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

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

NORM-V Dose assessment

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1. Purpose

To provide guidance on the methods for the assessment of radiation doses received by employees and members of the general public, arising from exposure to naturally occurring radioactive materials (NORM).

This Guideline updates the previous version (NORM-5) by incorporating the revised Dose Coefficients for members of the ^{238}U, ^{235}U and ^{232}Th decay series as published in the International Commission for Radiological Protection (ICRP) publications 137 [1] and 141 [2].

2. Scope

The Mines Safety Directorate has embraced a risk-based approach to regulating the potential radiation exposures of Western Australian mine workers. For the purposes of the suite of NORM Guidelines:

(a) A substance that has a head of decay chain (^{232}Th, ^{238}U or a combination of ^{232}Th and ^{238}U) activity concentration > 1 Bq/g is deemed as radioactive material; and

(b) Mining operations, as defined in the applicable WA mine safety framework (referenced as "the regulations") that encounter radioactive materials, but can demonstrate that annual doses to workers and members of the public are consistently less than 1 mSv are able to be exempted from complying with radiation regulations in all, or part of their operations.

However, exploration, mining, or mineral processing operations in Western Australia (WA) that encounter, use, handle, store or dispose of, naturally occurring radioactive material (NORM) and that cannot demonstrate that workers or members of the public will receive annual doses of less than 1 mSv, come within the scope of the regulations. Hereinafter, these mining operations are referenced as 'reporting entities'.

3. Legislative status

Although this document is for general guidance, it is an expectation of the State mining engineer that reporting entities will:

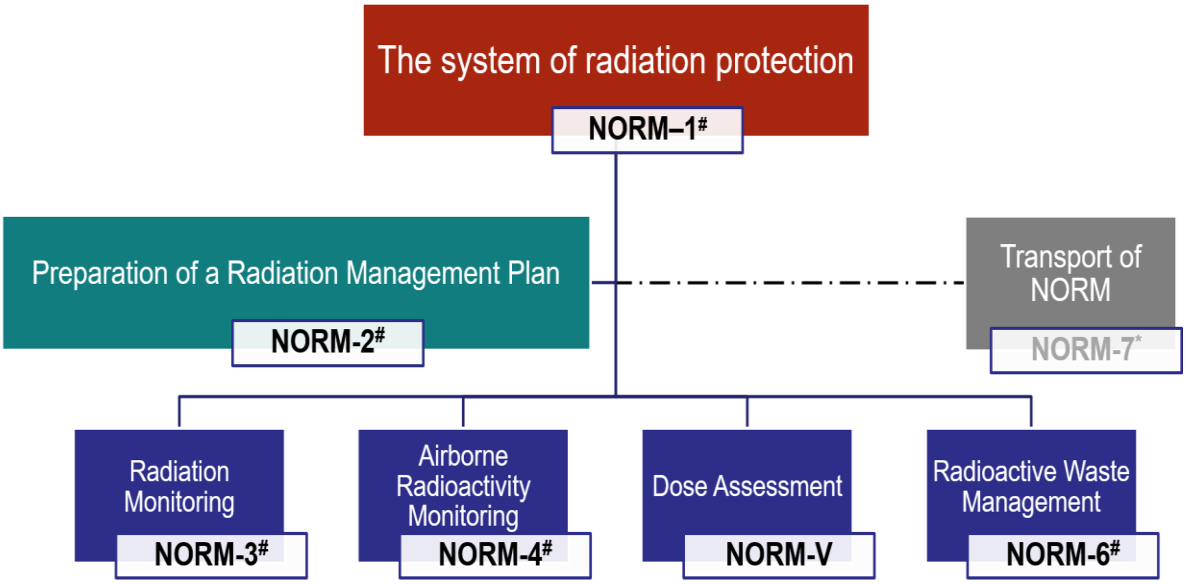
(a) Make commitments in their Radiation Management Plan to apply relevant Sections of the Guideline for the evaluation and control of worker exposures to NORM ; and

(b) Apply the methods, calculations and dose coefficients in this Guideline for the purpose of assessment of doses.

4. Relationship to other NORM guidelines

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

Note that other Guidelines 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 Guideline details the dose assessment procedures to be used when assessing radiation exposure from NORM in WA reporting entities, and is based on the recommendations in relevant documents produced by the International Commission on Radiological Protection (ICRP), International Atomic Energy Association (IAEA) and Australian Radiation Protection and Nuclear Safety Agency (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 radionuclides,
- Inhalation of radon (both ^{222}Rn and ^{220}Rn) and their 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 reporting entity, justification must be provided for it to be not considered in the dose assessment process.

5.2. Baseline monitoring program

In accordance with Regulation 16.6 of the regulations, the State mining engineer is to be provided with the results of an approved environmental radiation monitoring programme (hereinafter referred to as ‘baseline’ data) 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*[1].

[1] Currently under review and will be renumbered as NORM-III in subsequent editions.

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 State mining engineer, 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

In accordance with Regulation 16.23 (1) (a) of the regulations, doses due to natural background are not to be considered when assessing the dose received by a worker or a member of the public.

It is important to note that the regulations reflect the national and international approaches to accounting for the contribution of natural background sources of radiation to worker exposures, [9, pp 14] viz:

> *"Exposure to radiation from natural sources is generally excluded from occupational or public exposure, except when the exposure is a direct consequence of a practice or is specifically identified by the appropriate authority as requiring control through the implementation of a program of radiation protection"; and*
>
> *"Dose limits do not apply to exposures from natural sources, except as described above, or to medical exposures."*

The one exception to this protocol occurs when assessing the dose delivered via inhalation of ^{222}Rn or ^{220}Rn and their decay products. The natural variation in concentrations of these sources of exposure are such that it is difficult to determine, with any precision, a "typical" background contribution to dose. As is discussed in Section 8.4, 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.

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 (SEG's) or Critical Groups2, the value attributed to natural background for each group is required to be stated; and
- 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.

[2] 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 devices3, gamma radiation survey results or a combination of both. The methods used are dependent on the person or group that is being assessed.

6.1. Designated employees (DEs)

The regulations defines a designated employee as a person who works, or *may work* in conditions that result in an annual effective dose greater than 5 millisieverts (mSv) per year.

DEs are monitored more intensively than non-designated employees (including, where appropriate, by personal monitoring) and their doses are assessed individually [12, p4].

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

6.1.1. Pregnant employees

In accordance with Regulation 16.22 (2) (a) of the regulations, once an employee learns that they are pregnant, they are not permitted to be a designated employee.

6.1.2. Designated employees: monitoring of external dose

The monitors used may be of a passive type, such as an OSL (Optically Stimulated Luminescence) dosimeter or TLD (thermo-luminescent dosimeter), or an active type, such as a PED (personal electronic dosimeter).

An external supplier typically provides personal badges (OSL's and/or TLD's) and the list of personal monitoring services approved for use in Western Australia is available from the WA Radiological Council4.

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 PED's can be used in conjunction with passive monitors (not as a substitute).

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

A comparison to the employee effective radiation dose limit should be provided. As per the regulations, the applicable limits are 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.

6.2. Non-designated employees

As per RPS 9.1 [12], "non-designated employees will be monitored less intensively than DEs; 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 [SEG]".

[3] As approved by the Radiological Council.

[4] A list of approved providers is available from PRMS_requirements (radiologicalcouncil.wa.gov.au)

Therefore, a cross-section of non-designated employees should be included in a monitoring program. This cross section should include employees from different SEGs with sufficient data collected for each SEG to be representative of the general exposure scenarios.

- All members of a SEG must have similar exposure to NORM's and work hours but may have very different roles or occupations;
- The validity of the dose assessment methodology should be periodically confirmed.

6.2.1. Non-designated employees: 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 (refer to Footnote 3).

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 B for calculations of dose to a SEG.

A comparison to the designated employee effective radiation dose "threshold" level of 5 mSv per year should be provided. As per section 6.6 below, the 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.

6.3. Other employees, Critical Groups[2] and members of the general public

The Registered Manager 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 (ie levels above background levels) exceeding the public effective dose limit.

In accordance with Regulation 16.4 (1) and (3) the State mining engineer may 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[2] or members of the public are potentially exposed to sources of radiation from more than one mining operation5.

The external gamma radiation exposures for other employees 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 State mining engineer imposed authorised limit (if applicable) should be provided.

6.4. 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.

6.4.1. Designated employees

The external radiation dose component is obtained directly from the results of the personal monitoring program. Further information is available in Guideline NORM-3[1] Radiation monitoring programs.

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

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

[5] Compliance with any discharge limits that may be imposed under the Radiation Safety Act is also required.

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 employees (see Section 6.4.2).

6.4.2. Non-designated employees, Critical Groups[2] 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 employee, 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;

- If an employee is at work for 40 hours per week, a default value of 2000 hours exposure per year can be applied.
- Calculations of exposure hours must be made for employees that have rosters significantly different to 8 hours per day for 5 days per week:
  - For example a person on a 2week on 1 week off roster with 12hr shifts will work approximately 2912 hours per year. When calculating work hours:
    - meal breaks and shower and change time should be included as work time; and
    - 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.
  - Shift rosters should be provided as part of the assessment.
- 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 microsieverts per hour (µSv/h) is to be used.

The calculation methods are illustrated in Examples 3 and 4 in Appendix B.

6.5. External gamma radiation readings

As detailed Guideline NORM-3[1] (Survey instruments and data interpretation), there is a vast variety 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 Rem = 1 Gray = 1 Sievert[6].

[6] 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.

6.6. 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:

- 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 minimum detection limit, 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 the particle's:

- Size, expressed as activity median aerodynamic diameter (AMAD) in microns (µm);
- Activity; and
- 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 if it can pass into blood and be preferentially deposited in target organs. For this reason the aerodynamic behaviour of the inhaled particles inhaled 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; and
- 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 radionuclide. Internationally accepted DC's are published by the ICRP and IAEA. The ICRP Occupational Intake of Radionuclides Data Viewer [25] is a valuable source of DCs.

The latest publications of the International Commission on Radiological Protection [1 – 4] provide the necessary DCs for inhalation of radionuclides. This data is summarised in Tables A.1 – A.10, for the radionuclides in the ^{238}U, ^{235}U and ^{232}Th decay chains.

- ICRP-130 [3] 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.
- ICRP-66 [5], 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 radionuclides, such as occurs in mineral dusts, where the radionuclides are in secular equilibrium, the dose coefficients 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 A1 to A10;
- 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 A.11.

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

- Please note that data in this Table is for preliminary assessment only, and regular analysis of the Th : U ratio (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 A.12, the dose coefficient values in Table A.11 shall be used.

7.4. AMAD and Dose Coefficients for fumes

The Dose Coefficients 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 radionuclides 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 A.13 contains the dose coefficients for dusts containing these radionuclides 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 naturally occurring radionuclides may be required. The values from the IAEA International Basic Safety Standards (IAEA, 2014) [6] are provided in Table A.14. The latest ICRP Publications [1 – 4] do not contain the dose coefficients for the inhalation of these radionuclides.

7.6. Additional AMAD dose coefficients 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 (15% of samarium is ^{147}Sm) may be required. The dose coefficients for ^{147}Sm are also provided in Table A.14, from the IAEA International Basic Safety Standards [6].

7.7. Particle sizing and AMAD determination

Further information and techniques for determining particle size distribution, MMAD and AMAD are covered in the 2021 version of the DMIRS Guideline NORM-3[1] in the section "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 therefore the activity of each radionuclide does 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;

- Any chemical processing of the material, such as leaching or adding flotation agents to the process;
- 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 radionuclides can volatilise and disrupt the equilibrium; 250-300ºC is suggested as a general guide at which additional analysis of the material may be required.
- 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.
- When scale or sludge precipitates or deposits e.g.
  - 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),
  - Processing and storage of natural gas, replacement of stack filters at different smelters (^{210}Pb and ^{210}Po films),
  - Oil exploration and production (^{226}Ra and ^{228}Ra scales),
  - Geothermal energy generation and hydraulic fracturing (^{226}Ra and ^{228}Ra in scales and process water).

Please refer to the Tables A.1 – A.10 for the values for individual radionuclides.

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 naturally occurring radionuclides in the ore, mineral or concentrate. The correct conversion coefficients for such calculations are provided in Table A.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 Guideline NORM-4[7] dealing with the different aspects of airborne radioactivity monitoring.

In accordance with Regulation 16.23(2)(b) of the 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:

1. A comprehensive and an auditable respiratory protection program has been established at the site in accordance with two relevant Australian Standards [7,8];
2. The respiratory protection program has been fully assessed by DMIRS and a written authorisation has been issued 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 State mining engineer, 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 dose coefficient (or DCF) to calculate the annual dose (mSv).

7.9.1. Worker Dose Assessment

Internal dose assessment for the workforce is determined by;

1. Grouping personnel with similar exposure profiles into SEGs

[7] Currently under review and will be renumbered as NORM-IV in subsequent editions.

- note that SEGs may contain employees from different occupation groups if their exposure profiles are similar

2. Using the results of the personal air monitoring program, 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, estimate the working hours that the average gross alpha activity concentration results apply. (HW_i).
4. Determine whether the task is light to moderate activity or heavy physical work, and determine the appropriate Breathing Rate (BR). The breathing rate 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 Dose Coefficient (DC), in units of mSv/Bq based on the AMAD and Th : U ratio of the dust in the air samples, and other applicable radionuclides.
6. Calculating the internal dose by adding all the task intakes and multiplying by the applicable DC as follows:

$$\boldsymbol{Individual\ internal\ dose} = \sum_i (\boldsymbol{AV_i} \times \boldsymbol{HW_i} \times \boldsymbol{BR}) \times \boldsymbol{DC_i}$$ **w**here:

AV_i – average of gross alpha-activity concentration for the work task *i* (Bq/m^3),

HW_i – hours worked by an employee for the work task *i*,

BR – assumed breathing rate of an employee (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),

DC_i – dose coefficient in mSv/Bq for different AMAD sizes, different uranium and thorium weight ratios and other radionuclides (all data is available in Appendix A).

7.9.2. Critical Group[2] Dose Assessment

A similar equation can be used for the assessment of internal exposures for the members of the general public with adjustment to breathing rate and DC. This assumes that the sampling results are of sufficient duration and frequency to be representative of annual exposures. If there are significant variations likely such as prevailing winds at certain times of the year then the summation method above may be required:

$$\boldsymbol{Individual\ internal\ dose} = \boldsymbol{AV} \times \boldsymbol{H} \times \boldsymbol{BR} \times \boldsymbol{DC}$$ 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),

$(AV \times H \times BR)$ – personal intake for the period of assessment for the member of the public (Bq), and

DC – dose coefficient in mSv/Bq for different uranium and thorium weight ratios (Tables in Appendix A, AMAD is always taken as 1 µm for members of the public).

7.10. 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 radionuclides 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 A16 and A17 in Appendix A.

(See Norm-3[1] 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 radionuclides, 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 radionuclides 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; and
- 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; and
- ^{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 dose delivered by this exposure pathway arises 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.

- Where it is used, it must be accompanied by an assumption as to the F that has been applied; and
- 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 (PAE) 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 thoron progeny must be determined. The PAEC is measured in mJh/m^3.

PAEC is determined from measurements of the potential alpha energy concentration in the air and the volume of air inhaled by using the following formulae (derived from Table A.1 of ICRP-137 [1] and paragraph 15 of ICRP-65 [11])

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:

P_{RnP}, P_{TnP} are the potential alpha energy concentrations of radon decay products and thoron decay products, respectively (mJh/m^3),

5.56 x 10^{-6} is the combined potential alpha energy concentration for the RnP series from ^{218}Po to ^{214}Po (mJ/Bq),

7.56 x 10^{-5} is the combined potential alpha energy concentration for the TnP series from ^{216}Po to ^{212}Po (mJ/Bq),

t is the exposure time (hours),

F_{RnP} is the Equilibrium Factor between radon and RnP (typically taken as 0.4 for indoor areas and 0.2 for outdoors),

F_{TnP} is the Equilibrium Factor between thoron and TnP (typically taken as 0.04 for indoor areas and 0.004 for outdoors [1]),

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

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

It is important to note that equilibrium factors (F) for both radon (^{222}Rn) and thoron (^{220}Rn) and their respective decay progeny can be highly variable. If directed by the State mining engineer 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.

- 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 <u>initial</u> estimates of potential exposure.
- In the absence of site-specific equilibrium factors the defaults listed above should be used [8].

8.2. Calculating Dose from EEC_{Rn} or EEC_{Tn} 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:

- 1 WLM = 3.54 mJh/m^3; and by extension,
- 1 mJh/m^3 = 0.282 WLM [1].

Potential alpha energy exposures to radon progeny and thoron progeny 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 and/or TnP (EEC_{Rn} and EEC_{Tn}) are conducted, in an indoor workplace a calculation of exposure from the inhalation of radon and thoron progeny depends on time of the exposure t and is carried out as follows:

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

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

Where, (see Table A.11 of [1]):

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

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 A.11 of [1] provides some guidance:

- 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); and
- 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 [25]:

- for radon (^{222}Rn) = 3.14 mSv per mJh/m^3 ;
- for thoron (^{220}Rn) = 1.36 mSv per mJh/m^3

[8] It is important to note that in the part 4.2.3 of the ARPANSA RPS-9.1 [12] the equilibrium factor 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, thus more practical values of 0.04 for indoors and 0.004 for the outdoors are adopted.

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

- for radon (^{222}Rn) = 5.59 mSv per mJh/m^3 ;
- 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 breathing rate.

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

8.4. Contribution from Background

Determining the contribution from natural background is problematic, and has resulted in ARPANSA issuing guidance that when calculating doses from inhalation of radon and radon progeny, no subtraction of background exposure is to be made in areas for which individual dose records are required to be kept [9][12].

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] As per [12], 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.

9. Internal radiation exposure (ingestion of drinking water)

All drinking water supplied on mining operations shall meet the Australian Drinking Water Guidelines (ADWG) (2018) [13]. As well as chemical and biological testing it must also be screened (as a minimum) for gross alpha and gross beta activity concentrations [10].

The naturally occurring radioisotope potassium-40 (^{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 (2018) [13] 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 (2018) [13], and should be regularly tested by the supplying authority and confirmed as compliant.

As such, radiation exposure to employees 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 principal employer and the radiation safety officer to ensure that sufficient testing occurs to be able to verify that drinking water meets the ADWG (2018) [13].

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 radionuclides must be determined.

The screening levels as defined in Information Sheet 2.2 provided as a supplement to Chapter 7 of the ADWG (2018) [13] shall apply, namely;

- Gross alpha-activity of 0.5 Bq/L and;
- 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, they are the concentrations that, if exceeded, screening the requirement to provide additional information, conduct further investigation to determine the risk to the workforce, public or ecosystems and take appropriate actions.

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 radionuclides 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 radionuclides must be identified and their activity concentrations determined.

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

[10] Copies of the results of radionuclide analyses of drinking water samples are to be provided to the Water Unit of the WA Department of Health.

- Thorium decay chain: ^{232}Th, ^{228}Ra, ^{228}Th (possibly also ^{224}Ra), and
- 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 radionuclides must be considered when an assessment of radiation exposures from ingestion of water is made.

The contribution to radiation dose for each radionuclide 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 radionuclide. 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.

- The level of water consumption is generally assumed to be 2 litres per day equivalent to 730L litres per year (ADWG (2018)) [13].
  - If consumption is significantly greater than 2 litres per day, then adjustments must be made.
- The 'natural background' concentrations of radionuclides in local drinking water are typically determined during the site 'baseline' survey (prior to the commencement of operations) for the radionuclides 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).
  - In the event where baseline data does not exist, the contribution from natural background levels is assumed to be zero.
- Dose coefficients or the dose per unit intake and other relevant data are presented in Tables A.19 and A.20.

For some operations, ingestion dose coefficients of ^{40}K, and ^{87}Rb (present in some lithium ores) and/or ^{147}Sm (present in Monazite) may be need to be assessed. These are provided in Table A.21 and are based on the 2014 IAEA Basic Safety Standards [6].

An example of calculations in given in Example 10 in Appendix B.

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 radionuclides, other than ingestion of water by a mine worker, is unlikely except in very rare scenarios.

The risk of ingestion of radionuclides 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 dose coefficients for different radionuclides are provided in Tables A19-21.

- The assessment can be quite a complex task.
- For more information please refer to the guides and reports issued by the International Atomic Energy Agency [14–23].
  - 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 B.

Appendix A – Dose coefficients (inhalation and ingestion) for Naturally Occurring Radionuclides

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

Table A.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 (dose coefficient):

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

Table A.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	$3.1 \times$-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 (dose coefficient):

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

Table A.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 (dose coefficient):

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

Table A.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)		Dose (Sv)
			Alpha	Beta	
^{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 (dose coefficient):

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

Table A.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 (dose coefficient):

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

Table A.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**	**1.19 × 10–4**

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

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

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

Table A.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) Alpha	Quantity inhaled (Bq) Beta	Dose (Sv)
^{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 (dose coefficient):

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

Table A.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 (dose coefficient):

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

Table A.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 (dose coefficient):

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

Table A.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**	**1.86 × 10–5**

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

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

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

Table A.11: Dose Conversion Factors (DCF), 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
25:1	0.0274	0.0214	0.0158	0.0085	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
1.25:1	0.0186	0.0146	0.0108	0.0059	0.0028
1:1	0.0179	0.0141	0.0104	0.0057	0.0027
1:1.25	0.0174	0.0136	0.0101	0.0055	0.0026
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
1:25	0.0145	0.0114	0.0085	0.0047	0.0022

Th : U weight ratio	Dose coefficient (mSv/Bq$_\alpha$), for an AMAD of:				
	1 µm	**3 µm**	**5 µm**	**10 µm**	**20 µm**
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 A.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)	Dose coefficient (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 A.13: Dose Coefficients for the inhalation of ^{210}Po and ^{210}Pb dusts

Radionuclide	Slowest lung absorption class	Dose coefficient (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 A.14: Dose Coefficients for the inhalation of dust containing ^{40}K and ^{87}Rb

Radionuclide	Slowest lung absorption class	Dose coefficient (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 A.15: Coefficients between content of radionuclides 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; and
- 15% of samarium is ^{147}Sm

Table A.16: Alpha activities and correction factors for thorium ore dust residing on an air sampling filter (reproduced from IAEA RS-G-1.6, [24])

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

Please note that IAEA Publication RS-G-1.6 [24] was superseded in 2018 by the new Occupational Radiation Protection Guide [22], however the table above was omitted from the latter document.

Table A.17: Alpha activities and correction factors for thorium ore dust residing on an air sampling filter (reproduced from IAEA RS-G-1.6, [24])

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

Please note that IAEA Publication RS-G-1.6 [24] was superseded in 2018 by the new Occupational Radiation Protection Guide [22], however the table above was omitted from the latter document.

Table A.18: Dose coefficients for the inhalation of radon (^{222}Rn) and thoron (^{220}Rn) [25]

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 dose coefficient of 6 mSv per mJh/m^3

Table A.19: Dose coefficients for the ingestion of radionuclides from thorium decay chain

Radionuclide	Dose coefficient (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 A.20: Dose coefficients for the ingestion of radionuclides 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 A.21: Dose coefficients for the ingestion of ^{40}K, ^{87}Rb and ^{147}Sm

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

Appendix B – Calculation examples

Example 1

An employee 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 (minimum detection level) 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

An employee is a part of a 'similar exposure group' (SEG) of workers consisting of 12 people. Although the employee was not individually monitored for external radiation exposure, five other employees in the SEG were monitored with OSL badges.

The annual average for an employee in this work category is 0.27 mSv.

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

Example 3

An employee works in the office but occasionally visits a production area where NORMs are processed. The average dose rates are: in the employee'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 employee:

- 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 = <u>0.32 mSv</u>.

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 = <u>0.27 mSv</u>.

Example 5

An employee worked for three months in a plant where thorium-containing mineral is processed. The employee 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 employee 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 breathing rate was assumed to have been 1.6 m^3/hour.

Assuming a default particle size value of 5 µm (using a dose coefficient of 0.0167 mSv/Bq taken from Table A.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 musculo-skeletal 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 once-off exercise it is decided to not apply to the State mining engineer 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 × 27 Bq/m^3 × 1.2 m^3/hour = 84 Bq,
- The dose coefficient for 5 µm dust containing ^{226}Ra is 0.0130 mSv/Bq (Table A.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 dose coefficients are (Table A.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 breathing rate of 0.96 m^3/hour, and a dose coefficient for 1 µm dust containing thorium to uranium in the weight ratio of 1:20 is 0.0145 mSv/Bq (Table A.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 employees 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:

- 1st quarter = 180 Bq/m^3;
- 2nd quarter = 250 Bq/m^3;
- 3rd quarter = 190 Bq/m^3;
- 4th 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 equilibrium factor for thoron was established at 0.003, the equilibrium factor 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 dose coefficient from Table A.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 dose coefficient from Table A.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 dose coefficient from Table A.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 radionuclides 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 employees 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 radionuclides:

- 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 radionuclide (Tables A.19 and A.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 radionuclide, 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

i. An employee 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 employee for two monitoring periods are 0.19 and 0.28 mSv.

ii. The employee spent two months in the year (380 hours) at a remote uranium exploration site (Site B).

- The area in which the employee 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 employee 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.

iii. The employee 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;

The sum of external doses

$0.47 + 0.13 + 0.08 + 0.07 = 0.75$ mSv

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

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

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

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

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

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

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

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

$5.56 \times 10^{-6} \times 380\ hours \times 0.4 \times 63\ Bq/m^3 = 0.05\ mJh/m^3$;

$0.05\ mJh/m^3 \times 3.14\ (mSv\ [mJh/m^3]) = 0.16\ mSv$

Internal dose 4 (Site B – ingestion):

$(0.72\ Bq/L \times 125\ L - 0.15\ Bq/L \times 125\ L) \times 0.00013\ mSv/Bq = 0.01\ mSv$

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

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

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

$7.56 \times 10^{-5} \times 200\ hours \times 0.04 \times 105\ Bq/m^3 = 0.06\ mJh/m^3$;

$0.06\ mJh/m^3 \times 1.5\ (mSv/[mJh/m^3]) = 0.09\ mSv$

$7.56 \times 10^{-5} \times 500\ hours \times 0.04 \times 25\ Bq/m^3 = 0.04\ mJh/m^3$;

$0.04\ mJh/m^3 \times 1.36\ (mSv/[mJh/m^3]) = 0.05\ mSv$

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 employee 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). The modelling indicates that if tailings are brought to the surface, concentration of radon in the air is expected to be around 18 Bq/m^3.

i. 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 if exposure to radon is taken into account in the assessment of the site or not is made by DMIRS, in our 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:

(a) 0.19 mSv/year if the use of the background level plus two GSDs was approved by DMIRS, and

(b) 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 (please see NORM-6, Management of radioactive waste for more information).

ii. 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 × 10^{-6} × 2000 hours × 0.4 × 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’.

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