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CHAPTER 14. OPERATIONAL HEALTH PHYSICS ASPECTS OF AIRBORNE
AND SURFACE CONTAMINATION

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ABSTRACT

Part 1 describes the derivation and use of control standards for airborne and surface contamination. The concept of derived limits is developed and sample calculations are presented. Derived limits for uranium and thorium are derived.

Part 2 presents a range of methods of measuring contamination with some discussion of their relative merits.

INTRODUCTION

When considering air or surface contamination in an operational health physics context it is important to emphasise the primary objective of all operational health physics - to control the radiation exposure of the worker to an acceptably low level while facilitating the particular operation. In current jargon this is ALARA (as low as reasonably achievable), although the same philosophy has been applied informally for years by operational health physicists.

Measurement of air and surface contamination is part of the optimisation and limitation procedures of ALARA. The measurements are only one part, albeit a major part, of a decision process that seeks to control worker exposure while, if possible, not preventing the operation. As part of the decision process the accuracy of the measurement is often relatively unimportant. Speed, simplicity and reliability may all be more important than a high degree of accuracy. After measurements have been made, there is a need to decide what to do - and this can be optimised. For example, sometimes

decontamination has been carried out because contamination has been detected and removing it is considered to be 'a good thing' (cleanliness is next to godliness). One should always ask why the decontamination is being carried out. If it produces a small reduction of exposure of one group of workers at the cost of greater exposure to the workers carrying out the decontamination then it would not be optimised. Even if there were an overall reduction of exposure, the size of this reduction should be balanced against the resources expended to effect the reduction and the opportunity cost of those resources.

While this lecture is part of a course for the uranium and thorium industries, the approach taken is to cover the subject in a generalised manner first and then highlight special aspects of these industries. Part 1 of this lecture discusses the basis of contamination control standards, and Part 2 discusses the measurement of contamination.

PART 1 - CONTAMINATION CONTROL STANDARDS

Derived Limits

There are many circumstances in which measurements made in a radiation protection program cannot be related directly to the primary dose-equivalent limits (DELS) recommended by the International Commission on Radiological Protection (ICRP). It is then useful to define a simple, artificial model of the situation. This model can be used to derive a standard which is directly comparable with the measurements to be made. This standard is the derived limit (DL). The derived air concentrations (DACs) published by ICRP are such DLs. Derived limits have in the past been called derived working levels (DWLs) and the draft guideline on dose assessment to the 1987 Code (DASETT 1987) uses the terms 'authorised limit' and 'reference level'.

Derived limits relate to a model and not to real life: the usefulness of the derived limit and, more particularly, the limitations which must be imposed on its use, then depend on how closely the model represents the particular situation. In practice, most models seem to fall into one of two broad classes. In one of these, the model is a representation of the real world, but only in certain specialised circumstances which have to be carefully defined. In these circumstances, the derived limit corresponds quite closely with the dose-equivalent limit, in other circumstances, this derived limit is inapplicable. At the other extreme, the model may be very generalised so that the resultant working limit can be applied over a wide range of circumstances. In this case, however, the quantitative relationship

to the ICRP recommendations will be very much less precise and it will probably be true to say only that operating at the DL will result in doses below the ICRP recommended limit, but by an amount which cannot be forecast. The second class of model is the more common, and most ICRP and other recommended derived limits fall into this class. The 1987 revision of the Code of Practice (DASETT 1987) has avoided the use of derived limits, but they do appear in Guidelines to the Code.

Examples of derived limits of great specificity (i.e. of the first class above) can be found in the public health field. When a radioactive material is released to the environment, there is usually one principal pathway by which the radioactivity reaches man, giving rise to higher radiation doses to a particular group than to the population at large. This pathway and the exposed group are usually called the Critical Pathway and Critical Group. ICRP recommends dose limits that should be applied in these circumstances to the critical group (ICRP 1984), and it is often convenient to define a point in the critical pathway where measurements will be made and to derive a DL for these measurements. A calculation can also be made of the corresponding discharge rate and a DL of discharge rate calculated. Both these DLs could correspond quite closely with the ICRP dose equivalent limit for the critical group.

Perhaps the most generally used and least clearly defined derived limit is that for surface contamination. Surface contamination can give rise to resuspended activity, which may be inhaled, it may be transferred to skin and subsequently ingested, and it may be a source of external radiation while remaining on the skin. The level of surface contamination can also be thought of as a retrospective measure of the degree of success of contamination control in an area. Any choice of a standard must thus be somewhat arbitrary. The figures in common use (see Table 1) have been obtained from a deliberately over-simplified model so that they can be applied to a wide range of circumstances (i.e. they are of the second class above). In any particular situation, their link with the ICRP recommended DELs is tenuous. All that can be said is that working below these derived limits is unlikely to give rise to airborne activity or skin radiation dose rates sufficient to cause ICRP recommendations to be infringed. Nevertheless, it is possible to envisage circumstances where this statement would be untrue and the DL for surface contamination has consequently to be applied with judgement. The values that have been in general use until recently were based on ICRP2 (ICRP 1959) and the impact of ICRP30 (ICRP 1979) is only now being felt.

Table 1Maximum Levels of Surface Contamination in 'Active' areas:1 Commonly used values based on ICRP2⁽¹⁾

Alpha emitters of high toxicity ⁽²⁾	3.7 Bq/cm ²
All other alpha emitters	37 Bq/cm ²
Beta emitters, of maximum energy greater than 0.2 MeV	37 Bq/cm ²
Beta emitters, of maximum energy less than 0.2 MeV	370 Bq/cm ²

2 Recommended values based on ICRP30⁽³⁾

Class I (4)	0.3 Bq/cm ²
Class II	3 Bq/cm ²
Class III	30 Bq/cm ²
Class IV	300 Bq/cm ²
Class V	3000 Bq/cm ²

(1) (Adapted from CEGB Safety Rules)

(2) Pb-210, Po-210, Ra-223, Ra-226, Ac-227, Th-227, Th-228, Th-230, Pa-231, U-230, U-232, U-233, U-234, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, Cm-242, Cm-243, Cm-244, Cm-245, Cm-246, Cf-249, Cf-250, Cf-252.

(3) From Wrixon et al. 1979

(4) Radionuclides by class:

Class I AC-227, Th-228, Th-230, Th-232, Th-nat, Pa-231, U-232, U-233, U-234, U-236, alpha Emitters with Z>92

Class II Sm-147, Pb-210, Th-227, U-235, U-238, U-depl, U-nat, U-enr, Pu-241

Class III Other nuclides except those in Classes IV and V

Class IV C-14, S-35, Mn-54, Co-57, Zn-65, Ga-67, Se-75, Br-77, Sr-85, Tc-99m, Cd-109, I-123, I-125, Cs-129, Hg-197

Class V H-3, Cr-51, Fe-55, Ni-63, Cs-131

The Use of DLs

Because derived limits are 'working' limits and are related to the primary limits strictly only in circumstances that closely correspond to the model on which they have been calculated, it is important to distinguish them clearly from the primary dose equivalent limits (DELs).

A significant difference is in the implications of exceeding one of these figures. Exceeding a DEL represents a failure to achieve the standards set by international recommendations and implies an unacceptable risk to the individual. Exceeding a derived limit, on the other hand, will have a significance depending on the local circumstances. It may be entirely legitimate and, in any case, will only imply the possibility of exceeding a DEL. It may, however, represent the infringement of a management instruction and thus call for improved supervision or discipline. In general, derived limits should be thought of as tools helping the health physicist to achieve a consistent protection policy, but they must not be allowed to achieve the status of rigid rules. The professional health physicist must be allowed to use his judgement in deciding how to make the best use of these tools in the achievement of his ultimate aim, the achievement of both safe and economic operation.

Contamination Control

Control of radioactive contamination has the same objective as control of external radiation, that is, limitation of the dose received by the worker. The concern is the limitation of the dose equivalent to specific organs, or the limitation of the committed effective dose equivalent.

The results of poor contamination control are less predictable than the results of poor external radiation control. We can calculate accurately the dose that results from working in a known external radiation dose rate for a known time. Working in a contaminated area for a known time, however, does not permit calculation of the resultant organ doses without making a number of assumptions about breathing rate, transportability (solubility), metabolism, resuspension factors, transfer of radioactive material from surfaces to hands, from hands to mouth, through the skin, through cuts, etc. Only if the contamination control has been so poor that direct measurement of radioactive content of the body is possible is one able to calculate a reliable value for the organ dose.

The available means of contamination control are in order of importance:

1. Remove the hazard: remove or reduce the cause of loose radioactive material. This can usually only be done at the design or planning stage.
2. Contain the hazard: work that is likely to produce radioactive contamination should be done in enclosed and ventilated areas. Loose radioactive material should be regularly cleaned up. Remember however that it is the total dose to all workers that is used in the optimisation process and thus for example doses received during any cleaning operation need to be less than the doses that would result if the cleaning were not carried out.
3. Protect the worker: where radioactive contamination is expected and unavoidable and other means of protection are unsuitable, the worker should be protected with protective clothing and respiratory protection as appropriate. Above all the worker should be protected by being given adequate and appropriate training.

Surface Contamination DLs

Any operational health physics measurement must provide a basis for decisions on control measures or it must contribute to the interpretation of other data so as to provide a basis for such decisions. In some cases, for example in personnel monitoring for external radiation, the decision may not be an immediate one and may be based on the accumulation of information from repeated measurements. It follows that operational measurements, either singly or in combination, must have a quantitative relationship to action and as the basis of radiological protection is the concept of dose equivalent limits, it is obviously desirable that, wherever practicable, health physics measurements should be related to the radiation dose to man.

The measurement of surface contamination is a widespread operational health physics technique. It is used as a basis for decisions concerning decontamination, release of material from controlled areas and the level of precautions to be adopted within controlled areas. Clearly it is desirable in principle to set limits of surface contamination that are derived from the DELs recommended by the ICRP. Unfortunately, however, experience has shown that there is not necessarily a good correlation between surface contamination in a workplace and the exposure of workers. Nevertheless, the systematic absence of surface contamination above some defined level usually indicates a

high standard of primary containment or working techniques and, if the level is suitably chosen, provides strong evidence that there is no routine internal contamination of workers and thus no need for routine individual monitoring for internal contamination.

Since surface contamination is not necessarily the only cause of internal or external exposure of workers, the setting of an appropriate limit for surface contamination must depend to some extent on experience rather than on attempts to assess a direct relationship between surface contamination and the radiation dose to workers. The figures in general use have developed over a long period of time. The earliest surface contamination limits were based on what was considered achievable and at the same time measurable with the instruments then available. In the early 1950's attempts were made to establish a quantitative relationship between surface contamination levels and the resulting dose to workers. As it happened these calculations in general indicated that the existing surface contamination limits were acceptable. Similar calculations have been done from time to time since then, using new data as they became available. If the recalculated limit is higher than the limit in use, there is usually no change made, however if the recalculated limit is lower than the limit in use, the new lower limit should be used. Examples of the type of model and calculation used are given below. Table 1 shows the figures in their most general form for 'contaminated' areas and Table 2 shows some recommended 'clearance' values for natural uranium and thorium.

The Choice of a DL of Surface Contamination in Controlled and Supervised Areas

Although the choice of a DL is largely based on experience it is necessary to confirm that the value chosen will not result in unacceptable exposure of workers directly from involving the surface contamination. In controlled areas, the mechanisms by which workers may be exposed to radiation as a result of surface contamination are direct external radiation from the contaminated surfaces, irradiation of contaminated skin, ingestion of radioactivity from contaminated skin, direct intake through cuts, and inhalation of radioactivity resuspended from the contaminated surfaces. The normal standards of hygiene in active areas should eliminate any significant risk of ingestion. Experience has also shown that minor wounds, such as cuts and scratches, do not result in significant intakes of radioactive material unless the instrument causing the wound was a device such as a tool used directly in operations with radioactive material. Thus the mechanisms usually considered are irradiation of the skin, ingestion and inhalation.

Table 2

Comparison of levels of contamination for inactive areas or 'clearance' recommended by various authorities.

Bq/cm²

	CEGB 1968	1980 Code	UK ⁽²⁾ 1979	UK ⁽³⁾ 1982	Based on ICRP30 ⁽⁴⁾ 1983	1987 Code ⁽⁵⁾ Guideline
U-nat	3.7	3.7 ⁽¹⁾	0.3	1	1	1
Th-nat	3.7	3.7 ⁽¹⁾	0.3	0.1	0.1	0.5

(1) The 1980 Code does not specify a 'clearance' level. This is the limit for 'offices', the limit for skin is one tenth of this.

(2) Wrixon et al. 1979.

(3) U.K. Health and Safety Executive.

(4) Carter 1983a,b; Ching 1985.

(5) Assessment of Doses and Control of Exposure to Meet the Radiation Protection Standards. A draft guideline to the 1987 Code.

The simplest method of considering these problems is to use a simplified and standardised model relating the surface contamination to the possible dose to workers. The model is chosen so that it is unlikely to underestimate the dose and it is then used to calculate a derived limit for surface contamination. If this limit is more than the corresponding derived limit currently in use, then the current figure can be regarded as satisfactory. If the calculated figure is less than the current figure, the current figure has to be applied with reservations, particularly if the workers are also exposed to external radiation and airborne contamination. Examples of the calculations are given and some of the current DL values are examined below.

External Radiation from Surface Contamination

This problem applies only to contamination by beta emitters since the skin dose from X and gamma radiation and alpha emitters is small at the levels of surface contamination encountered. The draft Guidelines to the 1987 Code (DASETT 1987) recommend a DL of 10 Bq/cm^2 for uranium alpha contamination in controlled areas (i.e. contamination levels in controlled areas may routinely reach this level but should not be allowed to exceed it for more than short periods). Most uranium dusts will have equal alpha and beta activity. The corresponding radiation dose rate depends on the beta energy but over a wide range of energies will be 1 or 2 $\mu\text{Gy/h}$ for a beta contamination level of 1 Bq/cm^2 at a depth of 5 mg/cm^2 of tissue in contact with the contaminated surface. [Note that this is less than the 7 mg/cm^2 skin depth frequently used for such calculations and corresponds to the skin thickness of the face and arms (Whitton 1973)]. The dose rate from a surface contaminated at the DL will thus be no more than about $20 \mu\text{Gy/h}$. With continuous contact for a working year of 2000 hours, this would amount to 40 mGy in the year compared with a maximum permissible dose to skin of 500 mSv in a year. (ICRP 1978) (Beta radiation has a QF of 1. This is covered in another lecture). Since neither contact with contaminated surfaces, nor contamination of the skin will be continuous, the DL is satisfactory for controlling skin dose. (But it must also be checked in the inhalation and ingestion mechanisms of the model).

Inhalation of Resuspended Dust

The general model for assessing this situation has to be grossly over-simplified. In the first place, each nuclide has its own inhalation DAC and they should in principle be considered individually; many nuclides have more than one DAC depending on transportability (solubility in body fluids).

Secondly the relationship between surface contamination and the concentration in air depends on the surface, the form of the contaminant, the way the surface is disturbed and the amount and pattern of the ventilation of the workplace. A common simplification is to select one highly toxic alpha emitter and one highly toxic beta emitter as representative of the two classes of radioactive materials. ^{239}Pu and ^{90}Sr are commonly used examples. The second simplification is to relate the surface contamination to the resulting airborne contamination by means of a resuspension factor having units of length^{-1} , so that multiplying the surface contamination by the resuspension factor gives a direct indication of the air contamination. Resuspension factors have been determined by experimental programs and by measuring the surface and air contamination in practical situations. Not surprisingly, a wide range of values has been reported. Data were reported in papers to the Gatlinburg Symposium on Surface Contamination (AACC, 1967) and some additional largely confirmatory data were reported at the Bournemouth Symposium on the Radiological Protection of the Worker by the Design and Control of his Environment (SRP 1966). The majority of figures range from 10^{-6} m^{-1} with a few results up to 10^{-3} m^{-1} . All the results relate to the average surface contamination over substantial areas, several square metres at least, and there was some evidence that when the contamination was restricted to small areas resuspension factors were very much lower. In general, the higher figures were associated with surfaces such as concrete, from which dust is readily resuspended.

A representative figure, conservative in most cases but not necessarily adequate to cope with regular widespread contamination, particularly if on dusty surfaces, would be 10^{-5} m^{-1} (10^{-7} cm^{-1}). For widespread contamination by the most toxic alpha emitters of 0.3 Bq/cm^2 , the corresponding air contamination would be $3 \times 10^{-2}\text{ Bq/m}^3$. This is very close to the DAC for ^{239}Pu ($8 \times 10^{-2}\text{ Bq/m}^3$) or that for Th-232 ($2 \times 10^{-2}\text{ Bq/m}^3$). The general DL for surface contamination by the more toxic alpha-active materials given in part 2 of table 1 thus could be inadequate for widespread contamination by the more toxic alpha emitters on dusty surfaces. It remains satisfactory for the more usual situation where the contamination is sporadic and patchy and where surfaces have been chosen to avoid excessive dust raising.

For most beta contamination, the DL from table 1 is 30 Bq/cm^2 and the corresponding concentration in air is 3 Bq/m^3 . This is more than a factor of 10 lower than the DAC for ^{90}Sr (60 Bq.m^3) and, from the point of view of inhalation, the DL is probably on the conservative side.

For the special case of uranium mine and mill dusts, using a resuspension factor of $5 \times 10^{-6} \text{ m}^{-1}$ as recommended for this class of dust by Wrixon et al. (Wrixon et al. 1979) and the uranium dust alpha surface contamination DL of 10 Bq/m^3 the corresponding air contamination would be 0.5 Bq/m^3 . This is slightly greater than the recommended DAC in Table 10 of the guideline (DASETT 1987), and so for the model used, the DL is only just adequate.

The Choice of a DL for Surface Contamination in Inactive Areas

Surface contamination may occur in inactive areas either as the result of the spread of contamination from active areas or because of the transfer of items of equipment from active areas to inactive areas. One of the principal reasons for setting a DL for inactive areas is to allow quantitative control to be kept over contamination on items being transferred from active areas. It is conventional to take the DL for inactive areas as 1/10th the DL for active areas. Historically this reduction by a factor of 10 was based on the public DELs being a factor of 10 below the occupational DELs. Now that that factor has been increased to 50 there could be a case for a reduction of clearance levels. The presence of surface contamination at any level in inactive areas however is much less common than it is in active areas hence the model used above would have to be modified. This plus the factor of 10 should be enough to avoid any problems either from external radiation or inhalation caused by surface contamination. However, the standards of hygiene in inactive areas may be lower than those in active areas and this could mean that intake by ingestion now becomes significant. (Table 2 gives limits for inactive areas.)

Air Contamination DLs (DACs)

Air contamination DLs can be derived from the ICRP recommended DELs more directly than can the surface contamination DLs. Once again however the parameters of the chosen model may not be the same as the real life situation and so non-compliance with the DL will not necessarily result in non-compliance with the DEL.

The derived limits of air contamination presented by the ICRP and in the Code assume that occupational exposure is to a worker who corresponds exactly with 'Reference Man' in ICRP 23 (ICRP 1975): a constant level of air contamination for 40 hours a week, 50 weeks a year; that the worker in this time breathes 0.02 m^3 of air each minute; that the contaminant in the air has a particle size of $1 \text{ }\mu\text{m}$ AMAD and that there are no other sources of internal or external exposure. In the unlikely event of a person being exposed in exactly this manner they would have an intake of one Annual Limit on Intake (ALI) at the end of the year and would thus be committed to receive the primary dose equivalent limit. It must again be emphasised that the primary dose equivalent limit is the overriding concern and derived limits should be used with some circumspection.

When the intake of radioactive material is by inhalation or ingestion, but not by both routes together, and with no external radiation present, the primary dose equivalent limits may be deemed to be complied with if the intake of radioactive material is not in excess of an appropriate annual limit. 'Appropriate' here means:-

- (a) For a single radionuclide, the appropriate annual limit may be taken as the published Annual Limit of Intake (ALI) (ICRP 1979).
- (b) For a mixture of radionuclides of known composition, the appropriate annual limit may be taken to be that for which the summed relative fractions of intake with respect to published ALIs does not exceed unity. (See below)
- (c) For a mixture of radionuclides in which the concentration and toxicity of one predominates, the appropriate annual limit may be taken to be the published ALI for that radionuclide.
- (d) For a mixture of radionuclides where the exact composition is not known, but the radionuclides in it have been identified, the appropriate annual limit may be taken to be the lowest published ALI from among those corresponding to the radionuclides known to be present.
- (e) For a mixture of radionuclides whose composition is not known, but for which the presence of certain radionuclides can be positively excluded, the appropriate annual limit may be taken to be the lowest published ALI from among those radionuclides that may be present.

In general in uranium or thorium mining and milling the mixture composition is well enough known to use (b) above, and appropriate data are given in the draft Dose Assessment Guideline to the 1987 Code.

Calculation of Limits (DACs or ALIs) Where Two or More Radionuclides irradiate an Organ or Tissue

When two or more radionuclides irradiate a single organ or tissue, the derived limits for the individual radionuclides no longer apply, as for the model used, each radionuclide inhaled or ingested at the derived limit will lead to the exposed person receiving the dose equivalent limit to the relevant organ or tissue. The limits given in the Code must be modified by the rule for mixtures as follows:

If C_1, C_2, C_3 , etc. are the actual intakes or concentrations of the different radionuclides, and L_1, L_2, L_3 , etc. are the corresponding derived limits (ALIs or DACs), then the primary limits will not be exceeded if the sum

$C_1/L_1 + C_2/L_2 + C_3/L_3 + C_n/L_n$ etc. is less than (or equal to) unity

In other words, the limit for the mixture L_m is given by:

$$C_m/L_m = \sum C_i/L_i,$$

where C_m is the intake or concentration of the mixture.

$$1/L_m = \sum (C_i/C_m)/L_i$$

$$1/L_m = \sum F_i/L_i$$

where F is the activity fraction of each radionuclide present.

Special Features of Uranium and Thorium

Tables 3 and 4 give the uranium and thorium decay chains. Because these are families of radioactive materials with constant production of progeny we rarely deal with a single pure radionuclide. The data on derived limits published for example by the ICRP are calculated for a single radionuclide and for a particular exposure model. Thus the published values should not be used unthinkingly when dealing with radionuclides in one of the natural decay series.

Table 3 U-238 Chain

Main Energies (in MeV) and intensities (in %)

Nuclide	Half Life	α	β	γ
U-238	4.5×10^9 y	4.15(25) 4.20(75)		
Th-234	24.1d		0.10(20) 0.19(79)	0.09(4)
Pa-234m	1.14m		2.29(98)	
U-234	2.67×10^5 y	4.72(28) 4.77(72)		
Th-230	8×10^4 y	4.62(24) 4.68(76)		
Ra-226	1620y	4.60(6) 4.78(95)		0.186(4)
Rn-222	3.82d	5.49(100)		
Po-218	3.05m	6.00(100)		
Pb-214	26.8m		0.65(50)0.71(40) 0.98(6)	0.0295(19) 0.352(36)
Bi-214	19.7m		1.0(23)1.51(40) 3.26(19)	0.609(47) 1.12(17) 1.76(17)
Po-214	164 μ s	7.69(100)		
Pb-210			0.016(85) 0.061(15)	0.047(4)
Bi-210			1.16(100)	
Po-210		5.30(100)		
(Pb-206)	(Stable)			
<hr/>				
Total activity per Bq of U-238		8	5.76	1.5
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Total activity per Bq of U-238 with radon + daughters missing		5	4	0.1
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Note: Low energy beta from Pb-210 may not be detected by some instruments.

Table 4 Th-232 Chain

Main Energies (in MeV) and intensities (in%)

Nuclide	Half Life	α	β	γ
Th-232	1.39×10^{10} y	4.01(76)		
Ra-228	6.7y		0.055(100)	
Ac-228	6.13h		2.1(12)	0.34(15)
			1.7(12)	0.91(25)
			1.2(35)	0.96(20)
Th-228	1.90y	5.42(71)		
		5.34(28)		
Ra-224	3.64d	5.68(94)		0.24(4)
		5.43(6)		
Rn-220	55s	6.29(100)		
Po-216	0.15s	6.78(100)		
Pb-212	10.6h		0.346(81)	0.239(47)
			0.586(14)	0.30(3)
Bi-212	60.6m	6.05(25)	1.55(5)	0.727(7)
		6.09(10)	2.26(55)	
PO-212	304ns	8.78(64)		
Tl-208	3.10m	1.80(18)		1.28(9), 1.52(8)
				0.86(4), 2.61(36)
(Pb-208)	(Stable)			
Total activity				
per Bq of				
Th-232	6	3.5	2	
Total activity				
per Bq of				
Th-232 with	3	1.5	0.6	
Thoron +				
daughters missing				

Note: Low energy beta from Ra-228 may not be detected by some instruments.

It is usually necessary to calculate a derived limit for the mixture using the 'method of mixtures'.

As an example, for thorium ore, assuming thoron and its daughters to be absent, and using ICRP30 values then the mixture derived limit is given by:

$$\frac{1}{\text{DAC}_m} = \frac{1}{4.6} \left[\frac{1}{2 \times 10^{-2}} + \frac{1}{2 \times 10^1} + \frac{0.6}{6 \times 10^2} + \frac{1}{2 \times 10^{-1}} + \frac{1}{3 \times 10} \right]$$

where 4.6 is the total activity in the chain Th-232 to Ra-224 for unit activity of Th-232 and DAC_m is the DAC for the mixture.

$$\text{DAC}_m = 8.4 \times 10^{-2} \text{ Bq/m}^3 \text{ which should be rounded to } 0.1 \text{ Bq/m}^3.$$

Note that this is total (alpha + beta) becquerels, note also that inclusion of thoron and its daughters would result in a DAC_m of about $1.7 \times 10^{-1} \text{ Bq/m}^3$.

Since some uranium is normally present in mineral sand, this DAC_m would increase to 0.21 Bq/m^3 for typical percentages of uranium. To avoid an unwarranted suggestion of precision, this should be rounded to 0.2 Bq/m^3 .

As this DAC is based on continuous exposure for 40 hours a week, 50 weeks a year, it is just as correct, and frequently more useful to quote it as a weekly exposure limit of 8 Bq.h/m^3 or for a standard breathing rate a weekly intake of 9.6 Bq.

The derived limit for the mixture this can be presented in a number of ways. The example above is used to illustrate these:

- (a) in terms of total activity (0.2 Bq/m^3 or 8 Bq.h/m^3);
- (b) in terms of alpha activity (0.1 Bq/m^3 or 4 Bq.h/m^3);
- (c) in terms of beta activity ($8 \times 10^{-2} \text{ Bq/m}^3$ or 3 Bq.h/m^3); but note that in this case that the low energy beta from Ra-228 may not be detected by many instruments;
- (d) in terms of the activity of a single radionuclide present in the mixture ($2 \times 10^{-2} \text{ Bq/m}^3$ or 0.7 Bq.h/m^3 of Th-232);

Table 5 U-238 Chain ALIs and DACs

Nuclide	Stochastic ALIs (Bq)			DACs (Bq/m ³)			
	Inhalation			Ingestion			
	D	W	Y	D	W	Y	
U-238	8×10^4	$3 \cdot 10^4$	$2 \cdot 10^3$	8×10^5	2×10^1	1×10^1	7×10^1
Th-234	-	7×10^6	6×10^6	1×10^7	-	3×10^3	2×10^3
Pa-234m						1×10^5	1×10^5
U-234	7×10^4	3×10^4	1×10^3	7×10^5	2×10^1	1×10^1	6×10^{-1}
Th-230	-	6×10^2	7×10^2	4×10^5	-	1×10^1	2×10^{-1}
Ra-226	-	2×10^4	-	2×10^5	-	1×10^1	-
Rn-222					1.5×10^5		
Po-218							
Pb-214	3×10^7	-	-	3×10^8	1×10^4	-	-
Bi-214	3×10^7	3×10^7	-	8×10^8	1×10^4	1×10^4	-
Po-214							
Pb-210	1×10^4	-	-	4×10^4	4	-	-
Bi-210	1×10^7	1×10^6	-	3×10^7	4×10^3	4×10^2	-
Po-210	2×10^4	2×10^4	-	1×10^5	1×10^1	1×10^1	-
(Pb-206)							

Table 6 Th-232 Chain ALIs and DACs

Nuclide	Stochastic ALIs (Bq)			Ingestion	DACs (Bq/m ³)		
	D	W	Y		D	W	Y
Th-232	-	1x10 ²	2x10 ²	7x10 ⁴	-	2x10 ⁻²	4x10 ⁻²
Ra-228						2x10 ¹	
Ac-228	6x10 ⁵	2x10 ⁶	2x10 ⁶	9x10 ⁷	1x10 ²	6x10 ²	7x10 ²
Th-228	-	8x10 ²	6x10 ²	5x10 ⁵	-	2x10 ⁻¹	3x10 ⁻¹
Ra-224	-	6x10 ⁴	-	6x10 ⁵	-	3x10 ¹	
Rn-220						2.5x10 ⁵	
Po-216							
Pb-212	1x10 ⁶	-	-	5x10 ⁶	5x10 ²	-	-
Bi-212	9x10 ⁶	1x10 ⁷	-	2x10 ⁸	4x10 ³	4x10 ³	-
Po-212							
Tl-208							
(Pb-208)							

Table 7

Comparison of levels of airborne contamination
recommended by various authorities

Bq/m³ (alpha activity)

	1980 Code	ICRP	IAEA	ICRP 30	1987 CODE Guideline ⁽⁷⁾
U-ore	-	1.3 (4) 0.7 (5)	-	0.6 (2)	1.0
U.nat (product)	3.7 (1)		0.7 (1)	0.6 (2)	1.7
Th-ore	-	0.16 (5)	-	0.1 (6)	3.3
Th-nat	2.2 (3)	0.016 (5)(3)	0.02 (3)	0.02 (3)	-

- (1) Value for U-238
- (2) Carter 1983b, Woods 1985
- (3) Value for Th-232
- (4) ICRP 24
- (5) ICRP 47
- (6) This lecture
- (7) Draft Guideline on Dose Assessment

- (e) in terms of the mass of a single radionuclide present in the mixture ($4.5 \mu\text{g}/\text{m}^3$ of Th-232).

In general the most useful form of the derived limit is the one that relates to the method of measurement used, for example, where alpha detection is used the DAC should be expressed in terms of alpha activity rather than total activity.

DACs for Uranium and Thorium Dusts

DACs for uranium and thorium dusts that have been recommended by various authorities for occupational exposures are given in Table 7. As with surface contamination it has been conventional to reduce occupational DACs by a factor of 10 for public use, plus in this case a further reduction by a factor of 3 to take account of the fact that in any week a member of the local population breathes about 3 times more air than a designated worker breathes at work.

The change in public DEL from 5 mSv per year to 1 mSv per year should be taken into account when deriving DACs for members of the public.

The IAEA has recommended that values of ALIs for members of the public should be one hundredth of the relevant occupational ALI (IAEA 1982). If this recommendation is accepted then public DACs should be one three hundredth of the occupational DACs.

Calculation of DACs for Aerosols with AMADs other than $1 \mu\text{m}$

As stated earlier the ICRP model used to calculate DACs assumes an aerosol with an AMAD of $1 \mu\text{m}$. Where it can be shown that the actual AMAD is not $1 \mu\text{m}$, corrections to the value of the DAC can be made.

The effect of particle size on the % of dust deposited in the three regions of the lung, (nasal, tracheo-bronchial and pulmonary), is given in Figure 5.1 of ICRP 30 Part 1. The method of making particle size corrections is also given in ICRP30 and has been amplified by Woods (Woods 1985) and in a draft guideline to the Code.

PART 2 MEASUREMENT OF CONTAMINATION

Sampling RegimesOre dust

Provided that ore dust concentrations in air are confirmed to be low (substantially less than DAC values) monitoring may be infrequent, either monthly or quarterly, depending on local conditions, the samples may be analysed for gross alpha activity and, on the assumption of equilibrium among the uranium decay products, the alpha concentration may be compared with the mixture DAC for the chain. In the event of high ore dust concentrations, frequent monitoring and regular analyses of separate radionuclides may be advisable.

Airborne ore dust concentrations in mills may be high, depending on the quality of dust control equipment. In particular, exposures in sample preparation and assay rooms may be sufficiently high to warrant assessing individual exposures. This may be done by personal samplers or by calculations based on area monitoring.

Surface contamination measurements will not usually be required in mines, but may be necessary in sample preparation and assays rooms and should be carried out in meal rooms. Because of the low specific activity of ore, normal 'good housekeeping' is usually sufficient to control surface contamination.

Concentrate dust

Concentrate dust occurs at the product-end of mills and airborne concentrations may be high enough to require monitoring of individual exposures. Concentrates usually contain 80 to 90% or more of the product, uranium or thorium, and so exposures are essentially to the one radionuclide. Exposures again may be measured by personal samplers or by calculation from area monitoring.

The higher specific activity of product dust means that routine surface contamination surveys are desirable. If plant and equipment are painted in a colour that contrasts with the product dust then contamination levels above the DL can be detected visually.

Table 8

Levels of total surface contamination above which a controlled or supervised area should be identified.

(as recommended by the UK Health and Safety executive in 1982 and based on ICRP30).

Radionuclide	Level of contamination (Bq/cm ²) for:	
	Controlled areas	Supervised areas
²³¹ Pa	10 ⁻²	3 x 10 ⁻³
²²⁷ Ac, ²²⁸ Th, ²³⁰ Th, ²³² Th, Th-nat ²³² U, ²³³ U, ²³⁴ U, ²³⁶ U, all alpha emitters with Z > 92	10 ⁻¹	3 x 10 ⁻²
¹⁴⁷ Sm, ²¹⁰ Pb, ²²⁷ Th, ²³⁵ U, ²³⁸ U, U-depl, U-nat, U-enr, ²⁴¹ Pu	1	3 x 10 ⁻¹
Other radionuclides except those given below	10	3
¹⁴ C, ³⁵ S, ⁵⁴ Mn, ⁵⁷ Co, ⁶⁵ Zn, ⁶⁷ Ga, ⁷⁵ Se, ⁷⁷ Br, ⁸⁵ Sr, ^{99m} Tc, ¹⁰⁹ Cd, ¹²³ I, ¹²⁵ I, ¹²⁹ Cs, ¹⁹⁷ Hg	100	30
³ H, ⁵¹ Cr, ⁵⁵ Fe, ⁶³ Ni, ¹³¹ Cs	1000	300

Footnote

For the purposes of this table the level of contamination may be determined by averaging over an area not exceeding:

- . in the case of floors, walls and ceilings, 1000 square centimetres;
- . in any other case, 300 square centimetres.

Methods of Measurement

To measure either air or surface contamination it is necessary first to collect a sample of that contamination. The sample must be related as closely as possible to the circumstances which would result in human exposure; this is not easy.

In the case of air contamination, the material that should be sampled is not available, since that is the material actually inhaled. One approach is to take a sample from as close to a worker's face as possible, using a 'personal air sampler' (PAS). While this collects the sample from air which should be very similar to that breathed, the quantity of sample collected is usually small which may limit sensitivity. In addition there is often worker resistance to wearing additional paraphernalia.

The use of 'area' samplers is more widespread and their use eliminates the problems of sensitivity and worker resistance encountered with the PAS. It is important however that the area air sampler is positioned so that the air sampled is likely to be representative of the air breathed or, failing this, so that the air sampled is likely to be more contaminated than that breathed. Evidence has been presented (Langmead 1969, Butterworth and Donoghue 1970, Carter and Lumsden 1979, Pringle 1971, Lay and Mahathy 1985) that PAS results may be significantly higher than simultaneous area results. This must be borne in mind when interpreting area results, and if no other information is available it is suggested that the contamination of the air breathed is assumed to be an order of magnitude greater than that collected by an area air sampler.

In the case of surface contamination, it is primarily the 'loose' contamination that should be measured as this can be transferred to hands or clothes. If direct measurement (with a probe, see below) is used this will respond to 'fixed' contamination as well as to 'loose' contamination. If indirect measurement is used (by smear etc., see below) then there is the possibility that most of the loose contamination present is removed during the measurement process, thus overestimating the level of contamination remaining.

Detection and measurement of radiations are covered in another lecture.

Surface Contamination

There are two general methods of measuring surface contamination, directly by probe or indirectly by smear (also called swab or swipe).

In the direct method a suitable detector probe (usually a scintillation or geiger counter) is passed over the surface to be measured. The sensitive area of the probe should be as large as practical in order to give good sensitivity and in order to provide an averaging effect where there may be discrete spots of contamination. It must be close enough to detect the radiation emitted but not so close as to pick up contamination on itself. 'Close enough' for alpha radiation is about 5 mm and for beta radiation is about 20 mm (see figure 1). It is important to note from figure 1 that small changes in source-detector distance have a large effect on the alpha response of the detector, but only a small effect on the beta response. Thus for effective probe surveys with an alpha detector a steady hand and a smooth surface are required.

The probe must be moved slowly enough to allow it to respond to isolated spots of contamination; this will be a function of the time constant of the electronic circuit.

Where alpha measurement is being used to assess surface contamination, consideration has to be given to self absorption. In the case of dense alpha emitting dusts (such as uranium chain dusts) valid measurements by probe may not be possible except at the lowest levels. Contamination levels above about 50 Bq/cm will not be reliably measured by the commonly used DP2R probe, and even below this level the measurement may not be easy (see figure 2).

Where beta measurement is being used, the probe used must be capable of detecting the beta energies of interest (see footnotes to tables 3 and 5). Since beta detectors will respond to general background beta and gamma radiation as well as to beta radiation from the surface being surveyed for contamination, some effort must be made to subtract this background. This can usually be achieved by making a series of measurements with the probe well away from contaminated surfaces to assess the background signal and subtracting this value from measurements of the surface contamination. Where background levels are significant, a lead shield around the detector may be useful to ensure that the detector only responds to radiation from one direction. Alternatively in areas where interference by background radiation is a problem, indirect means of measuring contamination should be used.

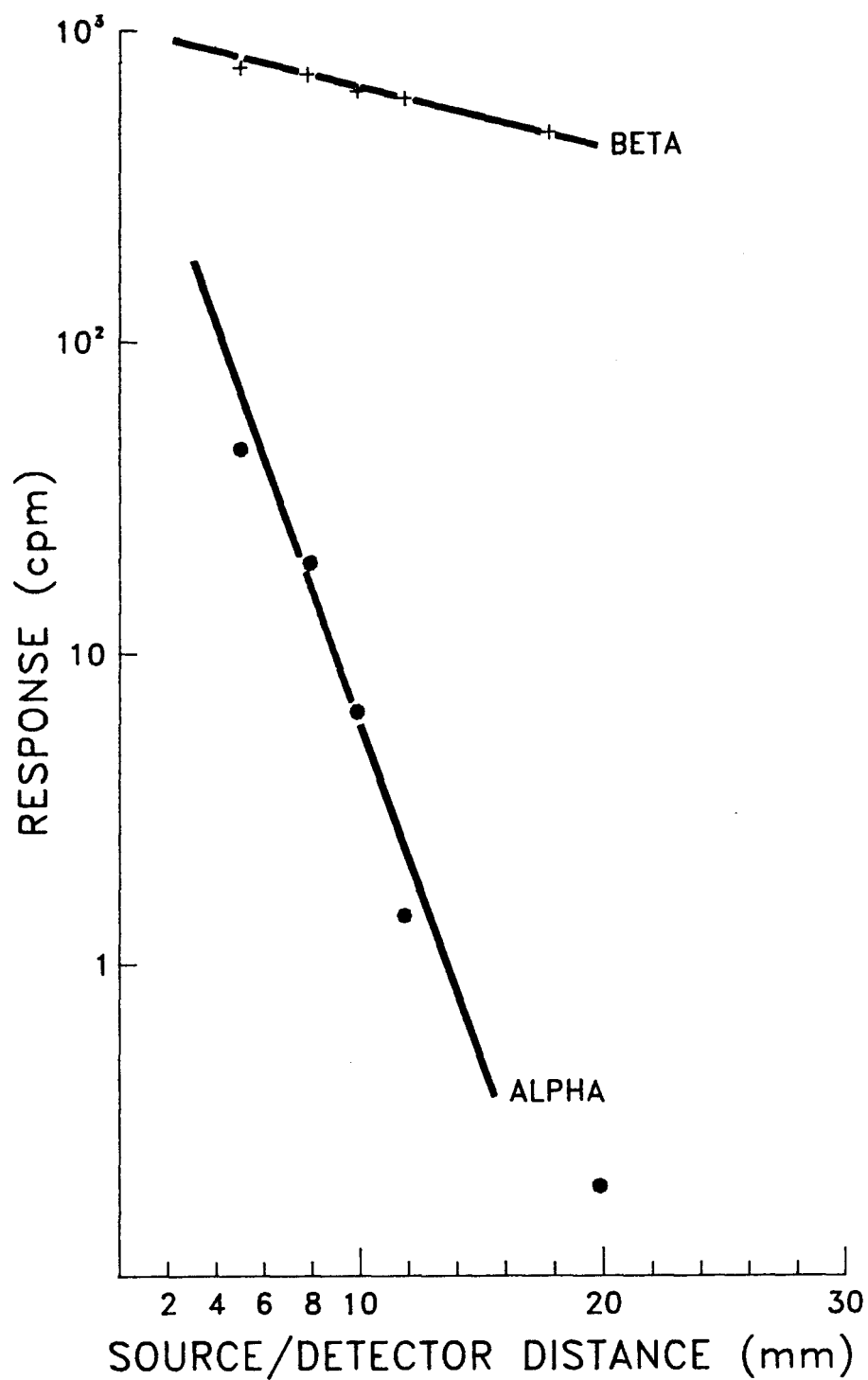


Figure 1. Variation of detector response with source-detector distance for alpha and beta particles.

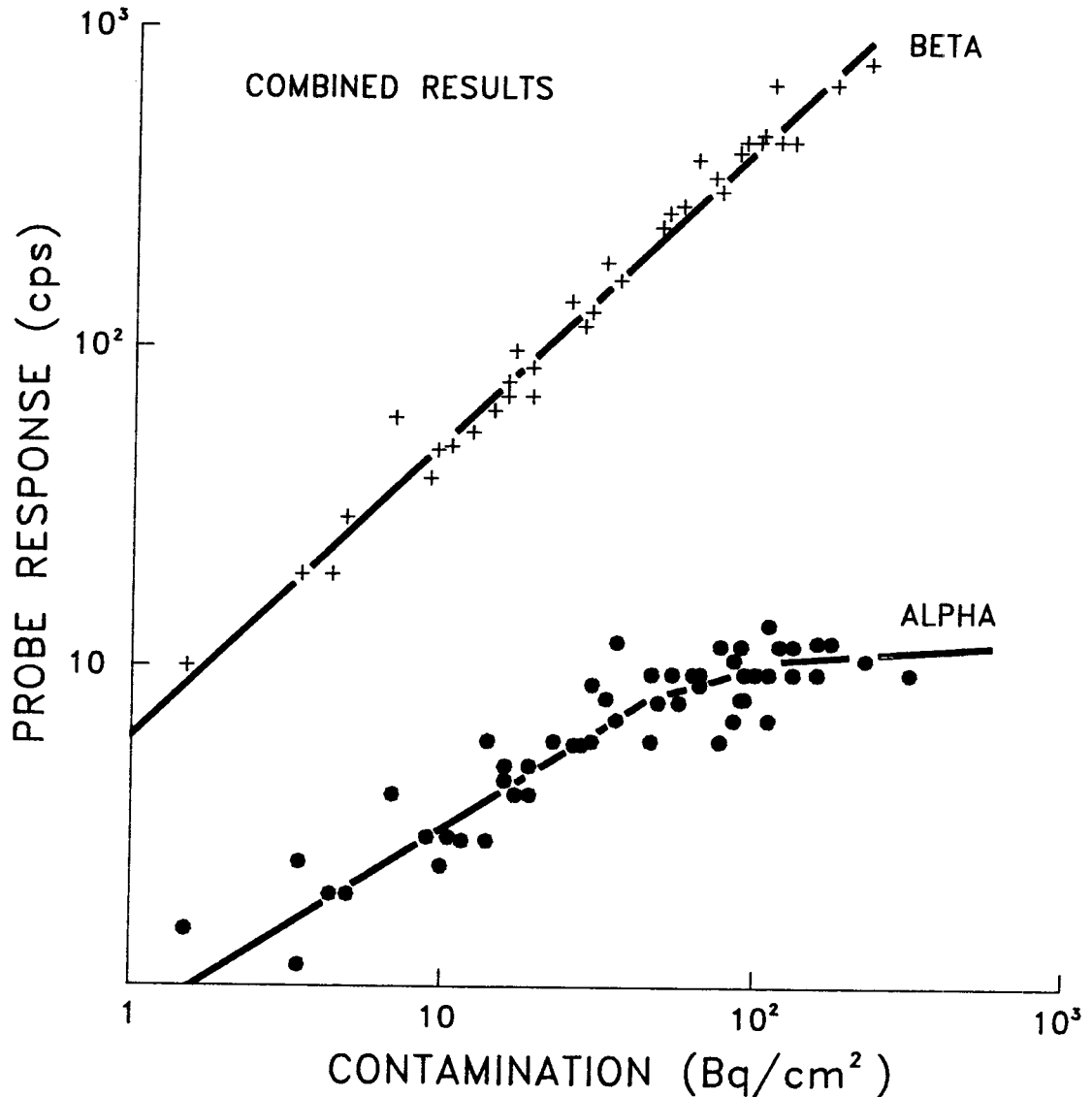


Figure 2. Variation of detector response with contamination level for "dense" alpha emitting dusts, showing the effect of self absorption.

In the case of indirect measurement of surface contamination, a filter paper is rubbed on a known area of the surface, and the radioactivity picked up on the filter paper is measured. The area over which the filter paper is rubbed is usually 300 cm but may be 1000 cm. It is usually assumed that 10% of the 'loose' contamination is transferred to the filter paper (e.g. Wrixon et al. 1979). Evidence has however been presented (Auty and Evans 1986) that using wet filter paper up to 60% may be transferred. If the contaminant is an alpha emitter and the filter paper is measured in a windowless alpha counter then self absorption is much less important than when using a probe and contamination levels up to at least 500 Bq/cm can be reliably measured (see figure 3).

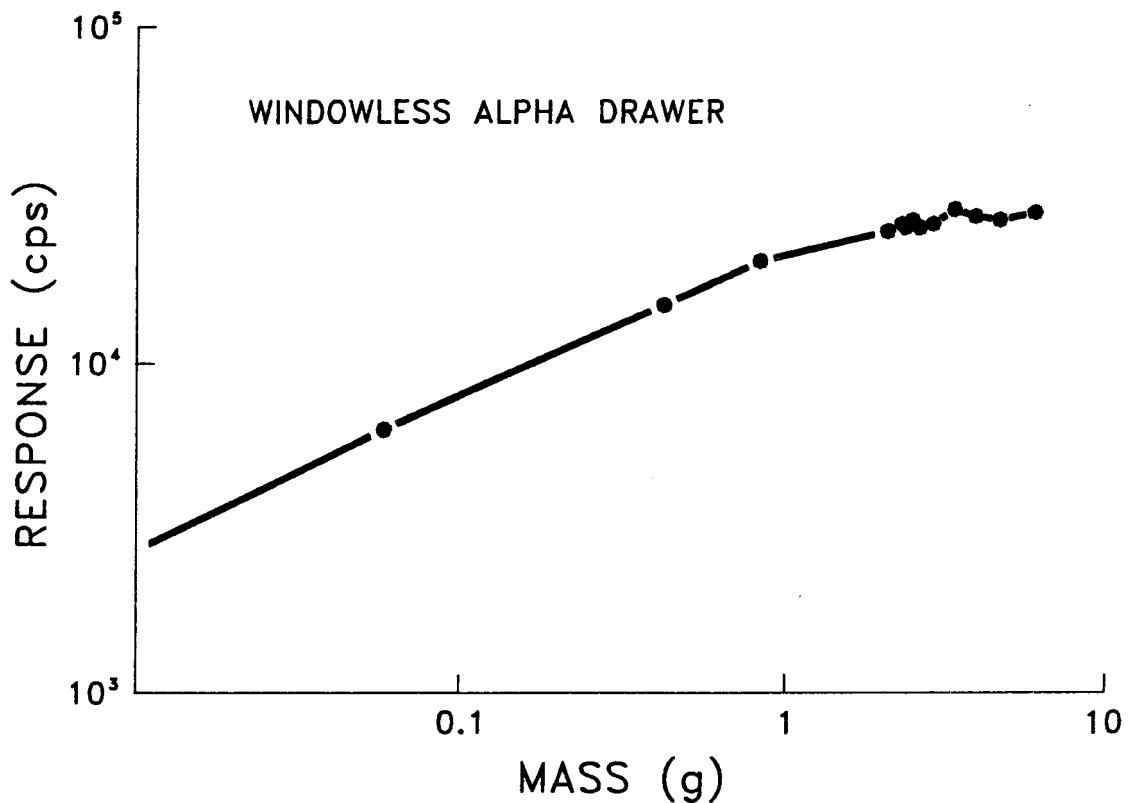


Figure 3. Variation of response of a windowless alpha counter with source mass.

Air contamination

There are a number of ways of collecting air samples, including drawing air through filters, grab sampling, electrostatic precipitation, adsorption and impingement. Most methods are 'active', that is they require some energy supply (electricity, compressed air) but a few are passive.

The special field of radon and thoron daughter sampling is covered in another lecture.

Before deciding on an air sampling program the desired sensitivity should be decided, as this will influence the choice of equipment and the details of the program. This is particularly so in the case of environmental surveys to assess exposure of members of the public. For example if the objective is to be able to measure down to a tenth of the public DAC, and the radionuclide is Thorium (occupational DAC 2×10^{-2} Bq/m see Table 6) then the target sensitivity could be to be able to detect 2×10^{-5} Bq/m - not an easy objective! In order to achieve adequate sensitivity in occupational monitoring it is usually necessary to collect samples from at least half a

cubic meter of air. In work place monitoring, personal air samplers are preferred to area air samplers if the air contamination levels are greater than $0.1 \times$ the DAC. Where personal air samplers are used, in order to collect a large enough sample, it is usually more important to use the highest flow rate that is available (and acceptable to the wearer) than to use a particle size selector. Where area samplers are used they should be positioned with regard for air flow and occupation patterns.

Mechanical sampling

Concentrations of ore dust or concentrate dust are measured by collecting air samples on high-efficiency filters and analysing the samples in the laboratory by alpha or beta counting, alpha or gamma spectrometry, chemical methods or chemical separation followed by counting individual radionuclides. Air may be drawn through the filters by means of electrically powered (battery or line power) pumps or compressed air ejectors. The sensitivity is mainly dependent on sample volume and can be adjusted to almost any desired value by choice of sampling time, and hence sample volume.

Air sampling equipment consists of a filter holder, an air pump and a flow meter. Filter holders in sizes of from about 2 to 10 cm diameter are commonly used. Battery-powered pumps of convenient size are available with capacities up to about 15L/min; air ejector and line-powered electric pumps have capacities up to 500L/min or more. Air samplers may be purchased from commercial sources as integral units comprising filter holder, pump and flow meter grouped in a convenient package.

High-efficiency membrane or glass fibre filters are suitable for dust sampling. However, if sensitive analyses are to be made for specific nuclides, the type of filter should be checked for the presence of the nuclide as a background constituent. Similarly if chemical processing and analysis of the sample is required the filter material will need to be chosen with this in mind. Envelopes or small containers are desirable for filter and forceps storage for handling. Since the sample will be on the front face of the filter some means of ensuring that this face is presented to the detector is required. This may be a marking provided by the filter manufacturer or a pen or pencil mark may be used, or the filter may be used in, and retained in a marked and numbered holder until after it has been counted.

Air is drawn through a high-efficiency filter at a combination of flow rate and time chosen to give the desired sensitivity. For measurements of ore dust, sample volumes of up to several cubic metres of air may be required, especially in uranium mines where the concentrations tend to be low.

In mines, ore dust is usually measured by means of general-air (area) samples. The samplers should be positioned away from walls and 1 m or more above the floor on a tripod or other means of support. Sampling location, time, duration and flow rate should be noted in each case.

In mills, area samples are collected in a similar fashion. Breathing-zone (BZ) samples are collected by positioning a personal air sampler within the breathing zone of the mill worker during the performance of specific, dusty tasks. A brief description of the operation being monitored as well as the aforementioned sampling data should be noted.

An example of an sampling record form is given in figure 4.

Note that radon or thoron daughters will be collected on the filter and assessment of the long-lived radioactivity on the filter paper will in general need to take account of this. One technique, used on figure 4, is to count the filter twice, with a 40 min decay period in between. During this 40 minute period any radon daughters present will have decayed to approximately half their initial value. This technique will not, of course, correct for the presence of thoron daughters; for these a decay period of 11 hours between counts is required. Where it is necessary to detect low levels of long-lived radionuclides and where radon or thoron daughters may be significantly present, a practical strategy is to re-count all filter papers 24 hours after the initial count.

Form: SS(HP)2g
(Rev. Sept. 1972)

HEALTH PHYSICS AIRBORNE CONTAMINATION SURVEY REPORT

C	R	Be

To: Area Supervisor: _____ Area (e.g. Lab.No.) _____ Area Classification: _____
 Building _____
 You are advised of the following results of a Health Physics Airborne Contamination Survey of your area which was carried out on _____/_____/_____(date). Your attention is drawn to the recommendations made in the last column.
 Note: See over for the radioactive contaminant groups.

1. SAMPLING Survey Ref. No: _____

Sampler Type	Sampler Position Breathing zone Other*	Time		Air Flow		Total Air Flow	Location and Job Details	Type of Breathing Protection
		On	Off	Start	End			

2. ASSESSMENT

Assessment Mode	Count Time Min	Total Count	Total cpm	Back-ground cpm	Total -Back-ground cpm*	2N ₂ - N ₁ cpm	Cntr. Eff.	dpm/m ³ ng/m ³ µg/m ³	3* Occ. 40 mpc Units
α, initial									
β, initial									
α, 40 min. decay									
α, _____ decay									
β, _____ decay									
Other: (initial)									

* Delete whichever does not apply.

3. SUMMARY OF RESULTS

Radioactivity	Concentration	Contaminant or Group	Recommendations or Remarks
Initial Count _____ After _____ h, _____ mpc	Initial Count _____ After _____ h, _____ mpc		
Other _____ mpc			

Distribution: 1) Head, Safety Section _____ (Health Physics Surveyor) _____ (Date)
 2) Area Supervisor _____
 3) Area Health Physics Office File _____ (Area Health Physicist) _____ (Date)

Figure 4. An example of a sampling record form.

Sampling period

In contrast to conventional industrial hygiene, where the limits are normally concentrations of toxic substances in air, the radiological protection limits are intakes of radionuclides. Hence the ideal air sample takes air from the workers' breathing zone at the same rate and for the same period that he breathes, and retains the radioactive contamination from that air with the same retention efficiency as his respiratory system. In radiological protection the instantaneous concentration of radionuclide is irrelevant, it is the total intake that needs to be assessed. In practice of course the ideal air sample cannot be achieved, but every effort must be made to ensure that the sample is taken over the full period when significant air contamination may be expected. If expediency dictates a shorter sampling period, care must be taken to ensure that the sample is representative of the working period. If the dust generating process is variable it is almost impossible to interpret the results of short term air samples in terms of compliance with intake limits. (A Code guideline on assessment of compliance is planned.)

Environmental samples will usually need to be taken for the longest time that is consistent with filter loading in order to obtain adequate sensitivity.

Particle size selection

Since DACs recommended by the ICRP are based on an aerosol with a $1 \mu\text{m}$ AMAD, it may be worthwhile to calculate a site specified DAC if it can be shown that the actual aerosol AMAD is significantly different from $1 \mu\text{m}$.

Several authors have presented recalculated DACs for uranium dusts (Thind 1987, Woods 1985) particularly since the reduction in uranium DAC recommended in ICRP30.

Thus measurement of aerosol particle size has become a more common health physics procedure in recent years.

A number of particle size selection devices exist which enable the sample collected to be related to 'respirable' dust (Woods 1986). They range from simple devices which reject all dust above a certain size to more complex devices that provide samples in up to eight or ten size ranges. The simple ones should be used with caution, as they reject part of the total 'signal' and one must be sure that the rejected information is indeed irrelevant. The complex devices are useful for special investigations to assess the particle

sizes of the aerosols of interest. It should not usually be necessary to use them on a routine basis.

Impingers and adsorbers

In cases where the contaminant to be measured is a gas or a vapour, ordinary filter papers will not collect it. For some contaminants loading the filter with a suitable chemical may allow standard mechanical air samplers to be used. Examples are filter papers loaded with charcoal or silver nitrate. The chemical may also be used apart from the filter, for example charcoal used to trap radon.

Passing the gas stream through a liquid (which may be water) is also used as a means of trapping samples of gas or vapour contaminants. In this case the bubble size should be as small as practical (use a diffuser) and the bubble path as long as practical (a deep rather than shallow container).

Non-mechanical or Passive Samplers

Where many air samples are required over long periods, conventional mechanical air sampling can prove to be too expensive. A number of non-mechanical sampling techniques have been developed.

The sticky plate deposition collector is, as its name suggests a horizontal or vertical plate covered with a sticky material that traps any 'fall out' that falls into it. The 'tacky cloth' sampler developed in the U.K. consists of a cloth stretched over a wire frame. In its original form the cloth is resin impregnated and the wire frame is a truncated cone (lampshade). An alternative version uses dry cloth and a frame which is rectangular in elevation and vee-shaped in plan (Fry 1982, Thomas 1981). A plane rectangular frame has also been used for indoor measurements (Goldfinch 1973).

Stack-sampling

A specialised form of air sampling is stack sampling. Dependant on the contaminant to be detected, filters, absorbers or impingers may be used to collect the sample. Care is needed to ensure that the sample is representative of what is going up the stack. The location, design of, and gas flow into the sample probe influence whether the sample is representative or not. Where the dust producing process is variable there is considerable advantage in continuous stack sampling. If continuous sampling is not used then it is necessary to carry out a comprehensive 'commissioning' sampling campaign to ensure that the timing and duration of 'grab' samples give results that are representative of the stack discharge.

Sample Counting

Standard counting equipment for total alpha or total beta determination consisting of an appropriate detector coupled to a scaler, is generally suitable, provided attention is paid to the mass and particle size of the sample (alpha self absorption) and the window thickness of the detector. (A windowless detector is therefore recommended for alpha counting.) Efficiencies greater than 10% (beta) and 20% (alpha) are readily achievable.

Since all uranium mine and mill dusts are both alpha and beta emitters either type of counting may be used. If available, alpha or gamma spectrometry could also be used. Because of the low cost, simplicity and the low background achievable, total alpha counting is the most commonly used technique. Where total alpha counting is used it should be supplemented from time to time by chemical or isotopic analysis to ensure that any assumptions on the mixture proportions made in setting limits for total alpha activity remain valid.

Sampling and counting time should be chosen in combination to achieve a lower limit of detection (LLD) that is one tenth or less of the appropriate derived limit. For example for a derived limit of 1 Bq/m^3 and sample volume of 1 m^3 it is necessary to be able to detect 0.1 Bq of sample. Using the definition, $\text{LLD} = 4.7 \text{ Sb}$ where Sb is the standard deviation of the background (USEPA 1980) and assuming an alpha counter with 20% efficiency and a background of 1 count per minute.

0.1 Bq sample would give 0.02 cps = 1.2 cpm

To achieve this as a LLD it would be necessary to count sample and background each for about 15 min.

(15 min background count = 15, $\text{Sb} = \sqrt{15} = 3.87$,

$\text{LLD} = 4.7 \times 3.87 = 18.2$ counts above background in 15 minutes, i.e. 1.2 cpm)

Note: this assumes that the background standard deviation is due only to counting statistics.

Errors due to self absorption in the filter paper

At least initially, filters should be checked for self absorption . If self absorption is found to be significant, each batch of filters should be tested, and the necessary correction factors applied to the alpha counts obtained.

Filter self absorption can be determined by first counting alpha activity on the filter front face (C_1), then counting the reverse face (C_2), and finally covering the front face with an unused filter of the same type and recounting the front side (C_3). Percentage self absorption is given by $100(C_2 - C_3)/(2C_1 + C_2 + C_3)$.

Interpretation of ResultsCalculation of individual exposures to airborne dusts

If personal samplers are used, and if they are worn for the full exposure period, calculation of intake is very simple. The activity on the filter paper compared with the intake of activity will be in the ratio of the total air sampled to the total air breathed (or for simplicity to the rates of sampling and breathing). Many personal air samplers have flow rates of 2L/min and reference man is assumed to breathe at a rate of 20L/min, thus the intake in these conditions would be ten times the activity measured on the filter paper.

Personal dust samplers allow the calculation of individual exposures directly. For determining individual exposures from general area surveys the following data are needed:-

- (a) Concentrations of airborne radionuclides (ore dust or concentrate dust, as applicable) in all occupied areas and
- (b) Time spent by each worker in his respective working locations.

For a given worker over a reference period (week, month or quarter) the exposure is

$$D = \sum Wt$$

where D is the radionuclide exposure (in $\text{Bq}\cdot\text{h}/\text{m}^3$), W is the radionuclide concentration in a particular area or at a particular operation, and T is the exposure period corresponding to W .

Where the airborne dust consists of a single known radionuclide, it is possible to calculate the intake in Bq from the exposure in $\text{Bq}\cdot\text{h}/\text{m}^3$ using the ICRP reference man (ICRP 1975) breathing rate of 9.6 m^3 in a working day. Where the composition is unknown, or is a mixture, then it is often simpler to calculate the exposure in DAC-hours using a generalised or mixture DAC. In this case the calculated exposure is compared with the weekly limit of 40 DAC-hours.

Calculation of doses and dose equivalent from exposure

Where the intake in Bq is very much lower than the pro rata ALI for the period being assessed it is usually sufficient to calculate the effective dose equivalent on the basis that one ALI is equivalent to 50 mSv. (Or that 2,000 DAC hours is equivalent to 50 mSv.) Under these circumstances the individual organ doses do not need to be calculated.

If however the intake is significant in relation to the pro rata ALI (more than say 50%) then a more carefully considered calculation may be necessary. The ALI is the result of calculations for reference man and an assumed aerosol. In an actual case of significant exposure it would be advisable to reassess the organ dose and effective dose equivalent using the parameters appropriate to the case. The actual particle size distribution (AMAD and GSD) and solubility in lung fluid of the aerosol should if possible be established. Any departure from reference man should also be assessed - the most likely being breathing rate.

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CHAPTER 15. DEVELOPMENTS IN THE MANAGEMENT OF RADIOACTIVE WASTE
FROM THE MINING AND MILLING OF RADIOACTIVE ORES

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ABSTRACT

The philosophy of a waste management system is discussed. The origins of the various wastes from the mining and milling processes are outlined and the development of a waste management program described. The technical aspects of a waste management plan, namely water management systems, waste rock and ore stockpile management, tailings impoundment and decommissioning and rehabilitation are discussed in detail.

A PHILOSOPHY OF WASTE MANAGEMENT

In Australia the mining and milling of radioactive ores has occurred since the early 1950's to produce yellow cake consisting principally of U_3O_8 . Mines were located in Northern Territory (eg Rum Jungle mill and associated mines, Moline mill and associated mines and Rockhole) and in South Australia at Radium Hill with the ore transported to Port Pirie for processing. The waste management practices carried out at these early facilities were not sensitive to their impact on the environment, although they may have been considered reasonable at the time. Practices have been improved markedly in recent years.

Significant in this improvement have been the increased awareness of environmental matters in general over the past two decades and a greater concern for the occupational health and safety of radiation workers. This is illustrated by the changing waste management practices adopted at the Rum Jungle Mine during its life. The debate was at its most public during the Ranger Uranium Environmental Inquiry (Fox Commission) in the late 1970's. Arising from the recommendations of the Inquiry were an integrated series of Environmental Requirements for the Ranger and Nabarlek uranium mines and the Commonwealth's Environment Protection (Nuclear Codes) Act (1978).

Three codes were developed under the Act namely:

- . Code of Practice on Radiation Protection in the Mining and Milling of Radioactive Ores. (1980) [the Health Code].
- . Code of Practice for the Safe transport of Radioactive Substances (1982) [the Transport Code].
- . Code of Practice on the Management of Radioactive Wastes from the Mining and Milling of Radioactive Ores (1982) [the Waste Management Code].

These codes are mutually dependent, with the Waste Management Code indicating the measures for handling of the Radioactive wastes arising from the mining and milling operations.

Central to this code is the requirement for the control of airborne and liquid wastes by the formulation of discharge limits or by the use of best practicable technology. In all recently approved uranium mines the formulation of discharge limits has come to be considered as a component of best practicable technology.

Best Practicable Technology

This concept has been defined in the code as that technology, from time to time relevant to a specific project, which enables radioactive wastes to be managed so as to minimise radiological risks and detriment to people and the environment, having regard to:

- (a) the achievable levels of effluent control and the extent to which pollution and degradation of the environment is minimised or prevented in comparable mining and milling operations elsewhere;
- (b) the cost of the application or adoption of that technology relative to the degree of radiological and environmental protection expected to be achieved by its application or adoption;
- (c) the evidence of detriment or lack of detriment to the environment after the commencement of mining or milling operations;

- (d) the location of the mine or mill;
- (e) the age of the equipment and facilities in use for mining and milling purposes and their relative effectiveness in achieving radiological and environmental protection; and
- (f) the potential hazards from the wastes over the long term.

This definition is similar to, but not the same as, definitions contained in other documentation including the environmental requirements for the current mines. This is because of the constraints of the bodies preparing the documents. The concept is open to considerable interpretation depending on the viewpoint adopted.

From an environment protection viewpoint, best practicable technology is concerned with outcomes. It is aimed at reducing the impact of the mining and milling process on the environment, the workmen and the general public for all periods of time, from the immediate short term during operation of the mine to the long-term when it is difficult to imagine what land use will be proposed, what population pressures there will be and what values civilisation will have. It is now seen as important to isolate all radioactive components from dispersive mechanisms of the environment for periods of thousands or hundreds of thousands of years and for the assimilation of environmentally hazardous components at a controlled rate which has no detriment. These are periods not generally considered in construction and so carry with them a requirement for measures which are conservative in their approach. In many ways, this conservative approach does not seem to sit well within the community's perception of improvement by the adoption of new technologies and the continued refinement of those technologies.

Best practical technology does imply the use of evolving technologies. However, these may be best used in defining the conservative measures to be used in management of wastes.

There cannot be a single best practical technology applicable to all situations. Each mine and mill will have different requirements, different site specific advantages and different site specific disadvantages. Best practical technology could then be considered as the planning and implementation process which is adopted to ensure the achievement of satisfactory outcomes.

However, central to and complimentary to best practical technology are the following concepts:

- . appropriate authority responsible for enforcement of legislation;
- . exposure to radiation of employees and members of the public from radioactive wastes in or released to the environment is as low as reasonably achievable; and
- . a restricted release zone being an approved zone about and below a mine or mill from which release of radioactive material shall be minimised in accordance with the requirements of the appropriate authority.
- . long-term a concept which gives perspective to measures to be adopted. In considering half lives of radioactive materials and the remaining radiation in wastes the period of hazard is very long. In engineering terms long-term implies the period following the operational and transitional phases where structures operate beyond their design life and to the full extent of the structural life of the engineered facilities.

There will be competing demands for limited resources (both economic and environmental) and in order to define measures to be adopted decisions will have to be made.

It is convenient to consider the decision making context in terms of belief about causes and the belief about outcomes. Decisions can then be classified into four areas with any combination of these areas constituting the process which leads to a complex decision.

These decisions contexts are:

	<u>Beliefs about</u> <u>causation</u>	<u>Preferences</u> <u>about possible</u> <u>outcomes</u>
. Rational (ie. computation)	Agreement	Agreement
. Bargaining (ie. compromise)	Agreement	Disagreement
. Problem solving (ie. judgement)	Disagreement	Agreement
. Dictatorial (ie. inspiration)	Disagreement	Disagreement

Each of these contexts is equally valid. However, in order to demonstrate the worthiness of a particular decision, presentations such as Environmental Impact Statements try to demonstrate the decision to have occurred in the rational context by providing the results of engineering studies, scientific research, benefit cost analysis etc. The decisions to be made regarding the viability of a proposed project are influenced by many factors. Primary amongst these are:

- . location of the ore body and
- . composition of the ore.

These are fixed and not variable. They determine other factors such as:

- . topography
- . geology
- . meteorology
- . land use
- . proximity or setting and
- . economic conditions

In developing a plan to mine and mill ore it is necessary to incorporate a sub-plan for the management of all wastes arising from the project so that all objectives are met. Those objectives must include:

- (1) the ability, on completion of all works, for the company mining and milling the ore to be relieved of all further responsibility;
- (2) for the site to be returned to a condition where a minimum of land use constraints need be applied; and
- (3) the need for ongoing monitoring and maintenance by authorities to be minimised.

The objectives should all be identified and considered as factors in determining the economic viability of a project during the planning stages so that all costs may be met by the project without passing an ongoing legacy to the community.

CONTENTS OF THE WASTE STREAM

Wastes generated from the mining and milling of radioactive ores generally include:

- (a) waste materials surrounding the ore which must be moved in order to expose the ore
- (b) materials added to during and by-products of the milling processes
- (c) residual components of the ore following extraction of the product and
- (d) materials which come into contact with the wastes.

This means that wastes are managed under three categories, namely:

- (a) tailings impoundment
- (b) water management and
- (c) waste rock (and ore stockpile).

At the end of these operations, along with the mine and mill, these facilities must be decommissioned and rehabilitated.

Wastes are a complex group of chemicals and include:

- . radioactive materials which may be in the solid, liquid or gas form
- . inert rock materials which may be eroded and provide sediments and turbidity in water
- . chemically reactive rock materials, such as the pyrite group of sulphide rich compounds, these can provide a source of reactive waste producing potential problems including acid mine drainage
- . chemicals added in the milling process, such as manganese and sulphate.

Wastes produced during mining of radioactive ores are mainly site specific and included waste rock, ore dust, mine drainage and seepage as well as liquid and airborne effluents. There are two processes used in uranium milling at present; namely, the acid leach process coupled with either solvent extraction or ion exchange and the alkaline leach process coupled with caustic precipitation. The acid leach process is the most widely used technology. The wastes can be categorised as shown in Table 1.

Mining and milling processes remove ore from its confined location, grind it to a fine sandy silt size to enable efficient extraction of uranium and then place the tailings in impoundments for disposal. The uranium contained in the ore contributes very little to the total radioactivity in the tailings where it is more exposed to the environment. If secular equilibrium exists, about 10 GBq of each isotope is associated with ore containing 1 tonne of uranium. Significant isotopes remaining are long-lived alpha emitters ^{226}Ra and ^{230}Th also ^{210}Pb , ^{210}Pb , ^{210}Po and ^{214}Po may appear in the wastes or the product.

^{222}Rn is the gaseous short-lived isotope released during the breakup of the ore during mining and crushing and grinding. This is associated with dust containing uranium and its decay products. ^{222}Rn will continue to be released after the operation has ceased.