust on a filter.

The filter must be carefully removed from the sampling device and placed into a marked container (e.g. petri dish with lid) for at least 7 days, during which the short lived alpha emitters from the thoron (^{220}Rn) and radon (^{222}Rn) series will decay.

The delay ensures that only the alpha particle emitted from the longer-lived NORs in the ^{232}Th and ^{238}U radioactive decay chain are counted.

The samples should then be counted 7-28 days after collection, no correction is required for loss of activity.

However, if samples are unable to be counted during this period, then a correction factor needs to be applied, as outlined in Tables 16 and 17 in Appendix 1.

(See NORM-3.4 for further information on the methodology to be applied).

8. Internal radiation exposure from the inhalation of radon, thoron and their decay products

Radon, thoron and several of their short-lived decay products (progeny) decay via the emission of alpha particles. Following inhalation of the short-lived NORs, most of their decay takes place in the lungs before clearance can occur. Therefore the pathway can be a significant source of exposure, and is treated as a separate exposure pathway to the internal irradiation from inhalation of long-lived alpha emitting NORs in dust.

The two significant isotopes of radon are:

- radon-222 (^{222}Rn), commonly called radon, the immediate decay product of ^{226}Ra, from the uranium decay series
- radon-220 (^{220}Rn), commonly called thoron, the immediate decay product of ^{224}Ra, from the thorium decay series.

Radon is a noble gas and both its isotopes decay to solid elements the atoms of which typically attach themselves to the dust particles present in air, presenting a potential inhalation exposure pathway. The relevant alpha-emitting radon decay products are:

- ^{218}Po, ^{214}Pb, ^{214}Bi and ^{214}Po, collectively called radon progeny (RnP) from the uranium decay series
- ^{216}Po, ^{212}Bi, ^{212}Po and ^{208}Tl, collectively called thoron progeny (TnP) from the thorium decay series.

Aside from the actual concentration of radon and thoron in the air, the most important variable for estimating doses from this exposure pathway is the equilibrium factor (F) between the parent noble gas, radon or thoron, and their progeny. It is important to note that the majority of doses delivered by this exposure pathway arise from the decay of the progeny, and not the gaseous radon or thoron. Therefore, while it is a relatively straightforward technique, monitoring for radon or thoron gas in isolation is usually not suitable for dose estimation, and:

- where it is used, it must be accompanied by an assumption as to the F that has been applied
- if there is significant variation in F, monitoring for RnP and/or TnP is the most accurate method for dose estimation.

8.1. Calculating potential alpha energy from radon or thoron concentration

To assess the contribution of inhaled radon and thoron, the potential alpha energy concentration (PAEC) released from the decay of radon and TnP must be determined. The PAEC is measured in mJh/m^3.

PAEC is determined from measurements of the PAEC in the air and the volume of air inhaled by using the following formulae (derived from Table 1 of ICRP-137 and paragraph 15 of ICRP-65):

For radon $$P_{RnP} = 5.56 \times 10^{-6} \times t \times F_{RnP} \times C_{Rn}$$

For thoron $$P_{TnP} = 7.56 \times 10^{-5} \times t \times F_{TnP} \times C_{Tn}$$

where:

8.1.1. P_{RnP}, P_{TnP} are the PAEC s of radon decay products and thoron decay products, respectively (mJh/m^3)

8.1.2. *5.56 x 10^{-6}* is the combined PAEC for the RnP series from ^{218}Po to ^{214}Po (mJ/Bq)

8.1.3. *7.56 x 10^{-5}* is the combined PAEC for the TnP series from ^{216}Po to ^{212}Po (mJ/Bq)

8.1.4. *t* is the exposure time (hours)

8.1.5. F_{RnP} is the F between radon and RnP (typically taken as 0.4 for indoor areas and 0.2 for outdoors)

8.1.6. F_{TnP} is the F between thoron and TnP (typically taken as 0.04 for indoor areas and 0.004 for outdoors [ICRP-137])

8.1.7. C_{Rn} is the radon gas concentration (Bq/m^3)

8.1.8. C_{Tn} is the thoron gas concentration (Bq/m^3).

It is important to note that F for both radon (^{222}Rn) and thoron (^{220}Rn) and their respective decay progeny can be highly variable. If directed by the regulator due to the potential for exposure to radon or thoron to significantly contribute to exposure, then site specific values should be measured, calculated and applied, considering:

- it is recognised that monitoring of decay products is labour intensive, and as such long term radon and thoron gas measurements and default equilibrium factors may be preferable for **initial** estimates of potential exposure
- in the absence of site-specific F the defaults listed above should be used.

It is important to note that in part 4.2.3 of RPS-9.1, the F for thoron is taken as 1, which is considered to be a very significant overestimation, as in practice this factor very rarely exceeds 0.1 and thus more practical values of 0.04 for indoors and 0.004 for the outdoors are adopted.

8.2. Calculating dose from equilibrium equivalent measurements

For reference purposes the potential alpha energy exposure of workers was previously expressed in the historical unit working level month (WLM).

The equivalent SI unit is the mJh/m^3 and the conversion is as follows:

8.2.1. 1 WLM = 3.54 mJh/m^3, and by extension

8.2.2. 1 mJh/m^3 = 0.282 WLM [ICRP-137].

Potential alpha energy exposures to RnP and TnP may be determined by integrating the PAEC over the exposure time, and is measured in the SI unit of the joule (J).

Where the measurements of the equilibrium equivalent concentrations of RnP (EEC_{Rn}) and/or TnP (EEC_{Tn}) are conducted, in an indoor workplace a calculation of exposure from the inhalation of radon and TnP depends on the time of the exposure t and is calculated as follows:

$$E_{Rn} = EEC_{Rn} \times f_{Rn} \times t$$
$$E_{Tn} = EEC_{Tn} \times f_{Tn} \times t$$

where (see Table 11 of ICRP-137):

- f_{Rn} is the effective dose per exposure factor for RnP = 1.2×10^{-5} mSv/(Bq·h·m^3)
- f_{Tn} is the effective dose per exposure factor TnP respectively = 1.2×10^{-4} mSv/(Bq·h·m^3).

In a mining environment, the assumption of F is problematic, especially for open pit operations, however, Table 11 provides some guidance:

8.2.3. by inference, using an F = 0.2, and the effective dose per exposure of 11 mSv per WLM, an equivalent effective dose per exposure is 3.45×10^{-6} mSv/(Bq·h·m^3)

8.2.4. using an EEC concentration of ^{220}Rn, the effective dose per exposure is 1.2×10^{-4} mSv/(Bq·h·m^3).

8.3. Calculating dose from RnP or TnP measurements

Although measuring the concentration of RnP and TnP is at times a more arduous task than measuring radon or thoron concentrations, the estimate of dose is relatively straightforward process, and does not rely on an assumption of F.

Once the potential alpha energy, in mJh/m^3, has been determined then conversion to a dose estimate is made by applying the effective dose per exposure factors for a mining environment (see the ICRP *Data Viewer* for P134, P137 and P141):

8.3.1. for radon (^{222}Rn) = 3.14 mSv per mJh/m^3

8.3.2. for thoron (^{220}Rn) = 1.36 mSv per mJh/m^3

The equivalent effective dose per exposure factors for indoor work are:

8.3.3. for radon (^{222}Rn) = 5.59 mSv per mJh/m^3

8.3.4. for thoron (^{220}Rn) = 1.57 mSv per mJh/m^3.

It should be noted that if the indoor work is physically demanding then a factor of 6 mSv per mJh/m^3 should be applied for radon, to compensate for the increased BR.

Application of the calculation method is illustrated in Example 9 in Appendix 3.

8.4. Contribution from background

Determining the contribution from natural background is problematic, and has resulted in ARPANSA issuing guidance in RPS 9.1 that, when calculating doses from inhalation of radon and RnP, no subtraction of background exposure is to be made in areas for which individual dose records are required to be kept.

As per RPS 9.1, 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 a worker had not engaged in the work duties undertaken.

Therefore, and as was discussed in Section 5.4, dose calculations from exposure to ^{222}Rn and ^{220}Rn and their decay products should include background, and be accompanied by a note indicating an estimated concentration of the applicable source of exposure (^{222}Rn or ^{220}Rn or their decay products) and any assumptions made in calculating the dose estimate.

9. Internal radiation exposure (ingestion of drinking water)

All drinking water supplied on mining operations shall meet the *Australian drinking water guidelines* (ADWG). As well as chemical and biological testing it must also be screened (as a minimum) for gross alpha and gross beta activity concentrations.

Copies of the results of NOR analyses of drinking water samples are to be provided to the Water Unit of the WA Department of Health.

The naturally occurring radioisotope ^{40}K is present in all drinking water. As such the contribution from ^{40}K to the gross beta-activity value should be subtracted from the total value when determining the contribution to radiation dose.

The ADWG recommends that a value of 1 mSv per year be used as a default action level, above which some corrective action will be necessary.

9.1. Town or scheme water supplies

At many mining and mineral processing sites in WA, all drinking and washing water is provided from town or scheme water supplies. These supplies are required to comply with the ADWG and should be regularly tested by the supplying authority and confirmed as compliant.

As such, radiation exposure to workers due to the ingestion of drinking water on sites with water supply from town or scheme systems is not attributed to the mining and processing activities and does not need to be considered.

A statement indicating that drinking water is sourced from a town or scheme supply should be made as justification for it to not be considered when assessing radiation exposure.

9.2. Site bore fields or non-scheme water supplies below screening levels

It is the responsibility of the PCBU and the radiation safety officer to ensure that sufficient testing occurs to be able to verify that drinking water meets the ADWG requirements.

In addition to chemical and biological testing all non-scheme drinking water supplies must be assessed on a regular basis to confirm that gross alpha and gross beta activity levels are below the screening level. For supplies above the screening levels, the activity of individual NORs must be determined.

The screening levels as defined in Information sheet 2.2 of the ADWG shall apply, namely:

9.2.1. gross alpha-activity of 0.5 Bq/L

9.2.2. gross beta-activity of 0.5 Bq/L (after subtracting the contribution from ^{40}K).

It should be noted that these screening levels are not health based limits but are the concentrations that, if exceeded, require further investigation to be conducted to determine the risk to the workforce, public or ecosystems and appropriate actions taken.

If the screening levels are not exceeded, then the contribution to radiation dose as a result of ingestion of water is not required to be included in the radiation dose assessment as it is less than levels that may occur in the general community from scheme water.

If either of these activity concentration values are exceeded, specific NORs must be identified and their activity concentrations determined in order to undertake an assessment of radiation exposures from ingestion of water.

9.3. Site bore fields or non- scheme water supplies above screening levels

If the activity concentration screening levels are exceeded, specific NORs must be identified and their activity concentrations determined.

When the exact concentrations of all NORs are unknown, the highest potential exposures are associated with the following NORs.

9.3.1. Thorium decay chain: ^{232}Th, ^{228}Ra, ^{228}Th (possibly also ^{224}Ra) Uranium decay chain: ^{230}Th, ^{226}Ra, ^{210}Pb, ^{210}Po (possibly also ^{234}Th and ^{234}U).

As such the concentrations of (at least) the above NORs must be considered when an assessment of radiation exposures from ingestion of water is made.

The contribution to radiation dose for each NOR is calculated by:

Annual dose (mSv/year) = dose per unit intake (mSv/Bq) x annual water consumption (litre/year) x [radionuclide concentration (Bq/L) – natural background radionuclide concentration (Bq/L)]

The total dose due to ingestion as a consequence of the mine is calculated as a sum of the values for each NOR. It is recommended that the contribution from each nuclide, the natural background contribution and the contribution as a result of the mine are all presented, and:

9.3.2. the level of water consumption is generally assumed to be 2 litres per day equivalent to 730L litres per year (ADWG)

9.3.2.1. If consumption is significantly greater than 2 litres per day, then adjustments must be made.

9.3.3. the natural background concentrations of NORs in local drinking water are typically determined during the site baseline survey prior to the commencement of operations for the NORs listed above (at least for ^{226}Ra and ^{228}Ra for the cases where it is expected that both thorium and uranium decay chains are in secular equilibrium)

9.3.3.1. In the event where baseline data does not exist, the contribution from natural background levels is assumed to be zero.

9.3.4. DCs or the dose per unit intake and other relevant data are presented in Tables 19 and 20.

For some operations, ingestion DCs of ^{40}K and ^{87}Rb (present in some lithium ores) and/or ^{147}Sm (present in monazite) may be need to be assessed:

9.3.5. these are provided in Table 21 and are based on IAEA 2014. An example of calculations in given in

Example 10 in Appendix 2.

10. Internal radiation exposure (ingestion of food, dust and soil)

As fruit, vegetables and meats for consumption by persons on site are rarely grown on mining operations in WA, the pathway of ingestion of NORs, other than ingestion of water by a mine worker, is unlikely except in very rare scenarios.

The risk of ingestion of NORs in foodstuffs is further decreased by maintaining suitable cleansed eating facilities and implementing personal hygiene practices such as regular handwashing, prevention of smoking and decontamination of clothing.

A statement indicating that food is not sourced from the mining operation and that suitable eating and washing facilities are provided should be made as justification for it to not be considered as a potential exposure pathway when assessing radiation exposure.

If food is produced in areas that may be impacted by a mining operation and an assessment is required on the potential for ingestion by members of the public, the DCs for different NORs are provided in Tables 19-21.

The assessment can be quite a complex task. For more information please refer to the guides and reports issued by the IAEA (see Appendix 5). These documents contain most information necessary for the assessment of doses via this pathway, but an adjustment for Australian conditions is almost always necessary – particularly for the reference data in regards to the annual consumption of meat, milk and vegetables.

11. Complex dose assessments

It is important to ensure that the first step in any dose assessment is always the analysis of all possible pathways of radiation exposure and their applicability for the particular situation.

Cases where multiple pathways of radiation exposure need to be assessed are given in Examples 11 and 12 in Appendix 3.

Appendix 1. Derivation of dose conversion factors from inhalation of NORs

Note: Data is extracted from the Occupational Intake of Radionuclides Data Viewer *for P134, P137 and P141, v4010419, July 30 2019.*

Table 1: Committed effective dose, inhalation of thorium ore dust, AMAD = 1 µm, lung absorption Class S (where applicable)

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq)		Dose (Sv)
			Alpha	Beta	
^{232}Th	alpha	1.0×10^{-4}	1		1.0×10^{-4}
^{228}Ra	beta	3.7×10^{-5}		1	3.7×10^{-5}
^{228}Ac	beta	1.3×10^{-8}		1	1.3×10^{-8}
^{228}Th	alpha	3.5×10^{-5}	1		3.5×10^{-5}
^{224}Ra	alpha	1.6×10^{-6}	1		1.6×10^{-6}
^{220}Rn*	alpha	–	1		–
^{216}Po*	alpha	–	1		–
^{212}Pb*	beta	1.1×10^{-7}		1	1.1×10^{-7}
^{212}Bi*	64.1% beta 35.9% alpha	2.4×10^{-8}	0.359	0.641	2.4×10^{-8}
^{212}Po*	alpha	–	0.641	0.359	–
^{208}Tl*	beta	–			–
Total			**6**	**4**	**1.74×10^{-4}**

* ^{220}Rn and short-lived decay products

Committed effective dose per unit intake of alpha activity (DCF):

$$DCF_{1\ \mu m\ Th\ dust} = \frac{1.74 \times 10^{-4}\ Sv}{6\ Bq_{\alpha}} = 2.90 \times 10^{-5} \frac{Sv}{Bq_{\alpha}} = 0.0290\ mSv/Bq$$

Table 2: Committed effective dose, inhalation of thorium ore dust, AMAD = 3 μm, Lung absorption Class S (where applicable)

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq)		Dose (Sv)
			Alpha	Beta	
^{232}Th	alpha	7.5×10^{-5}	1		7.5×10^{-5}
^{228}Ra	beta	3.0×10^{-5}		1	3.0×10^{-5}
^{228}Ac	beta	1.1×10^{-8}		1	1.1×10^{-8}
^{228}Th	alpha	2.9×10^{-5}	1		2.9×10^{-5}
^{224}Ra	alpha	1.3×10^{-6}	1		1.3×10^{-6}
^{220}Rn*	alpha	–	1		–
^{216}Po*	alpha	–	1		–
^{212}Pb*	beta	1.1×10^{-7}		1	1.1×10^{-7}
^{212}Bi*	64.1% beta 35.9% alpha	3.1×10^{-8}	0.359	0.641	3.1×10^{-8}
^{212}Po*	alpha	–	0.641	0.359	–
^{208}Tl*	beta	–			–
Total			**6**	**4**	**1.35×10^{-4}**

* ^{220}Rn and short-lived decay products

Committed effective dose per unit intake of alpha activity (DCF):

$$DCF_{3\ \mu m\ Th\ dust} = \frac{1.35 \times 10^{-4}\ Sv}{6\ Bq_{\alpha}} = 2.25 \times 10^{-5} \frac{Sv}{Bq_{\alpha}} = 0.0225\ mSv/Bq$$

Table 3: Committed effective dose, inhalation of thorium ore dust, AMAD = 5 µm (default), lung absorption Class S (where applicable)

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq)		Dose (Sv)
			Alpha	Beta	
^{232}Th	alpha	5.4×10^{-5}	1		5.4×10^{-5}
^{228}Ra	beta	2.2×10^{-5}		1	2.2×10^{-5}
^{228}Ac	beta	8.4×10^{-9}		1	8.4×10^{-9}
^{228}Th	alpha	2.3×10^{-5}	1		2.3×10^{-5}
^{224}Ra	alpha	1.1×10^{-6}	1		1.1×10^{-6}
^{220}Rn*	alpha	–	1		–
^{216}Po*	alpha	–	1		–
^{212}Pb*	beta	9.4×10^{-8}		1	9.4×10^{-8}
^{212}Bi*	64.1% beta 35.9% alpha	2.9×10^{-8}	0.359	0.641	2.9×10^{-8}
^{212}Po*	alpha	–	0.641	0.359	–
^{208}Tl*	beta	–			–
Total			**6**	**4**	**1.00×10^{-4}**

* ^{220}Rn and short-lived decay products

Committed effective dose per unit intake of alpha activity (DCF):

$$DCF_{5\,\mu m\,Th\,dust} = \frac{1.00 \times 10^{-4}\,Sv}{6Bq_\alpha} = 1.67 \times 10^{-5}\,\frac{Sv}{Bq_\alpha} = \mathbf{0.0167\,mSv/Bq}$$

Table 4: Committed effective dose, inhalation of thorium ore dust, AMAD = 10 µm, lung absorption Class S (where applicable)

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq) Alpha	Quantity inhaled (Bq) Beta	Dose (Sv)
^{232}Th	alpha	2.6×10^{-5}	1		2.6×10^{-5}
^{228}Ra	beta	1.3×10^{-5}		1	1.3×10^{-5}
^{228}Ac	beta	5.1×10^{-9}		1	5.1×10^{-9}
^{228}Th	alpha	1.4×10^{-5}	1		1.4×10^{-5}
^{224}Ra	alpha	6.5×10^{-7}	1		6.5×10^{-7}
^{220}Rn*	alpha	–	1		–
^{216}Po*	alpha	–	1		–
^{212}Pb*	beta	6.2×10^{-8}		1	6.2×10^{-8}
^{212}Bi*	64.1% beta 35.9% alpha	2.1×10^{-8}	0.359	0.641	2.1×10^{-8}
^{212}Po*	alpha	–	0.641	0.359	–
^{208}Tl*	beta	–			–
Total			**6**	**4**	**5.37×10^{-5}**

* ^{220}Rn and short-lived decay products

Committed effective dose per unit intake of alpha activity (DCF):

$$DCF_{10\,\mu m\,Th\,dust} = \frac{5.37 \times 10^{-5}\,Sv}{6\,Bq_\alpha} = 0.90 \times 10^{-5}\frac{Sv}{Bq_\alpha} = 0.0090\,mSv/Bq$$

Table 5: Committed effective dose, inhalation of thorium ore dust, AMAD = 20 µ, lung absorption Class S (where applicable)

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq)		Dose (Sv)
			Alpha	Beta	
^{232}Th	alpha	1.1×10^{-5}	1		1.1×10^{-5}
^{228}Ra	beta	6.4×10^{-6}		1	6.4×10^{-6}
^{228}Ac	beta	2.9×10^{-9}		1	2.9×10^{-9}
^{228}Th	alpha	7.7×10^{-6}	1		7.7×10^{-6}
^{224}Ra	alpha	3.0×10^{-7}	1		3.0×10^{-7}
^{220}Rn*	alpha	–	1		–
^{216}Po*	alpha	–	1		–
^{212}Pb*	beta	3.4×10^{-8}		1	3.4×10^{-8}
^{212}Bi*	64.1% beta 35.9% alpha	1.3×10^{-8}	0.359	0.641	1.3×10^{-8}
^{212}Po*	alpha	–	0.641	0.359	–
^{208}Tl*	beta	–			–
Total			**6**	**4**	**2.54×10^{-5}**

* ^{220}Rn and short-lived decay products

Committed effective dose per unit intake of alpha activity (DCF):

$$DCF_{20\,\mu m\,Th\,dust} = \frac{2.54 \times 10^{-5}\,Sv}{6\,Bq_{\alpha}} = 0.42 \times 10^{-5}\,\frac{Sv}{Bq_{\alpha}} = 0.0042\,mSv/Bq$$

Table 6: Committed effective dose, inhalation of uranium ore dust, AMAD = 1 µm, lung absorption Class S (where applicable)

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq)		Dose (Sv)
			Alpha	Beta	
^{238}U	alpha	2.0×10^{-5}	1		2.0×10^{-5}
^{234}Th	beta	4.9×10^{-9}		1	4.9×10^{-9}
$^{234}Pa_m$	beta	1.7×10^{-10}		1	1.7×10^{-10}
^{234}U	alpha	2.3×10^{-5}	1		2.3×10^{-5}
^{230}Th	alpha	2.5×10^{-5}	1		2.5×10^{-5}
^{226}Ra	alpha	2.3×10^{-5}	1		2.3×10^{-5}
^{222}Rn*	alpha	–	1		–
^{218}Po*	alpha	–	1		–
^{214}Pb*	beta	1.1×10^{-8}		1	1.1×10^{-8}
^{214}Bi*	beta	1.0×10^{-8}		1	1.0×10^{-8}
^{214}Po*	alpha	–	1		–
^{210}Pb	beta	1.5×10^{-5}		1	1.5×10^{-5}
^{210}Bi (Class M)	beta	8.7×10^{-8}		1	8.7×10^{-8}
^{210}Po	alpha	2.8×10^{-6}	1		2.8×10^{-6}
^{235}U	alpha	2.1×10^{-5}	0.046		9.7×10^{-7}
^{231}Th	beta	1.7×10^{-10}		0.046	7.8×10^{-12}
^{231}Pa	alpha	8.4×10^{-5}	0.046		3.9×10^{-6}
^{227}Ac	beta	1.1×10^{-4}		0.046	5.1×10^{-6}
^{227}Th	alpha	3.3×10^{-6}	0.046		1.5×10^{-7}
^{223}Ra	alpha	3.2×10^{-6}	0.046		1.5×10^{-7}
^{219}Rn*	alpha	–	0.046		–
^{215}Po*	alpha	–	0.046		–
^{211}Pb* (Class F)	beta	1.1×10^{-8}		0.046	5.1×10^{-10}
^{211}Bi*	alpha	–	0.046		–
^{207}Tl*	beta	–		0.046	–
Total			**8.322**	**6.184**	$\mathbf{1.19 \times 10^{-4}}$

* ^{222}Rn, ^{219}Rn, short-lived decay products

Committed effective dose per unit intake of alpha activity (DCF):

$$DCF_{1\ \mu m\ U\ dust} = \frac{1.19 \times 10^{-4}\ Sv}{8.322\ Bq_{\alpha}} = 1.43 \times 10^{-5} \frac{Sv}{Bq_{\alpha}} = 0.0143\ mSv/Bq$$

Table 7: Committed effective dose, inhalation of uranium ore dust, AMAD = 3 µm, lung absorption Class S (where applicable)

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq)		Dose (Sv)
			Alpha	Beta	
^{238}U	alpha	1.6×10^{-5}	1		1.6×10^{-5}
^{234}Th	beta	3.8×10^{-9}		1	3.8×10^{-9}
$^{234}Pa_m$	beta	2.1×10^{-10}		1	2.1×10^{-10}
^{234}U	alpha	1.8×10^{-5}	1		1.8×10^{-5}
^{230}Th	alpha	2.0×10^{-5}	1		2.0×10^{-5}
^{226}Ra	alpha	1.8×10^{-5}	1		1.8×10^{-5}
^{222}Rn*	alpha	–	1		–
^{218}Po*	alpha	–	1		–
^{214}Pb*	beta	1.4×10^{-8}		1	1.4×10^{-8}
^{214}Bi*	beta	1.4×10^{-8}		1	1.4×10^{-8}
^{214}Po*	alpha	–	1		–
^{210}Pb	beta	1.2×10^{-5}		1	1.2×10^{-5}
^{210}Bi (Class M)	beta	7.3×10^{-8}		1	7.3×10^{-8}
^{210}Po	alpha	2.3×10^{-6}	1		2.3×10^{-6}
^{235}U	alpha	1.6×10^{-5}	0.046		7.4×10^{-7}
^{231}Th	beta	1.6×10^{-10}		0.046	7.4×10^{-12}
^{231}Pa	alpha	6.4×10^{-5}	0.046		2.9×10^{-6}
^{227}Ac	beta	8.7×10^{-5}		0.046	4.0×10^{-6}
^{227}Th	alpha	2.7×10^{-6}	0.046		1.2×10^{-7}
^{223}Ra	alpha	2.7×10^{-6}	0.046		1.2×10^{-7}
^{219}Rn*	alpha	–	0.046		–
^{215}Po*	alpha	–	0.046		–
^{211}Pb* (Class F)	beta	1.4×10^{-8}		0.046	6.4×10^{-10}
^{211}Bi*	alpha	–	0.046		–
^{207}Tl*	beta	–		0.046	–
Total			**8.322**	**6.184**	**9.43×10^{-5}**

* ^{222}Rn, ^{219}Rn, short-lived decay products

Committed effective dose per unit intake of alpha activity (DCF):

$$DCF_{3\ \mu m\ U\ dust} = \frac{9.43 \times 10^{-5}\ Sv}{8.322\ Bq_\alpha} = 1.13 \times 10^{-5}\ \frac{Sv}{Bq_\alpha} = 0.0113\ mSv/Bq$$

Table 8: Committed effective dose, inhalation of uranium ore dust, AMAD = 5 µm (DEFAULT), lung absorption Class S (where applicable)

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq)		Dose (Sv)
			Alpha	Beta	
^{238}U	alpha	1.2×10^{-5}	1		1.2×10^{-5}
^{234}Th	beta	2.9×10^{-9}		1	2.9×10^{-9}
$^{234}Pa_m$	beta	2.0×10^{-10}		1	2.0×10^{-10}
^{234}U	alpha	1.3×10^{-5}	1		1.3×10^{-5}
^{230}Th	alpha	1.5×10^{-5}	1		1.5×10^{-5}
^{226}Ra	alpha	1.3×10^{-5}	1		1.3×10^{-5}
^{222}Rn*	alpha	–	1		–
^{218}Po*	alpha	–	1		–
^{214}Pb*	beta	1.4×10^{-8}		1	1.4×10^{-8}
^{214}Bi*	beta	1.4×10^{-8}		1	1.4×10^{-8}
^{214}Po*	alpha	–	1		–
^{210}Pb	beta	9.2×10^{-6}		1	9.2×10^{-6}
^{210}Bi (Class M)	beta	5.7×10^{-8}		1	5.7×10^{-8}
^{210}Po	alpha	1.8×10^{-6}	1		1.8×10^{-6}
^{235}U	alpha	1.2×10^{-5}	0.046		5.5×10^{-7}
^{231}Th	beta	1.3×10^{-10}		0.046	6.0×10^{-12}
^{231}Pa	alpha	4.6×10^{-5}	0.046		2.1×10^{-6}
^{227}Ac	beta	6.5×10^{-5}		0.046	3.0×10^{-6}
^{227}Th	alpha	2.1×10^{-6}	0.046		9.7×10^{-8}
^{223}Ra	alpha	2.2×10^{-6}	0.046		1.0×10^{-7}
^{219}Rn*	alpha	–	0.046		–
^{215}Po*	alpha	–	0.046		–
^{211}Pb* (Class F)	beta	1.3×10^{-8}		0.046	6.0×10^{-10}
^{211}Bi*	alpha	–	0.046		–
^{207}Tl*	beta	–		0.046	–
Total			**8.322**	**6.184**	**6.99×10^{-5}**

* ^{222}Rn, ^{219}Rn, short-lived decay products

Committed effective dose per unit intake of alpha activity (DCF):

$$DCF_{5\,\mu m\ U\ dust} = \frac{6.99 \times 10^{-5}\ Sv}{8.322\ Bq_{\alpha}} = 0.84 \times 10^{-5} \frac{Sv}{Bq_{\alpha}} = 0.0084\ mSv/Bq$$

Table 9: Committed effective dose, inhalation of uranium ore dust, AMAD = 10 µm, lung absorption Class S (where applicable)

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq)		Dose (Sv)
			Alpha	Beta	
^{238}U	alpha	6.3×10^{-6}	1		6.3×10^{-6}
^{234}Th	beta	1.6×10^{-9}		1	1.6×10^{-9}
^{234}Pa$_m$	beta	1.6×10^{-10}		1	1.6×10^{-10}
^{234}U	alpha	7.2×10^{-6}	1		7.2×10^{-6}
^{230}Th	alpha	7.8×10^{-6}	1		7.8×10^{-6}
^{226}Ra	alpha	7.2×10^{-6}	1		7.2×10^{-6}
^{222}Rn*	alpha	–	1		–
^{218}Po*	alpha	–	1		–
^{214}Pb*	beta	1.0×10^{-8}		1	1.0×10^{-8}
^{214}Bi*	beta	1.1×10^{-8}		1	1.1×10^{-8}
^{214}Po*	alpha	–	1		–
^{210}Pb	beta	5.1×10^{-6}		1	5.1×10^{-6}
^{210}Bi (Class M)	beta	3.4×10^{-8}		1	3.4×10^{-8}
^{210}Po	alpha	1.1×10^{-6}	1		1.1×10^{-6}
^{235}U	alpha	6.6×10^{-6}	0.046		3.0×10^{-7}
^{231}Th	beta	8.6×10^{-11}		0.046	4.0×10^{-12}
^{231}Pa	alpha	2.3×10^{-5}	0.046		1.1×10^{-6}
^{227}Ac	beta	3.6×10^{-5}		0.046	1.7×10^{-6}
^{227}Th	alpha	1.2×10^{-6}	0.046		5.5×10^{-8}
^{223}Ra	alpha	1.3×10^{-6}	0.046		6.0×10^{-8}
^{219}Rn*	alpha	–	0.046		–
^{215}Po*	alpha	–	0.046		–
^{211}Pb* (Class F)	beta	9.2×10^{-9}		0.046	4.2×10^{-10}
^{211}Bi*	alpha	–	0.046		–
^{207}Tl*	beta	–		0.046	–
Total			**8.322**	**6.184**	**3.79×10^{-5}**

* ^{222}Rn, ^{219}Rn, short-lived decay products

Committed effective dose per unit intake of alpha activity (DCF):

$$DCF_{10\,\mu m\,U\,dust} = \frac{3.79 \times 10^{-5}\,Sv}{8.322\,Bq_{\alpha}} = 0.46 \times 10^{-5}\,\frac{Sv}{Bq_{\alpha}} = 0.0046\,mSv/Bq$$

Table 10: Committed effective dose, inhalation of uranium ore dust, AMAD = 20 µm, lung absorption Class S (where applicable)

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq)		Dose (Sv)
			Alpha	Beta	
^{238}U	alpha	3.1×10^{-6}	1		3.1×10^{-6}
^{234}Th	beta	7.8×10^{-10}		1	7.8×10^{-10}
$^{234}Pa_m$	beta	1.2×10^{-10}		1	1.2×10^{-10}
^{234}U	alpha	3.5×10^{-6}	1		3.5×10^{-6}
^{230}Th	alpha	3.7×10^{-6}	1		3.7×10^{-6}
^{226}Ra	alpha	3.6×10^{-6}	1		3.6×10^{-6}
^{222}Rn*	alpha	–	1		–
^{218}Po*	alpha	–	1		–
^{214}Pb*	beta	5.9×10^{-9}		1	5.9×10^{-9}
^{214}Bi*	beta	6.8×10^{-9}		1	6.8×10^{-9}
^{214}Po*	alpha	–	1		–
^{210}Pb	beta	2.6×10^{-6}		1	2.6×10^{-6}
^{210}Bi (Class M)	beta	1.9×10^{-8}		1	1.9×10^{-8}
^{210}Po	alpha	5.8×10^{-7}	1		5.8×10^{-7}
^{235}U	alpha	3.2×10^{-6}	0.046		1.5×10^{-7}
^{231}Th	beta	4.6×10^{-11}		0.046	2.1×10^{-12}
^{231}Pa	alpha	9.4×10^{-6}	0.046		4.3×10^{-7}
^{227}Ac	beta	1.8×10^{-5}		0.046	8.3×10^{-7}
^{227}Th	alpha	6.1×10^{-7}	0.046		2.8×10^{-8}
^{223}Ra	alpha	6.1×10^{-7}	0.046		2.8×10^{-8}
^{219}Rn*	alpha	–	0.046		-
^{215}Po*	alpha	–	0.046		-
^{211}Pb* (Class F)	beta	5.2×10^{-9}		0.046	2.4×10^{-10}
^{211}Bi*	alpha	–	0.046		–
^{207}Tl*	beta	–		0.046	–
Total			**8.322**	**6.184**	$\mathbf{1.86 \times 10^{-5}}$

* ^{222}Rn, ^{219}Rn, short-lived decay products

Committed effective dose per unit intake of alpha activity (DCF):

$$DCF_{20\ \mu m\ U\ dust} = \frac{1.86 \times 10^{-5}\ Sv}{8.322\ Bq_{\alpha}} = 0.22 \times 10^{-5} \frac{Sv}{Bq_{\alpha}} = 0.0022\ mSv/Bq$$

Table 11: Dose Conversion Factors in mSv/Bq$_\alpha$, for dust containing both thorium and uranium in different weight ratios

Th : U weight ratio	Dose coefficient (mSv/Bq$_\alpha$), for an AMAD of:				
	1 µm	3 µm	5 µm	10 µm	20 µm
All thorium	0.0290	0.0226	0.0167	0.0090	0.0042
50:1	0.0282	0.0219	0.0162	0.0087	0.0041
40:1	0.0280	0.0218	0.0161	0.0087	0.0041
30:1	0.0276	0.0216	0.0159	0.0086	0.0040
20:1	0.0271	0.0211	0.0156	0.0084	0.0039
15:1	0.0265	0.0207	0.0153	0.0083	0.0039
10:1	0.0256	0.0200	0.0148	0.0080	0.0037
9:1	0.0253	0.0197	0.0146	0.0079	0.0037
8:1	0.0249	0.0195	0.0144	0.0078	0.0036
7:1	0.0245	0.0192	0.0142	0.0077	0.0036
6:1	0.0240	0.0188	0.0139	0.0075	0.0035
5:1	0.0234	0.0183	0.0136	0.0073	0.0034
4:1	0.0226	0.0177	0.0131	0.0071	0.0033
3:1	0.0216	0.0169	0.0125	0.0068	0.0032
2:1	0.0201	0.0158	0.0117	0.0063	0.0030
1.75:1	0.0197	0.0154	0.0114	0.0062	0.0029
1.5:1	0.0191	0.0150	0.0111	0.0060	0.0029
1:1	0.0179	0.0141	0.0104	0.0057	0.0027
1:1.5	0.0169	0.0133	0.0099	0.0054	0.0026
1:1.75	0.0166	0.0131	0.0097	0.0053	0.0025
1:2	0.0164	0.0129	0.0096	0.0052	0.0025
1:3	0.0157	0.0124	0.0092	0.0050	0.0024
1:4	0.0154	0.0122	0.0090	0.0049	0.0024
1:5	0.0152	0.0120	0.0089	0.0049	0.0023
1:6	0.0151	0.0119	0.0088	0.0048	0.0023
1:7	0.0150	0.0118	0.0088	0.0048	0.0023
1:8	0.0149	0.0117	0.0087	0.0048	0.0023
1:9	0.0148	0.0117	0.0087	0.0048	0.0023
1:10	0.0148	0.0117	0.0087	0.0047	0.0023
1:15	0.0146	0.0115	0.0086	0.0047	0.0022
1:20	0.0145	0.0115	0.0085	0.0047	0.0022
1:30	0.0145	0.0114	0.0085	0.0046	0.0022
1:40	0.0144	0.0114	0.0085	0.0046	0.0022
1:50	0.0144	0.0114	0.0085	0.0046	0.0022
All uranium	0.0143	0.0113	0.0084	0.0046	0.0022

Table 12: Dose Conversion Factors for a default AMAD of 5 µm for dusts typically generated from processing of WA minerals

Relevant mineral or material (typical Th : U weight ratio)	DC (mSv/Bq_{α})
Bauxite (1.5:1)	0.0111
Coal (all U)	0.0084
Copper concentrate (all U)	0.0084
Heavy mineral sands concentrate (10:1)	0.0148
Ilmenite (15:1)	0.0153
Iron ore (all U)	0.0084
Monazite (30:1)	0.0159
Phosphate ore and fertilisers (1:20 – 1:25)	0.0085
Rare earth concentrate (25:1)	0.0158
Red mud (1.5 to 1)	0.0111
Rutile (1.25:1)	0.0108
Silica fume (1:4)	0.0090
Tantalum concentrate (1:10 to 1:25)	0.0086
Uranium ore (all U)	0.0084
Zircon and zirconia (1:1.25)	0.0101

Note: For preliminary assessments only, the Th:U weight ratio is to be confirmed where regular dust monitoring may need to be carried out

Table 13: Dose coefficients for the inhalation of ^{210}Po and ^{210}Pb dusts

Radionuclide	Slowest lung absorption class	DC (Sv/Bq), for AMAD of:				
		0.03 µm	0.1 µm	0.3 µm	1 µm	3 µm
^{210}Po	S	5.6×10^{-5}	2.8×10^{-5}	2.0×10^{-5}	1.5×10^{-5}	1.2×10^{-5}
^{210}Pb	S	1.1×10^{-5}	5.5×10^{-6}	3.9×10^{-6}	2.8×10^{-6}	2.3×10^{-6}

Table 14: Dose coefficients for the inhalation of dust containing ^{40}K and ^{87}Rb

Radionuclide	Slowest lung absorption class	DC (Sv/Bq), for AMAD of:	
		1 µm	5 µm
^{40}K	F	2.1×10^{-9}	3.0×10^{-9}
^{87}Rb	F	5.1×10^{-10}	7.6×10^{-10}
^{147}Sm	M	8.9×10^{-6}	6.1×10^{-6}

Table 15: Coefficients between content of NORs and activity concentration

Element	From oxide to element (µg/g, ppm)		Activity concentration (Bq/µg)
Uranium	U_3O_8 to U	× 0.848	1.2384 × 10^{-2}
Thorium	ThO_2 to Th	× 0.879	4.055 × 10^{-3}
Potassium	K_2O to K	× 0.879	8.68 × 10^{-4} (*)
Rubidium	Rb_2O to Rb	× 0.914	3.2 × 10^{-5} (*)
Samarium	Sm_2O_3 to Sm	× 0.862	1.25 × 10^{-4} (*)

* Taking into account that:

- 0.012% of potassium is ^{40}K
- 28% of rubidium is ^{87}Rb
- 15% of samarium is ^{147}Sm

Table 16: Alpha activities and correction factors for thorium ore dust residing on an air sampling filter (reproduced from IAEA RS-G-1.6 p. 84)

Alpha activity residing on the filter for various retention fractions of ^{220}Rn (Bq)

Alpha-emitting radionuclide		Realistic range		Hypothetical extreme case
	100%	75%	50%	0%
^{232}Th	1	1	1	1
^{228}Th	1	1	1	1
^{224}Ra	1	1	1	1
^{220}Rn*	1	0.75	0.5	-
^{216}Po*	1	0.75	0.5	-
^{212}Bi*	0.359	0.269	0.180	-
^{212}Po*	0.641	0.481	0.321	-
Total (gross) alpha activity on the filter	6	5.25	4.5	3
Correction factor for determining alpha activity	**1**	**1.14**	**1.33**	**2**

* ^{220}Rn and short-lived decay products

Table 17: Alpha activities and correction factors for uranium ore dust residing on an air sampling filter (reproduced from IAEA RS-G-1.6 p. 82)

Alpha activity residing on the filter for various retention fractions of ^{222}Rn (Bq)

Alpha-emitting radionuclide		Realistic range		Hypothetic extreme case
	100%	75%	50%	0%
^{238}U	1	1	1	1
^{234}U	1	1	1	1
^{230}Th	1	1	1	1
^{226}Ra	1	1	1	-
^{222}Rn*	1	0.75	0.5	-
^{218}Po*	1	0.75	0.5	-
^{214}Po*	1	0.75	0.5	-
^{210}Po	1	1	1	1
^{235}U	0.046	0.046	0.046	0.046
^{231}Pa	0.046	0.046	0.046	0.046
^{227}Th	0.046	0.046	0.046	0.046
^{223}Ra	0.046	0.046	0.046	0.046
^{219}Rn*	0.046	0.035	0.023	
^{215}Po*	0.046	0.035	0.023	
^{211}Bi*	0.046	0.035	0.023	
Total (gross) alpha activity on the filter	8.3226	7.538	6.753	5.184
Correction factor for determining alpha activity	**1**	**1.10**	**1.23**	**1.61**

* ^{222}Rn, ^{219}Rn and short-lived decay products

Table 18: Dose coefficients for the inhalation of radon (^{222}Rn) and thoron (^{220}Rn)

Radionuclide	Factor (mSv/[mJh/m^3])
Radon (^{222}Rn)	3.14*
Thoron (^{220}Rn)	1.36

* For indoor workplaces where workers are engaged in substantial physical activities, the ICRP recommends a higher DC of 6 mSv per mJh/m^3

Appendix 2. Dose coefficients (ingestion) for NORs

Table 19: Dose coefficients for the ingestion of NORs from thorium decay chain

Radionuclide	DC (Sv/Bq)
^{232}Th	7.0×10^{-8}
^{228}Ra	3.4×10^{-7}
^{228}Ac	1.6×10^{-10}
^{228}Th	3.1×10^{-8}
^{224}Ra	2.9×10^{-8}
^{220}Rn*	
^{216}Po*	
^{212}Pb*	5.6×10^{-9}
^{212}Bi*	1.1×10^{-10}
^{212}Po*	
^{208}Tl*	

* ^{220}Rn and short-lived decay products

Table 20: Dose coefficients for the ingestion of NORs from the uranium decay chain

Radionuclide	Decay	Dose coefficient (Sv/Bq)	Quantity inhaled (Bq)		Dose (Sv)
			Alpha	Beta	
^{238}U	Alpha	3.1×10^{-9}	1		3.1×10^{-9}
^{234}Th	Beta	5.9×10^{-10}		1	5.9×10^{-10}
$^{234}Pa_m$	Beta	1.7×10^{-10}		1	1.7×10^{-10}
^{234}U	Alpha	3.5×10^{-9}	1		3.5×10^{-9}
^{230}Th	Alpha	6.0×10^{-8}	1		6.0×10^{-8}
^{226}Ra	Alpha	1.3×10^{-7}	1		1.3×10^{-7}
^{222}Rn*	Alpha		1		
^{218}Po*	Alpha		1		
^{214}Pb*	Beta	7.7×10^{-11}		1	7.7×10^{-11}
^{214}Bi*	Beta	4.7×10^{-11}		1	4.7×10^{-11}
^{214}Po*	Alpha		1		
^{210}Pb	Beta	3.2×10^{-7}		1	3.2×10^{-7}
^{210}Bi	Beta	1.1×10^{-9}		1	1.1×10^{-9}
^{210}Po	Alpha	1.8×10^{-7}	1		1.8×10^{-7}
^{235}U	Alpha	3.3×10^{-9}	0.046		1.5×10^{-10}
^{231}Th	Beta	1.7×10^{-11}		0.046	7.8×10^{-13}
^{231}Pa	Alpha	1.8×10^{-7}	0.046		8.3×10^{-9}
^{227}Ac	Beta	1.7×10^{-7}		0.046	7.8×10^{-9}
^{227}Th	Alpha	1.3×10^{-9}	0.046		6.0×10^{-11}
^{223}Ra	Alpha	4.1×10^{-8}	0.046		1.9×10^{-9}
^{219}Rn*	Alpha		0.046		
^{215}Po*	Alpha		0.046		
^{211}Pb*	Beta	1.0×10^{-10}		0.046	4.6×10^{-12}
^{211}Bi*	Alpha		0.046		
^{207}Tl*	Beta			0.046	

* ^{222}Rn, ^{219}Rn and short-lived decay products

Table 21: Dose coefficients for the ingestion of ^{40}K, ^{87}Rb and ^{147}Sm

Radionuclide	DC (Sv/Bq)
^{40}K	6.2×10^{-9}
^{87}Rb	1.5×10^{-9}
^{147}Sm	4.9×10^{-8}

Appendix 3. Calculation examples

Example 1

A worker was monitored for the external radiation exposure by the use of OSL badges. The quarterly monitoring results were: 0.19, <MDL, 0.23, and 0.15 mSv. The laboratory certificate states that the MDL for this type of OSL is 0.02 mSv.

The annual external dose is calculated as a sum of TLD badges results:

0.19 + 0.02 + 0.23 + 0.15 = **0.59 mSv**

Example 2

A worker is a part of a SEG of workers consisting of 12 people. Although the worker was not individually monitored for external radiation exposure, five other workers in the SEG were monitored with OSL badges.

The annual average for a worker in this work category is 0.27 mSv.

Therefore, the annual external exposure of this worker is estimated at **0.27 mSv**.

Example 3

A worker works in the office but occasionally visits a production area where NORMs are processed. The average dose rates are: in the worker's office 0.22 µSv/h; in the production area 1.12 µSv/h; background dose rate for the site is 0.15 µSv/h.

Exposure is estimated on the basis of a 'time and motion' study for the worker:

- 1800 hours at 0.22 µSv/hour = 396 µSv,
- 200 hours at 1.12 µSv/hour = 224 µSv;

The sum of external exposure is: 396 µSv + 224 µSv = 620 µSv = 0.62 mSv;

The `background' exposure is: 2000 hours at 0.15 µSv/h = 300 µSv = 0.30 mSv;

The total external exposure is: 0.62 mSv – 0.30 mSv = 0.32 mSv.

Example 4

A parcel of land has been rehabilitated and it is expected that it will be used for residential development. The average result of the post-mining radiation survey is 0.18 µSv/h; natural background gamma dose rate for the site prior to operations was 0.15 µSv/h.

Potential maximum exposure of the member of the general public to the external gamma-radiation is estimated as follows:

- 8760 hours at 0.18 µSv/h = 1577 µSv = 1.58 mSv,
- the `background' exposure is 8760 hours at 0.15 µSv/h = 1314 µSv = 1.31 mSv;

The total external exposure is: 1.58 mSv – 1.31 mSv = 0.27 mSv.

Example 5

A worker worked for three months in a plant where thorium-containing mineral is processed. The worker worked in several SEGs and the corresponding average dust activity concentrations were:

- shift coordinator: 200 hours, 0.039 Bq/m^3 ;
- dry plant operator: 100 hours, 0.213 Bq/m^3 ; and
- wet concentrator operator: 200 hours, 0.021 Bq/m^3.

A ‘special exposure’ was declared for this worker during work in the ‘dry plant operator’ SEG and an incident investigation was undertaken: 8 hours’ exposure to the alpha-activity concentration of 1.435 Bq/m^3; heavy workload – the BR was assumed to have been 1.6 m^3/hour.

Assuming a default particle size value of 5 μm (using a DC of 0.0167 mSv/Bq taken from Table 11), the internal dose from dust inhalation is calculated by calculating the intake for each work category:

- shift coordinator: 200 hours × 0.039 Bq/m^3 × 1.2 m^3/hour = 9.36 Bq
- plant operator: (100 – 8) hours × 0.213 Bq/m^3 × 1.2 m^3/hour = 23.51 Bq
- wet plant operator: 200 hours × 0.021 Bq/m^3 × 1.2 m^3/hour = 5.04 Bq
- special exposure: 8 hours × 1.435 Bq/m^3 × 1.6 m^3/hour = 18.37 Bq

All intakes are summed: 9.36 + 23.51 + 5.04 + 18.37 = 56.28 Bq

The internal dose due to dust inhalation for the three-month period is: 56.28 Bq × 0.0167 mSv/Bq = **0.94 mSv**.

Example 6

After consultation with the workforce, it is decided that the most appropriate method to remove scale from the internal surface of a processing vessel is by grinding the contaminated surface. As well as the introduction of non-radiation hazards such as noise and musculoskeletal stress, the descaling operation has the potential to liberate large amounts of dust into the workplace atmosphere. As a result, it is mandated that respiratory protection is to be worn in the work area.

Because this is a one-off exercise it is decided to not apply to the regulator for approval of the use of a respiratory program protection factor, and that the dose estimate will not consider the use of respiratory protection.

Monitoring results indicate that dust contains 7 Bq/m^3 of predominantly ^{226}Ra and the size of dust particles is 5 μm. The duration of the task was 10 hours. The (potential) internal dose from dust inhalation is calculated as follows:

- the (potential) intake of ^{226}Ra is: 10 hours × 7 Bq/m^3 × 1.2 m^3/hour = 84 Bq
- the DC for 5 μm dust containing ^{226}Ra is 0.0130 mSv/Bq (Table 8)
- the (potential) internal dose due to dust inhalation is: 84 Bq × 0.0130 mSv/Bq = **1.09 mSv**.
    - A note is to be made in the dose assessment record that respiratory protection was worn.

Example 7

In one year an operator in a zircon processing plant is exposed to 5 μm zirconia dust containing 0.039 Bq/m^3 for 800 hours, and to 3 μm silica fume dust containing 0.054 Bq/m^3 for 400 hours. The internal dose from dust inhalation is calculated as follows:

- the intake from inhalation of zirconia dust is: 800 hours × 0.039 Bq/m^3 × 1.2 m^3/hour = 37.44 Bq
- the intake from inhalation of silica fume dust is: 400 hours × 0.054 Bq/m^3 × 1.2 m^3/hour = 25.92 Bq
- the DCs are (Table 11):
    - for 5 μm zirconia dust (Th : U = 1:1.25) is 0.0101 mSv/Bq
    - for 3 μm silica fume dust (Th : U = 1:4) is 0.0122 mSv/Bq
- the internal dose due to dust inhalation is: 37.44 Bq × 0.0101 mSv/Bq + 25.92 mSv/Bq × 0.0122 mSv/Bq = 0.378+ 0.316 = **0.69 mSv**.

Example 8

Members of the public can be potentially exposed to the dust from a rehabilitated mining and processing operation.

Monitoring results indicate that the dust contains, on average, 400 ppm uranium and 20 ppm thorium and an average alpha-activity level in the dust of 0.0045 Bq/m^3. The natural background concentration of alpha-activity in the dust prior to the commencement of operations was 0.0004 Bq/m^3.

It is estimated that in the worst case, a member of the public would spend 6,500 hours per year at this particular location. Assuming a BR of 0.96 m^3/hour, and a DC for 1 µm dust containing thorium to uranium in the weight ratio of 1:20 is 0.0145 mSv/Bq (Table 11).

The estimated possible internal exposure is calculated as follows:

- potential intake is: 6500 hours × 0.0045 Bq/m^3 × 0.96 m^3/hour = 28.08 Bq
- 'background' intake is: 6500 hours × 0.0004 Bq/m^3 × 0.96 m^3/hour = 2.50 Bq
- potential annual internal dose due to dust inhalation is:
  - (28.08 – 2.50) Bq × 0.0145 mSv/Bq = 0.37 mSv.

Example 9

Two workers work in a workplace where radon (^{222}Rn) measurements have been taken on a quarterly basis using radon cups. The monitoring results were as follows:

- 1^{st} quarter = 180 Bq/m^3
- 2^{nd} quarter = 250 Bq/m^3
- 3^{rd} quarter = 190 Bq/m^3
- 4^{th} quarter = 220 Bq/m^3
- Annual average = 210 Bq/m^3.

The first worker was employed for the whole year (2000 hours).

The second worker was at this site only during the first quarter (500 hours) and also worked for the rest of the year (1500 hours) in a workplace where thoron (^{220}Rn) concentrations were measured at 780 Bq/m^3. The site-specific F for thoron was established at 0.003, the F for radon is assumed to be at the default value, 0.4.

As individual dose records are required to be kept at the site, the background concentrations of radon and thoron were not taken into account in dose calculations.

For the first worker:

- the conversion factor is 4.45 × 10^{-3} mJh/m^3 per 1 Bq/m^3
  - 210 Bq/m^3 = 0.93 mJh/m^3
- using the DC from Table 18 (3.14 mSv per 1 mJh/m^3):
  - 0.93 mJh/m^3 × 3.14 mSv/[mJh/m^3] = 2.93 mSv.

For the second worker:

- exposure to radon (^{222}Rn):

$$P_{RnP} = 5.56 \times 10^{-6} \times t \times F_{RnP} \times C_{Rn} = 5.56 \times 10^{-6} \times 500 \times 0.4 \times 180 = 0.20\, mJh/m^3$$

Then, using the DC from Table 18 (3.14 mSv per 1 mJh/m^3):

- 0.20 mJh/m^3 × 3.14 mSv/[mJh/m^3] = 0.63 mSv.

- exposure to thoron (^{220}Rn):

$$P_{TnP} = 7.56 \times 10^{-5} \times t \times F_{TnP} \times C_{Tn} = 7.56 \times 10^{-5} \times 1500 \times 0.003 \times 650 = 0.22\ mJh/m^3$$

Then, using the DC from Table 18 (1.36 mSv per 1 mJh/m^3):

- 0.22 mJh/m^3 × 1.36 mSv/[mJh/m^3] = 0.30 mSv.

The total exposure due to inhalation of radon and thoron is 0.63 + 0.30 = **0.93 mSv/year**.

Example 10

The concentrations of NORs in drinking water extracted from ground water bores on site are: ^{226}Ra=0.267 Bq/L and ^{228}Ra=0.764 Bq/L. The consumption of drinking water by workers is estimated to be 600 L/year.

The concentration prior to the commencement of mining and processing operations in this water were: ^{226}Ra=0.107 Bq/L and ^{228}Ra=0.096 Bq/L.

Detailed analysis of drinking water indicates that only isotopes of radium are present in relatively significant quantities. The internal dose from ingestion of drinking water is calculated as follows:

Intake of radioactivity is calculated separately for different NORs:

- for ^{226}Ra = 0.267 Bq/L × 600 L = 160.2 Bq
- for ^{228}Ra = 0.764 Bq/L × 600 L = 458.4 Bq.

Dose is calculated separately for each NOR (Tables 19 and 20):

- for ^{226}Ra = 160.2 Bq × 0.00013 mSv/Bq = 0.021 mSv
- for ^{228}Ra = 382.0 Bq × 0.00034 mSv/Bq = 0.130 mSv.

Total dose from water ingestion is: 0.021 mSv + 0.130 mSv = 0.151 mSv.

Intake of radioactivity based on background' concentrations is also calculated separately:

- for ^{226}Ra = 0.107 Bq/L × 600 L = 64.2 Bq
- for ^{228}Ra = 0.096 Bq/L × 600 L = 57.6 Bq.

Dose is calculated separately for each NOR, as above:

- for ^{226}Ra = 64.2 Bq × 0.00013 mSv/Bq = 0.008 mSv
- for ^{228}Ra = 57.6 Bq × 0.00034 mSv/Bq = 0.020 mSv.

Total `background' dose is: 0.008 mSv + 0.020 mSv = 0.028 mSv.

Thus, the total dose from drinking water ingestion is 0.151 mSv – 0.028 mSv = **0.12 mSv**.

Example 11

Conditions

1. A worker was working in a mineral processing plant (Site A) for five months:
   - 520 hours as dry plant operator (average dust activity concentration = 0.305 Bq/m^3)
   - 300 hours as control room operator (average dust activity concentration = 0.027 Bq/m^3)
   - 140 hours as wet plant operator (average dust activity concentration = 0.066 Bq/m^3).

   Particle size characterisation program is not carried out. The mineral contains both thorium and uranium in an approximate ratio of Th:U = 25:1.

   The results of TLD badges worn by the worker for two monitoring periods are 0.19 and 0.28 mSv.

2. The worker spent two months in the year (380 hours) at a remote uranium exploration site (Site B):
   - the area in which the worker worked has a gamma-radiation level of 0.49 μSv/hour (background radiation level for this site is 0.14 μSv/hour)
   - drinking water was supplied from an on-site bore and ^{226}Ra concentration in the water was on average 0.72 Bq/L (background level is 0.15 Bq/L)
   - average dust activity concentration of uranium dust was 0.117 Bq/m^3 and on one occasion the worker was exposed to the dust with activity concentration of 3.692 Bq/m^3 for 10 hours (this was treated as a special exposure)
   - the average radon (^{222}Rn) concentration is 63 Bq/m^3.

3. The worker spent four months in the year (700 hours) working in a mineral storage area at the wharf (Site C):
   - 500 hours were spent working in the office
   - 200 hours were spent inside the product storage shed (dust activity concentration = 0.089 Bq/m^3, material contains both thorium and uranium in an approximate ratio of Th:U = 1:1.25), thoron concentrations were measured at 105 Bq/m^3 in the product storage shed and 25 Bq/m^3 inside the office
   - gamma-radiation level in the office was 0.32 μSv/hour, in the storage shed – 0.53 μSv/hour, background in the surrounding area – 0.16 μSv/hour.

Dose assessment

External dose 1 (Site A):

0.19 mSv + 0.28 mSv = 0.47 mSv

External dose 2 (Site B):

(0.49 μSv/hour × 380 hours) – (0.14 μSv/hour × 380 hours) = 133 μSv = 0.13 mSv

External dose 3 (Site C):

(0.32 μSv/hour × 500 hours) – (0.16 μSv/hour × 500 hours) = 80 μSv = 0.08 mSv

(0.53 μSv/hour × 200 hours) – (0.16 μSv/hour × 200 hours) = 74 μSv = 0.07 mSv

Sum of external doses

0.47 + 0.13 + 0.08 + 0.07 = 0.75 mSv

Internal dose 1 (Site A – inhalation – dust):

(0.305 Bq/m^3 × 1.2 m^3/hour × 520 hours) × 0.0158 mSv/Bq = 3.01 mSv

(0.027 Bq/m^3 × 1.2 m^3/hour × 300 hours) × 0.0158 mSv/Bq = 0.15 mSv

(0.066 Bq/m^3 × 1.2 m^3/hour × 140 hours) × 0.0158 mSv/Bq = 0.18 mSv

Internal dose 2 (Site B – inhalation – dust):

(0.117 Bq/m^3 × 1.2 m^3/hour × 380 hours) × 0.0084 mSv/Bq = 0.45 mSv

(3.692 Bq/m^3 × 1.2 m^3/hour × 10 hours] × 0.0084 mSv/Bq = 0.37 mSv

Internal dose 3 (Site B – inhalation – radon):

5.56 × 10^{-6} × 380 hours × 0.4 × 63 Bq/m^3 = 0.05 mJh/m^3

0.05 mJh/m^3 × 3.14 (mSv [mJh/m^3]) = 0.16 mSv

Internal dose 4 (Site B – ingestion):

(0.72 Bq/L × 125 L – 0.15 Bq/L × 125 L) × 0.00013 mSv/Bq = 0.01 mSv

Internal dose 5 (Site C – inhalation – dust):

(0.089 Bq/m^3 × 1.2 m^3/hour × 200 hours) × 0.0101 mSv/Bq = 0.22 mSv

Internal dose 6 (Site C – inhalation – thoron):

7.56 × 10^{-5} × 200 hours × 0.04 × 105 Bq/m^3 = 0.06 mJh/m^3

0.06 mJh/m^3 × 1.5 (mSv/[mJh/m^3]) = 0.09 mSv

7.56 × 10^{-5} × 500 hours × 0.04 × 25 Bq/m^3 = 0.04 mJh/m^3

0.04 mJh/m^3 × 1.36 (mSv/[mJh/m^3]) = 0.05 mSv

Sum of internal doses

3.01 + 0.15 + 0.18 + 0.45 + 0.37 + 0.16 + 0.01 + 0.22 + 0.09 + 0.05 = 4.69 mSv

The annual radiation exposure of the worker is estimated to be:

0.75 + 4.69 = **5.44 mSv**

Example 12

Conditions

There is a possibility of an industrial or residential development to be established on a rehabilitated processing site. The gamma-dose rate is 0.19±0.02 μSv/hour; background in the area was 0.13±0.02 μSv/hour.

The dust activity concentration is the same as it was prior to the construction of a plant. Some tailings have been buried on site and concentrations of ^{226}Ra in the ground water are slightly elevated (0.45 Bq/L in comparison with background value of 0.22 Bq/L). Modelling indicates that if tailings are brought to the surface, concentration of radon in the air is expected to be of the order of 18 Bq/m^3.

Residential development case

Potential external dose (Case 1 – exact background value):

[0.19 μSv/hour – 0.13 μSv/hour) × 8760 hours = 0.526 mSv

Potential external dose (Case 2 – background value plus two GSDs):

(0.19 μSv/hour – 0.17 μSv/hour) × 8760 hours = 0.175 mSv

Potential internal dose – ingestion:

(0.45 Bq/L – 0.22 Bq/L) × 500 L × 0.00013 mSv/Bq = 0.015 mSv

Potential internal dose – inhalation:

5.56 × 10^{-6} × 8760 hours × 0.2 × 18 Bq/m^3 = 0.175 mJh/m^3

0.17 mJh/m^3 × 3.14 (mSv [mJh/m^3]) = 0.534 mSv

The determination of whether exposure to radon is taken into account in the assessment of the site or not is made by the regulator. In this case it is assumed that the exposure is excluded from dose calculations for members of the general public, as it is close to general background levels in WA.

The potential exposure level for the member of the general public in the area is:

- 0.19 mSv/year if the use of the background level plus two GSDs was approved by the regulator
- 0.54 mSv/year if the use of the “background level plus two GSDs” was not approved.

In the first case, the site does not need to be classified in any way and any development of the land will be permissible. However, in the second case, the exposure of a member of the public is above the dose constraint of 0.3 mSv/year, at which a classification of a site as ‘radiologically contaminated’ may be required.

Industrial development case

Potential external dose:

[0.19 μSv/hour – 0.13 μSv/hour) × 2000 hours = 0.120 mSv

Potential internal dose – ingestion:

(0.45 Bq/L – 0.22 Bq/L) × 500 L × 0.00013 mSv/Bq = 0.015 mSv

Potential internal dose – inhalation:

$5.56 \times 10^{-6} \times 2000$ hours $\times 0.4 \times 18$ Bq/m^3 = 0.08 mJh/m^3

0.08 mJh/m^3 × 3.14 (mSv [mJh/m^3]) = 0.251 mSv

It is estimated that an industrial worker will receive a dose of approximately 0.39 mSv/year, which may require the classification of the site as ‘radiologically contaminated’.

Appendix 4: Glossary and Abbreviations

ADWG	*Australian Drinking Water Guidelines*
AED	Aerodynamic equivalent diameters
AMAD	Activity median aerodynamic diameter
ARPANSA	Australian Radiation Protection And Nuclear Safety Agency
BR	Breathing rate
DC	Dose coefficient
DCF	Dose conversion factor
Designated worker	A worker who works, or may work, under conditions so that the effective dose of radiation the worker receives may exceed 5 millisievert per year (r. 641S(1)(a))
EEC	Equilibrium equivalent concentrates
F	Equilibrium factor
GM	Geometric mean
GSD	Geometric standard deviation
IAEA	International Atomic Energy Agency
ICRP	International Commission On Radiological Protection
MAD	Median aerodynamic diameter
MDL	Minimum detectable limit
MMAD	Mass median aerodynamic diameter
NOR	Naturally occurring radionuclide
OSL	Optically stimulated luminescence
PAEC	Potential alpha energy concentration
PED	Personal electronic dosimeter
Pre-operational monitoring program	A program submitted to the regulator for the mining operation for the monitoring of radiation and dose levels at the mine (r. 641M(1))
Radioactive material	Material that has an activity concentration that exceeds 1 Bqg-1 and either: • exhibits radioactivity; or (ii) • emits ionising radiation or particles, or • contains radionuclides of natural origin (r. 641K)

Relevant mines	A mine is considered a relevant mine if minerals or radioactive materials that have an activity concentration of radioactivity of 1 Bqg-1 or more are mined at the mine, and either: • workers at the mine are likely to receive doses of radiation over 1 millisievert per year, arising from mining operations at the mine, or • members of the public at, or in the vicinity of, the mine are likely to receive doses of radiation, arising from mining operations at the mine, in excess of one-half of the effective dose as above (r. 641L)
RnP	Radon progeny
RPS-9	The *Radiation Protection Series 9 – Code of practice and safety guide for radiation protection and radiative waste management in mining and mineral processing (2005)* published by the Chief Executive officer of the Australian Radiation protection and Nuclear Safety Agency in August 2005 and updated in January and December 2015
SEG	Similar exposure group
TLD	Thermo-luminescent dosimeter
TnP	Thoron progeny
WLM	Working level month

Radionuclides	
^{40}K	Potassium-40
^{210}Pb	Lead-210
^{210}Po	Polonium-210
^{228}Ra and ^{226}Ra	Radium-228 and Radium-226
^{87}Rb	Rubidium-87
^{222}Rn ^{220}Rn	Radon-222 and Radon-220
^{147}Sm	Samarium-147
^{238}U and ^{235}U	Uranium-238 and Uranium-235
^{232}Th	Thorium-232

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