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SECONDARY LIMITS OF EXPOSURE IN FACILITIES
HANDLING URANIUM

by

M. Raghavayya
Health Physics Division

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21	<i>Affiliation of author(s) :</i>	Health Physics Division, Bhabha Atomic Research Centre, Mumbai
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60	<i>Abstract :</i>	Annual limits of exposure and intake for radiation workers in nuclear installations have been recommended by the International Commission on Radiological Protection and the same have been adopted by the Indian Atomic Energy Regulatory Board for all the radionuclides of interest. The prescribed limits cannot be directly used for day to day radiation protection work. Hence secondary limits have to be derived for routine applications. The modeling steps may be simple in some situations and more complicated in some others. The limits recommended are for individual radionuclides. But in facilities handling natural or enriched uranium the radionuclides (isotopes of uranium and its decay products) generally occur together in specific ratios. Derivation of secondary limits has to take this into consideration. The present document is an attempt at deriving the secondary limits required for routine application in facilities handling uranium (Mine, mill, refineries and fuel fabrication etc.). Secondary limits of exposure have been derived in this document for air borne activity, activity in water, surface contamination and internal exposures.
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C O N T E N T S

Section	Title	Page No
1.	INTRODUCTION	1
2.	GENERAL	2
3.	Derived Air Concentration (DAC)	2
3.4.	Air Borne Uranium	3
3.4.3.	Chemical Toxicity	4
3.6.	Long Lived Alpha emitters in ore dust	5
3.7.	Radon & Progeny	8
3.7.3.	DAC for Radon Progeny	9
3.7.4.	DAC for Radon (EER)	9
4.	Drinking Water Concentration	9
4.3.	Uranium	10
4.4.	Associated Decay products	11
4.5.	Apportionment of Dose by Route of Exposure	11
5.	EXTERNAL EXPOSURE	12
5.2.	Beta & Gamma radiation	12
6.	DERIVED CONTAMINATION LEVEL (DCL)	12
6.2.	Contamination of Surfaces	13
6:2.1.	Alpha Contamination	13
6.2.2.	Beta Contamination	15
6.3.	PERSONNEL CONTAMINATION	17
6.4.	Contamination of Clothing	18
6.5.	Contamination of Footwear	19
6.6.	Contamination of Equipment	19
7.	INTERNAL EXPOSURE	20
8.	CONCLUSION	21
9.	ACKNOWLEDGEMENT	22
	Appendix I	24
	Appendix II	24
	Appendix III	24
	Appendix IV	25
	Appendix V	26

SECONDARY LIMITS OF EXPOSURE IN FACILITIES HANDLING URANIUM

M. Raghavayya

1. INTRODUCTION

- 1.1. Wherever open radioactive sources are handled such as in a uranium refining plant, radioactivity is bound to get air borne, surface and personnel contamination are bound to occur despite adoption of all precautionary measures. Permissible limits of contamination have been derived and prescribed to meet such situations [So 61]. The limits set forth in that document were derived on the basis of the exposure and intake limits adopted at that time. Further, the reference isotopes chosen were some of the most toxic viz. ^{239}Pu , and ^{90}Sr . Limits of exposure and intake have been since revised. Moreover, using the most toxic isotopes for reference may be over restrictive for situations where only less toxic isotopes are involved. It is therefore felt appropriate to rederive the secondary limits on more realistic bases for application in uranium handling facilities. With this in view the Derived Air Concentration (DAC), Drinking Water Concentration (DWC), Surface Contamination Limit (SCL) etc have been rederived in this document for use in such facilities. For facilities handling enriched or depleted uranium also the secondary limits can be derived using the same logic.
- 1.2. Although, in theory, uranium is only alpha active, beta & gamma activities due to rapid build up of the daughter products ^{234}Th and ^{234}Pa are also invariably present along with the uranium isotopes. In uranium mines and mills (ore processing facilities) other decay products of ^{238}U such as ^{230}Th , ^{226}Ra , ^{210}Pb and ^{210}Po may also be significant. These radionuclides will not significantly contribute to the air borne activity in refining plants because the uranium in the ore gets separated from its decay products in the milling process. But in all types of uranium handling facilities they affect drinking water because they will be mostly in the effluents discharged from the plant to the environment. In this document therefore only Drinking Water Concentrations have been computed in their case.

2. GENERAL

2.1. The following assumptions are made in deriving the DAC and DWC values.

- i. Exposure is through the specified route only. Where more than one route are actually involved, rule of mixtures has to be applied appropriately at the time of implementation.
- ii. Number of days of work in a year is 250 (5 days/week x 50 weeks). Number of hours an individual radiation worker works in the year is for 2000 hours ('T' = 8 hours/day x 250 days).
- iii. Breathing rate during work is $1.2 \text{ m}^3 \cdot \text{h}^{-1}$. Thus the total air breathed by the worker in a year during the work period is $2400 \text{ m}^3 (= V_a)$.
- iv. Water intake by Indian reference man is 4.05 l/day [Ra69]. The total water consumption during the year is $V_w (=1.48 \text{ m}^3)$
- v. Secondary limits of exposure & intake for members of public is based on an annual dose limit of $1 \text{ mSv} \cdot \text{y}^{-1}$. (The Annual Limit of Intake (ALI) of radioactive material therefore is 1/20th of ALI for Occupational exposure).
- vi. Default size of aerosols is $5 \mu\text{m}$ AMAD (IC 94).
- vii. The specific activity ($\text{Bq} \cdot \text{mg}^{-1}$) of uranium of different isotopic compositions is as given in Appendix I.

3. Derived Air Concentration (DAC)

3.1. ICRP [IC94] and IAEA [IA96] have specified the Effective Dose Coefficients ($\text{Sv} \cdot \text{Bq}^{-1}$) for intake of radionuclides by inhalation and ingestion. For intake by inhalation the dose coefficients have been listed for a default particle size. While the earlier publications had recommended a default particle size of $1 \mu\text{m}$ AMAD (Activity Median Aerodynamic Diameter) for the aerosols, ICRP-68 [IC94] has recommended the coefficients for $5 \mu\text{m}$ also since it is now believed that the latter is more realistic and representative of actual work atmosphere. ICRP now recommends the default value of $5 \mu\text{m}$ rather than $1 \mu\text{m}$. In this document the ALI values quoted have been derived using the dose coefficients for $5 \mu\text{m}$ AMAD particulates.

3.2. Dose coefficients recommended by ICRP are for individual radioisotopes even where they always occur as mixtures. For example, although uranium naturally exists always as a mixture of

^{238}U , ^{234}U and ^{235}U isotopes in specific proportions the dose coefficients given are for the isotopes individually. During radiological monitoring, results of air borne activity measurements or concentrations in water are generally obtained as total activity concentrations (Bq.m^{-3}) or as total mass concentration (mg.m^{-3}) of natural or enriched uranium. Moreover, uranium compounds handled in a given facility usually have the same isotopic compositions in most situations. Because of this, it is valid to express the concentrations in terms of the gross values. Keeping this in mind, DAC values have been derived for composite uranium rather than the individual isotopes.

- 3.3. Uranium compounds are classified into three types (see Appendix III) depending on lung deposition kinetics and metabolic clearance rates [IC94] viz. 'S (Slow)', 'M (Moderate)' and 'F (Fast)'. Expressed as approximate half-times for one or two components of clearance, these absorption rates correspond to

- * Type (Fast) 10 min (100%)
- * Type M (Moderate) 10 min (10%) to 140 d (90%)
- * Type S (Slow) 10 min (0.1%) to 7000 d (99.9%)

3.4. Air Borne Uranium

- 3.4.1. Let ' M_j ' be the composite ALI for a given uranium type ('S' 'M' or 'F' as the case may be) expressed in mg.y^{-1} . Let the specific activity (Appendix I, col.2,4,6) and ALI (Appendix IV) of the i^{th} uranium isotope be denoted by ' S_i ' and ' A_i '. Then by applying the rule of mixtures (see Appendix V for sample calculation),

$$\sum_{i=1}^3 \frac{S_i M_j}{A_i} = 1 \quad \dots [1]$$

from which M_j is obtained as

$$M_j = \frac{1}{\sum_{i=1}^3 [(S_i) / (A_i)]} \quad \dots [2]$$

The annual air intake by an occupational worker is (2000×1.2) 2400 m^3 . If the total specific activity of uranium (Appendix I, col. 7) is denoted by ' S ' (Bq.mg^{-1}) the DAC (D_j) is given by

$$D_j \text{ (Bq.m}^{-3}\text{)} = (M_j S) / V_a \quad \dots [3]$$

3.4.2. DAC may be calculated as shown above for uranium of any isotopic compositions and any type. In Table 1 below are listed DAC values calculated for a few typical compositions (both natural and enriched uranium) based on radiological considerations only.

**Table 1: DAC for uranium of different isotopic compositions
(calculated on the basis of radiotoxicity alone)**

Isotopic composition (%)			DAC (Bq.m ⁻³)			Remarks
²³⁵ U	²³⁸ U	²³⁴ U	Type F	Type M	Type S	
0.400	99.597	0.003	13.9	4.8	1.4	Depleted U
0.720	99.275	0.005	13.6	4.5	1.3	Nat. U.
1.000	98.992	0.007	13.6	4.4	1.3	Enriched U
2.000	98.980	0.019	13.3	4.2	1.3	-do-
3.000	96.966	0.033	13.2	4.1	1.2	-do-
4.000	95.950	0.049	13.2	4.1	1.2	-do-
5.000	94.933	0.066	13.1	4.0	1.2	-do-
6.000	93.915	0.084	13.1	4.0	1.2	-do-

3.4.3. Chemical Toxicity

3.4.3.1. The ALI and DAC values may vary depending on whether chemical toxicity of the uranium compound of interest is more restrictive or its radiotoxicity and also on the particle size of the aerosols.

3.4.3.2. Chemical toxicity: ICRP-54 document [IC89] states "...intake of more transportable uranium compounds are limited by considerations of chemical toxicity rather than radiation dose. The earlier recommendation has been adopted in the Euratom Directive of Basic Safety Standards [OJ80]. This limits the intake of soluble compounds by inhalation in any one day to 2.5 mg. regardless of isotopic composition. For continuous occupational exposure, the American Conference of Governmental Industrial Hygienists (ACGIH) [AC98] has recommended a Threshold Limit Value of 0.2 mg.m⁻³ for concentration of more soluble uranium in air. A subsidiary short-term exposure limit of 0.6 mg.m⁻³ is permitted for period of 15 minutes. In the U.K. the Health and Safety Executive has adopted the same values [HS80]'.

3.4.3.3. For a standard breathing rate of 1.2 m³.h⁻¹, a continuous concentration of 0.2 mg.m⁻³ corresponds to an average daily intake

of 2 mg (during the 8 hour work period) i.e. an annual intake of 500 mg (250 days) or 12500 Bq which is more than the ALI for Type M & S uranium compounds but less than for Type F compounds based on radiotoxicity alone. Thus for natural uranium of Type F limit based on chemical toxicity may be adopted while for Type M & S, limits based on radiotoxicity may be adopted.

- 3.4.3.4. Thus the DAC for Type F natural uranium compounds and moderately enriched uranium, must be adopted on the basis of chemical toxicity.

TLV (Uranium type F): 0.2 mg.m⁻³ (= 5 Bq.m⁻³)

The deposition of the aerosols in the respiratory system is controlled by the particle size characterised by the AMAD. Air Borne Beta Activity

- 3.5.1. The beta activity in a uranium refining plant and facilities handling purified uranium is due to the decay products of ²³⁸U viz. ²³⁴Th and ²³⁴Pa. ²³⁴Pa being very short-lived (Half-life - 1.17 m) is always in secular equilibrium with ²³⁴Th. The DAC for air borne ²³⁴Th beta activity works out to 1570 Bq.m⁻³ for both Type M & S compounds. The half-life of ²³⁴Th (24.1 d) being very short compared to the half-life of uranium, the beta activity will always be either equal to or less than the alpha activity. The DAC figures for alpha activity are very much lower than those for beta activity. Thus if the air borne alpha activity is kept within the limits, beta activity will always remain so.

3.6. Long Lived Alpha emitters in ore dust

- 3.6.1. The refining plant, fuel fabrication Unit and such other facilities handle uranium in a relatively pure state either as compounds or as metal. It is only in the mine and mill that uranium ore is handled in large quantities. During the mining and milling process fine ore dust gets air borne despite actions taken for dust suppression such as wetting. The grade of the ore processed in Indian mines and mill is quite low because of which the air borne radioactivity is also expected to be in low concentrations. One should remember that unlike in the later stages, the air borne dust also contains,

in addition to uranium, its long-lived decay products. But it is unlikely that the other alpha emitting long-lived decay products of uranium will be airborne as separated radionuclides. It may be assumed that wherever uranium ore is handled ore dust containing the mixture of radionuclides gets air borne. It is reasonable to assume that the equilibrium status of the ore is maintained in the air borne dust also. But due to preferential separation, the air borne dust is likely to be richer in uranium (and also decay products) than the bulk ore from which the dust was created. This is discussed in greater detail later in Appendix V.

3.6.2. The long lived alpha emitting radionuclides in the ore are natural uranium (^{238}U , ^{234}U & ^{235}U), ^{230}Th , ^{226}Ra , ^{210}Po & ^{231}Pa (long lived alpha emitting decay product of ^{235}U). The 238 and 234 uranium isotopes of course are always in secular equilibrium. The other radionuclides may or may not be in secular equilibrium with the parent uranium isotopes depending on the ore characteristics especially age and the chemical species. The activity contribution from 235 isotope of uranium is small (~2% of total long-lived alpha activity in the ore). Thus there are seven long-lived alpha emitters contributing to the radioactivity of the ore. The DAC for uranium ore based on radiological considerations can be calculated as shown in section 3.4.1, applying the rule of mixtures.

3.6.3. It is difficult to determine to which type classification (type **S**, **M** or **F**) uranium and other radionuclides in the ore belong because they are in complexed form. Most of the uranium in the ore is relatively insoluble and therefore the more restrictive type 'S' may be assumed for the bulk of uranium in the ore. The ore usually also has a small fraction of more soluble component, (exact fraction varies from ore to ore) which may be assumed to be of Type 'M' or even type 'F'. The choice will depend on the chemical species of the radionuclides in the ore, which may be different for different ores. Some of the decay products under consideration belong to only one or two of the types. While ^{226}Ra is classified only into 'M' type only, ^{230}Th & ^{231}Pa are classified into type 'M' & 'S' and ^{210}Po is classified into type 'F' & 'M'. In view of these uncertainties, the most restrictive of each of the radionuclides may be used for deriving the DAC for the uranium ore.

3.6.4. The amounts of the decay products in the ore are controlled by the amount of uranium. The activity of any of the long lived alpha emitter in the ore may be expressed in terms of its parent activity as

$$S_{ji} = \frac{F_{ji} S_j P}{100}$$

where

S_{ji} is the specific activity of the i^{th} decay product of the j^{th} uranium isotope in the ore. It is expressed as Bq.mg^{-1} of the ore.

P is the ore grade (percent uranium. Some times the ore grade is expressed in percent of U_3O_8 equivalent. Then appropriate correction for uranium equivalence has to be applied. If P' is the ore grade in percent of U_3O_8 equivalent, then $P = 0.85P'$).

S_j is the specific activity of the parent uranium isotope ^{238}U or ^{235}U (Bq.mg^{-1} of natural uranium - see Appendix I).

F_{ji} is the equilibrium fraction of the radionuclide in question. The value of F_{ji} is unity for ^{238}U , ^{234}U and ^{235}U . The value will vary from zero to 1 for the decay products of uranium depending on the equilibrium status in the ore.

3.6.5. Let ' M_0 ' be the composite ALI (mg.y^{-1}) of ore dust. Let the ALI (Appendix IV) of the i^{th} decay product of j^{th} parent uranium isotope be denoted by ' A_{ji} ' (Bq.y^{-1}). Applying the rule of mixtures as in equation [1]

$$\left[\sum_{j=1}^5 \frac{F_{ji} S_j P M_0}{100 A_{ji}} + \sum_{j=2}^2 \frac{F_{ji} S_j P M_0}{100 A_{ji}} \right] = 1 \quad \dots \dots [4]$$

from which M_0 is obtained as

$$M_0 (\text{mg.y}^{-1}) = \frac{100}{P \left[\sum_{j=1}^5 \frac{F_{ji} S_j}{A_{ji}} + \sum_{j=2}^2 \frac{F_{ji} S_j}{A_{ji}} \right]} \quad \dots \dots [5]$$

The DAC for ore dust is calculated by dividing equation [5] by the volume of air breathed by an occupational worker in an year. Or

$$D_0 (\text{mg.m}^{-3}) = (M_0)/V_a \quad \dots \dots [6]$$

The DAC in Bq.y^{-1} units is obtained by multiplying the above DAC with the specific activity of the ore dust $S_0 (\text{Bq.mg}^{-1})$ where

$$S_0 = P \left[\sum_{j=1}^5 \frac{F_{j1} S_j}{100} + \sum_{j=2}^2 \frac{F_{j1} S_j}{100} \right] \dots [7]$$

A typical calculation is shown in Appendix V.

3.7. Radon & Progeny

- 3.7.1. Among Facilities handling uranium, exposure to radon and its short-lived decay products is significant only in a uranium mine and the mill where the ore is processed. The organ most exposed when radon progeny is inhaled is the lung. Dose received by other organs of the body due to dissolution of radon progeny in body fluid and subsequent translocation is only about 2% of the total effective dose [IC93]. Exposure to radon progeny is expressed in time integral of either the radon progeny concentration (Working Level Month, WLM) or the radon progeny potential alpha energy concentration (PAEC in mJ). ICRP assumes 80% occupancy (7000 hours/y) indoors and 20% (2000 hours/y) at work. The conversion from cumulative exposure to equivalent dose (Sv) depends on several factors, such as the lung model chosen, site of deposition, clearance rates etc. In its earlier recommendations (IC81, IC86) the Commission considered three lung models viz. the JACOBI EISFELD model, the JAMES-BIRCHALL model and the ICRP lung model (old). Based on these models and also on the then available epidemiological data, an annual limit of 4.8 WLM was recommended by ICRP for occupational workers as corresponding to the then annual effective dose equivalent limit of 50 mSv. On the other hand while making the current recommendations (IC93) the epidemiological approach has been preferred for setting the annual limits for radon. The modified multiplicative risk projection model (ICRP, 91) has been used to arrive at the fatality coefficients for both males and females. The value used is 8.0×10^{-5} per mJ.h.m^{-3} (3×10^{-4} per WLM).
- 3.7.2. Conversion factors of 1.43 mSv per mJ.h.m^{-3} (5.06 mSv/WLM) for occupationally exposed workers and 1.10 mSv per mJ.h.m^{-3} (3.88 mSv/WLM) in the case of general public have been used by ICRP in their current recommendations based on epidemiological

considerations [IC93]. Thus the annual limits of exposure to workers would be as shown in Table 2.

**Table - 2. Annual Exposure Limits (Rn Progeny)
for occupational workers**

Annual Limit			Reference
Effective Equivalent Dose	Radon Progeny		
(mSv)	(mJ.h.m ⁻³)	(WLM)	
20 (5 year average)	14	4	IC90, IC93

3.7.3. DAC for Radon Progeny

3.7.3.1. The DAC for radon progeny can be calculated as follows:

$$\begin{aligned} \text{Annual exposure Limit} &= 4 \text{ WLM} = 4 \times 170 = 680 \text{ WLH.} \\ \text{No. of hours of exposure} &= 2400 \text{ h.y}^{-1} \end{aligned}$$

$$\text{DAC} = 680/2400 = 0.28 \text{ or say } 0.3 \text{ WL.}$$

3.7.4. DAC for Radon (EER)

3.7.4.1. In work places it is often simpler and easier to measure the concentration of radon in air rather than the concentration of radon progeny. A given concentration of radon is always associated with its progeny, although usually not in equilibrium. If the equilibrium factor (F) is known, the radon progeny DAC can be converted to the corresponding Equilibrium Equivalent Radon (EER) concentrations. ICRP has recommended a default equilibrium factor of 0.4 when the actual measured value is not available.

3.7.4.2. Radon progeny concentration of 1 WL corresponds to a radon concentration of 3.7 kBq.m⁻³ (EER). DAC for Radon Progeny = 0.28 WL
Hence DAC for radon (EER) = 3700 x 0.28 = 1036 or say 1000 Bq.m⁻³

4. Drinking Water Concentration

4.1. Safe, treated drinking water is provided as a matter of course in work places. Hence setting a Drinking Water Concentration limit (for uranium) for occupational exposure does not arise. But this is not the case where members of the public are concerned. In many localities, safe, treated drinking water is scarce. Moreover, consumption of water in public is difficult to control. Therefore

setting a limit for concentration of uranium and other associated radionuclides in drinking water for the public domain is very much relevant. The drinking water limits for occupational workers is no different from those for general public since in most cases both the groups obtain their drinking water from the same source.

4.2. When considering intake by ingestion, uranium isotopes have been classified into two Types only, viz. Type 'M' & 'F'. The ALI for Type 'M' & 'F' natural uranium isotopes are given in Appendix IV. Composite ALI (Ingestion) for uranium may be computed in the same manner as shown in section 3.4.1 above.

4.3. Uranium

4.3.1. Let 'W_j' (mg.y⁻¹) the composite ALI for a given uranium type ('M' or 'F' as the case may be). Let the specific activity (Appendix I, col.2,4,6) and ALI (Appendix IV) of the ith uranium isotope be denoted by 'S_i' and 'A_i' respectively. The ALI value adopted for members of public must be 1/20th of the occupational ALI values since the annual dose limit for the public is 1/20th of the occupational dose limit. Then by applying the rule of mixtures (see Appendix V for sample calculation),

$$20 \sum_{i=1}^3 \frac{S_i W_j}{A_i} = 1 \quad \dots [8]$$

from which W_j is obtained as

$$W_j \text{ (mg.y}^{-1}\text{)} = \frac{1}{20 \sum_{i=1}^3 [(S_i)/(A_i)]} \quad \dots [9]$$

The annual water intake by Indians is about 1.48 m³ (V_w) [Ra69]. Thus the Derived Water Concentration (DWC_j) for a specific type of uranium would be

$$\text{DWC}_j \text{ (mq.m}^{-3}\text{)} = (W_j)/(V_w) \quad \dots [10]$$

DWC_j works out to 3400 mg.m⁻³ and 600 mg.m⁻³ respectively for type M and type F uranium (see Appendix V). Since the value for Type F uranium compounds is more restrictive the latter figure viz. 600

mg.m^{-3} may be adopted as the basis for prescribing the Drinking Water Concentration 'DW,' for uranium.

4.4. Associated Decay products

- 4.4.1. The uranium series has, other than uranium, four long-lived decay products which are alpha and beta emitters. These radionuclides also may be found in the water sources. Derived Water Concentrations in respect of the decay products of uranium may also be calculated as above by choosing the appropriate ALI values (Appendix IV). Thus the DWC values for the four long-lived decay products considered work out to

^{230}Th	...	2.7 kBq.m^{-3}	^{226}Ra	...	2.0 kBq.m^{-3}
^{210}Po	...	2.3 kBq.m^{-3}	^{210}Pb	...	0.8 kBq.m^{-3}

4.5. Apportionment of Dose by Route of Exposure

- 4.5.1. It is customary to apportion the dose limit to the public according to the route of exposure viz. air, water, food chain etc. Since the water route in the present case is the most critical, reasonably it may be assumed that 50% of the exposure is due to the ingestion route (i.e. due to the radionuclides in water), 30% of the exposure is due to the air route and 20% as due to all other routes. In other words, of the 1 mSv exposure to the public 0.5 mSv can be allocated as due to the radionuclides in water. We have considered 5 radionuclides {U (nat.), ^{230}Th , ^{226}Ra , ^{210}Po and ^{210}Pb }. For simplicity it may be assumed that each of the radionuclides may account for a dose commitment of 0.1 mSv. The DWC figures given above correspond to a dose commitment of 1 mSv each. Hence the DWC to be adopted must be 1/10th of these figures given above in sections 4.3.1 & 4.4.1 after appropriate rounding off. The drinking water limit which may be adopted are therefore

U (nat)	..	60 mg.m^{-3}	^{230}Th	..	300 Bq.m^{-3}	^{226}Ra	..	200 Bq.m^{-3}
^{210}Po	..	200 Bq.m^{-3}	^{210}Pb	..	80 Bq.m^{-3}			

- 4.6. The recommended DAC & DWC values for uranium of different isotopic compositions are summarised later in Table 6. The lower of the values considering both chemical toxicity and radiotoxicity are

chosen in the case of airborne activity. DAC values quoted are for default particle size of 5 μm AMAD. The DAC for radon and progeny are given in Table 7.

5. EXTERNAL EXPOSURE

5.1. In facilities handling uranium in a purified form the external radiation exposure stems from the immediate decay products of ^{238}U viz. ^{234}Th and ^{234}Pa . They emit both beta and gamma radiation. Where open sources are involved, exposure of skin to beta radiation cannot be ruled out. The beta radiation will result mostly in a shallow dose to the skin, hands, and feet. Derived limits of external exposure are therefore computed on consideration of exposure of skin, hands, and feet in the case of beta radiation and whole body exposure in the case of gamma radiation.

5.2. Beta & Gamma radiation

Annual dose limit: 20 mSv.y^{-1} (whole body)
500 mSv.y^{-1} (skin, hands & feet).

No. of hours worked = 40 x 50 = 2000 hours.

Derived external exposure limit: γ : 10 $\mu\text{Gy.h}^{-1}$ (1 mR.h^{-1})
 β : 250 $\mu\text{Gy.h}^{-1}$ (25 mR.h^{-1})

5.2.1. Most of the measuring instruments in use at present are yet to be calibrated in $\mu\text{Gy.h}^{-1}$ units. Until new sets of instruments are available or the existing ones are recalibrated, the old unit mR.h^{-1} will continue to be used. Considering the conversion factors one should remember that the above derived exposure limit would be equivalent to 1.15 mR.h^{-1} and 28.75 mR.h^{-1} respectively. However in view of the uncertainties involved in calibration, and since in any case the error in assuming the equivalence $10 \mu\text{Gy.h}^{-1} = 1 \text{mR.h}^{-1}$ is insignificant, it is not necessary to be very rigid in conversion and the derived external exposure limits may also be expressed as 1 mR.h^{-1} (γ) and 25 mR.h^{-1} (β).

6. DERIVED CONTAMINATION LEVEL (DCL)

6.1. Surface contamination is a potential source air borne activity as well as internal exposure of workers. Surface/personal

contamination is not desirable since it is likely to result directly or indirectly in exposure of personnel. In extreme cases body contamination may even result in significant external exposure of contaminated personnel. In uranium handling facilities therefore, alpha as well as beta surface contamination has to be given due attention. For deriving the surface contamination levels, the following assumptions are made.

- i. Number of working days (days of exposure) in a year is 250
- ii. Intake of uranium is chronic and uniform. Daily intake by way of inhalation or ingestion takes place at the rate of 1/250 of the Annual Limit of Intake. The intake and DAC figures for the relevant radionuclides are as given in Table 3 below (Ref. Sec. 3 & 4, also Appendix V(3)).
- iii. The Annual dose limit for skin and extremities of the body (hands & feet) is 500 mSv.
- iv. The average thickness of human skin is 7 mg.cm⁻².
- v. The surface area of the hands is about 300 cm².
- vi. An additional safety factor of 10 is used.

Table 3. DAC and Daily Limits of Intake (DLI) for uranium and ²³⁴Th.

Radio-nuclide	Radiation	DAC (Bq.m ⁻³)			Daily Limit of Intake (Bq.d ⁻¹)	
		By Inhalation			By Ingestion	
		Type S	Type M	Type F	Type M	Type F
²³⁴ Th	β + γ	1440	1570	-	23520	23520
U (nat)	α	1.3	4.5	5.0	10040	1720

6.2. Contamination of Surfaces

It is assumed that contamination of surfaces is extensive and loosely bound and that the radiological hazards are due to

- a. External exposure
- b. Inhalation and,
- c. Ingestion.

6.2.1. Alpha Contamination

6.2.1.1. External Exposure: Alpha rays from uranium do not constitute an external radiation hazard. This is therefore neglected.

6.2.1.2. Inhalation: The contamination, since it is extensive and loosely attached to the surface, is a potential source of airborne

activity. Such contamination on surfaces may be correlated with airborne activity by means of a "Resuspension Factor [Cl70 defined as "the ratio of airborne contamination (Bq.cm^{-3}) to surface contamination causing it (Bq.cm^{-2})". This factor has the dimension of cm^{-1} and the mean value quoted in literature is $5 \times 10^{-7} \text{ cm}^{-1}$. The surface contamination which will give rise to an airborne activity of natural uranium (Type S) equal to the DAC i.e. $1.3 \times 10^{-6} \text{ (Bq.cm}^{-3})$ is therefore

$$C = \frac{1.3 \times 10^{-6} \text{ (Bq.cm}^{-3})}{5 \times 10^{-7} \text{ (cm}^{-1})} = 2.6 \text{ Bq.cm}^{-2} \quad \dots [11]$$

Introducing a safety factor of 10, the derived contamination level (DCL) for surface works out to 0.26 or 0.3 Bq.cm^{-2} in respect of Type S uranium. Similarly the DCL for both Type M and Type F uranium compounds would be 1.0 Bq.cm^{-2} .

6.2.1.3. Ingestion: It is assumed that the radiation worker washes himself/herself thoroughly with soap and water at the end of work and that the easily removable loose contamination is washed off the hands. What may therefore be carried home after work is contamination fixed on the hands which is later completely ingested, although gradually. This residual fraction is about 10% [Jh74] of the total contamination on the hands. The residual contamination (fixed) therefore shall not exceed the DLI for the reference isotope.

When considering ingestion uranium compounds are divided into Type M and Type F only. The DLI for Type M uranium compounds is $1.004 \times 10^4 \text{ Bq.d}^{-1}$ (above Table 3 in section 6.1.). Since this is the contamination remaining after a normal wash, the original contamination on the hands can be $1.004 \times 10^5 \text{ Bq}$. The surface contamination from which this amount of hand contamination results can be calculated on the basis of the experimental findings of Rohr & Baily [Ro63]. They found that a surface contamination level of 1.833 Bq.cm^{-2} (11000 dpm/100 cm^2) results in a saturation hand contamination of 1383.3 Bq (83000 dpm). Thus the surface contamination which ultimately results in the hand contamination of $1.004 \times 10^5 \text{ Bq}$ is given by

$$C = \frac{1.004 \times 10^5 \times 1.833}{1383.3} = 133.2 \text{ Bq.cm}^{-2} \dots [12]$$

Introducing a safety factor of 10, the derived surface contamination level (DCL) works out to 13.2 or say 13 Bq.cm⁻² in respect of Type M uranium. Similarly the DCL for Type F uranium compounds would be 2.3 Bq.cm⁻² or say 2 Bq.cm⁻².

6.2.2. Beta Contamination

6.2.2.1. External Exposure: The annual dose limit to the skin is 500 mSv. The skin dose rate at a depth of 7 mg.cm⁻² (average thickness of the epidermal layer of the skin) due to beta rays of average energy $E_\beta > 0.3 \text{ MeV}$ is given by [Un73]

$$D_7 (\mu\text{rad/h}) = \frac{393.4 E_\beta + 17.7}{E_0} \dots [13]$$

when the beta surface contamination is 1 Bq.cm⁻² and E_0 is the weighted average of the end point energies of all the different betas emitted by the isotope. E_β may be calculated from [IC59]

$$E_\beta (\text{MeV}) = 0.33E_0 [1 - 0.02 Z^4] [1 + 0.25 (E_0)^4] \dots [14]$$

The reference isotope for beta contamination is ²³⁴Th + ²³⁴Pa. Then $E_0 = 2.22 \text{ MeV}$ and $E_\beta = 0.81 \text{ MeV}$ and from (14) D_7 works out as $151.5 \mu\text{rad.h}^{-1}$ ($=1.515 \times 10^{-3} \text{ mSv}$). The skin contamination level, which results in the annual skin dose of 500 mSv, can be calculated as:

$$C_{\text{skin}} = \frac{500}{1.515 \times 10^{-3} \times 24 \times 365} = 37.67 \text{ Bq.cm}^{-2} \dots [15]$$

The surface area of the hand is about 300 cm². The total contamination on the hands is therefore 11300 Bq. The beta surface contamination that gives rise to this skin contamination on repeated contact is derived based on the observations of Rohr & Baily using as before in section 6.2.1.3 (equation 12). After

applying the safety factor of 10, DCL works out to 1.5 Bq.cm^{-2} for beta activity based on external exposure considerations.

6.2.2.2 Inhalation: Proceeding as before the surface contamination producing an air activity equivalent to the DAC of ^{234}Th of both type M and type S (section 3.5.1) i.e. 1570 Bq.m^{-3} is calculated as in section 6.2.1.2 (equation 11). After applying the safety factor of 10, the DCL works out to 314 or say 310 Bq.cm^{-2} based on consideration of inhalation.

Ingestion: The DLI of ^{234}Th (both Type M and Type F) 23520 Bq. Proceeding as in section 6.2.1.3 (using equation 12) and introducing the safety factor, the DCL (beta) works out to 31.17 or say 30 Bq.cm^{-2} based on consideration of ingestion.

6.2.3. The derived surface contamination levels considering external exposure, inhalation and ingestion as the basis for calculation, the alpha and beta contamination levels are listed in Table 4.

Table 4. Derived Surface Contamination Levels considering different types of exposure

Type of radiation & reference nuclide	DCL (Bq.cm^{-2})		
	Ext. Exposure	Inhalation	Ingestion
Alpha (U - Type S)	N.A.	0.3	N.A.
Alpha (U - Type M)	N.A.	1.0	13.0
Alpha (U - Type F)	N.A.	1.0	2.0
Beta (^{234}Th)	1.5	310.0	30.0

As a matter of principle when several values are available for the derived limits of the same parameter the lowest is chosen for application to be on the safe side. Thus the DCL for alpha contamination is 0.3 Bq.cm^{-2} for Type S uranium and 1.0 Bq.cm^{-2} for Type M & F uranium. Although the DCL chosen should be 1.5 Bq.cm^{-2} for beta contamination, in reality the beta activity in uranium facilities is due to the ^{234}Th & ^{234}Pa isotopes which are the short lived daughters of uranium and are usually in secular equilibrium with the parent. When secular equilibrium exists the alpha and beta activities are numerically equal. The presence of beta activity in surface contamination at any location therefore means that an equivalent alpha activity is also concurrently present at that location. In view of this the DCL for beta radiation in

uranium facilities should also be numerically equivalent to the figures for Alpha Contamination viz. 0.3 Bq.cm^{-2} and 1.0 Bq.cm^{-2} depending on the type of uranium involved.

6.3. Personnel Contamination

6.3.1. The derivation of derived personnel contamination levels (DCL) is also dealt with in a way similar to surface contamination. All possible types of hazards due to the reference isotope are considered. In this context, hand and skin contamination are of interest.

6.3.2. Inhalation: Direct inhalation of skin contamination is trivial since loose contamination on skin is easily removed by washing [Du64]. There is however a finite probability of the contamination remaining on the hands being transferred to tobacco in cigarette smoking which may cause inhalation of contamination. However no more than 0.1 percent of the contamination is likely to be thus inhaled and hence is not really significant [Ro53].

6.3.3. Absorption: Absorption of radioisotopes into the system through skin has been studied by Fink [Fi50]. He has observed that no more than 2 percent of the total skin contamination is absorbed into the system.

6.3.4. Ingestion: Direct ingestion of contaminants during work is negligible because consumption of food, drinks and tobacco etc. in active areas is statutorily prohibited by regulations. Therefore only indirect ingestion is of considerable importance. Since most Indians eat with their hands and fingers, contamination of hands is potentially the cause of indirect Ingestion. Although the average surface area of the hand is about 300 cm^2 , residual contamination is more likely to be confined to a smaller area of 30 cm^2 per hand; particularly finger tips and lower edges of the hands [Du64]. While considering ingestion, no difference is made between hands and the rest of the body

6.3.5. Alpha Contamination: The DLI for Type M uranium is $1.004 \times 10^4 \text{ Bq}$. The levels of hand and skin contamination (fixed) can be such as not to exceed the DLI.

$$C = \frac{1.004 \times 10^4}{30 \times 2} = 167.3 \text{ Bq.cm}^{-2} \dots [16]$$

Introducing the safety factor of 10 and rounding off the DCL for hands works out to 17 Bq.cm⁻² or say 15 Bq.cm⁻². Similarly for Type F uranium the corresponding DCL figure would be 3 Bq.cm⁻². It would be difficult to determine if the contamination on hands is of type M or type F uranium in most cases. Therefore the lower DCL of 3 Bq.cm⁻² for hands may be adopted.

- 6.3.6. Beta Contamination: The DLI for the beta reference isotope is 2.352 x 10⁴ Bq. The DCL for hand/skin contamination can be calculated as in section 6.3.5. With the safety factor of 10 introduced the DCL works out to 40 Bq.cm⁻².
- 6.3.7. External Exposure: This is applicable only in the case of beta contamination. In sec. 6.2.2.1 it was seen that the skin contamination which delivers the annual dose of 500 mSv is 37.7 Bq.cm⁻². With the safety factor of 10 the DCL for hands and skin considering external exposure as the basis would then be 3.77 Bq.cm⁻² or say 4 Bq.cm⁻².
- 6.3.8. In the case of alpha contamination the critical hazard is ingestion (3 Bq.cm⁻²) and in the case of beta contamination it is external exposure (4 Bq.cm⁻²). However since the beta activity is expected to be numerically same as alpha activity, the lower the two figures viz. 3 Bq.cm⁻² may be adopted as applicable to both situations.

6.4 Contamination of Clothing

- 6.4.1. Contamination of clothing also is treated in the same manner as for skin since clothing is always in close contact with the skin. This implies that the DCL for clothing also should be 3 Bq.cm⁻². A radiation worker wears work clothes for 40 hours a week and personal clothes for the remaining period of 128 hours. Applying these qualifying time factors, the skin contamination level of 3 Bq.cm⁻² will correspond to the following clothing contamination.

$$\text{Work clothes: } 3 \times \frac{168}{40} = 12.6 \text{ or say } 13 \text{ Bq.cm}^{-2}.$$

$$\text{Personal clothes: } 3 \times \frac{168}{128} = 3.9 \text{ or say } 4 \text{ Bq.cm}^{-2}.$$

These figures imply that both the work clothes and personal clothing will contribute the same dose to the skin. In order to allow only one equivalent of skin dose, half of the above figures may be adopted as the DCL for clothing. Thus the DCL shall be 6.5 Bq.cm⁻² and 2 Bq.cm⁻² for work clothes and personal clothes respectively.

6.5 Contamination of Footwear

- 6.5.1. Risk due to inhalation and ingestion of contamination from footwear may be neglected for obvious reasons. The only significant hazard is therefore due to external exposure. This being the case it is enough to prescribe the DCL for beta activity since alpha activity does not pose an external hazard. The DCL should therefore be 4 Bq.cm⁻² (see section 6.3.7.). Applying the appropriate qualifying time factors as in section 6.4.1. The DCL shall be 8 Bq.cm⁻² and 3 Bq.cm⁻² for work clothes and personal clothes respectively.

6.6 Contamination of Equipment

- 6.6.1. The equipment used in an operating plant also get contaminated. The contaminated equipment are usually decontaminated before they are handed over for maintenance etc. Since at times it may not be possible to completely decontaminate them due to construction or other equipment features, it is worthwhile to derive a surface contamination level for such equipment. In doing so activity on the equipment getting airborne and causing significant air borne activity may be neglected since the ratio of surface area of the equipment to the volume of the room is likely to be very low and the resuspension factor adopted in this document may not be valid. The critical hazards are therefore limited to ingestion in the case of alpha contamination and external exposure in the case of beta contamination as can be seen from Table 4. The figures are 2 Bq.cm⁻² and 1.5 Bq.cm⁻² respectively. The lower of the two values viz. 1.5 Bq.cm⁻² is chosen assuming that alpha and beta activities would be in secular equilibrium on the equipment surfaces also.

6.7. The levels of surface and personnel contamination recommended for adoption are summarised later in Table 8. It must be remembered that these figures are applicable to only installations handling natural uranium. Moreover in view of the number of assumptions made one should be careful in applying these limits. These figures should be taken as only guidelines, while every effort must be made to ensure that contamination of floors, equipment and persons is avoided.

7. INTERNAL EXPOSURE

7.1. Intake of uranium (chronic or acute) by workers is generally monitored by means of bioassay. For this purpose, overnight urine samples are collected and analysed for uranium. The uranium content of the urine sample may be correlated with the intake.

7.2. The fraction of uranium intake found in urine depends on the frequency of sampling [IC69]. The frequency of monitoring should depend on the type of uranium compounds handled. For instance, for Type S (formerly class y) compounds the predicted activity in urine as a fraction of intake is 1.8×10^{-5} when monitoring interval is 360 days (annual sampling). The ALI for Type S compounds of natural uranium is 125 mg or 3125 Bq (see Appendix V). Thus under above conditions a daily excretion rate of 2.25 $\mu\text{g U}$ will correspond to an intake equal to 1 ALI. Table 5 gives predicted urinary activity under various intake scenarios.

Table 5. Predicted activity in urine.

Type of U	ALI (5 $\mu\text{m AMAD}$)		Monitoring interval (d)	Predicted daily urinary excretion		
	(Bq)	(mg)		as fraction of intake	corresponding to ALI	
					($\text{Bq}\cdot\text{d}^{-1}$)	($\mu\text{g}\cdot\text{d}^{-1}$)
S	3125	125	360	1.8×10^{-5}	0.056	2.25
M	12000	480	180	2.7×10^{-4}	3.24	130
F	10750	430	30	4.7×10^{-3}	50.5	2020

7.1.2. In practice, it is very difficult to maintain the required frequency of urine sampling as recommended by ICRP due to practical difficulties. So until a better system is evolved, the former equilibrium uranium body burden indicated by the tolerance limit of 50 $\mu\text{g}\cdot\text{l}^{-1}$ of urine may be continued to be adopted.

8. CONCLUSION

- 8.1. The secondary limits derived here are mutually exclusive. When more than one route of exposure is involved the rule of mixtures has to be applied. In a given facility, exposure to all three types of uranium compounds may or may not occur. For example, in a mill processing uranium ore to obtain a chemical concentrate (MgU_2O_7 , -MDU or 'yellow cake') only type S and type M uranium compounds are present. In a refining plant both type S and type M uranium may be involved. And in a conversion plant or fuel fabrication facility type M and type F compounds are handled. Thus the secondary limits appropriate to the different situations have to be used. When in doubt or where a clear demarcation is not possible, it is customary to adopt the more restrictive limits.
- 8.2. All the secondary limits derived in this document are summarised in Table 6 (DAC, DWC), Table 7 (radon & progeny) Table 8 (DCL) and Table 9 (external radiation and bioassay tolerance limit)

Table 6: DAC & DWC for uranium & decay products

Radio-nuclide		DAC						DWC	
Name	% of ^{235}U	Type F		Type M		Type S		(mg.m ⁻³)	(Bq.m ⁻³)
		($\mu g.m^{-3}$)	(Bq.m ⁻³)	($\mu g.m^{-3}$)	(Bq.m ⁻³)	($\mu g.m^{-3}$)	(Bq.m ⁻³)		
U	0.4	*200.0	3.6	267.2	4.8	77.5	1.4		
-do-	0.72	*200.0	5.0	180.2	4.5	53.1	1.3	60.0	1500
-do-	1.00	*200.0	6.1	143.2	4.4	42.6	1.3	-	-
-do-	2.00	*200.0	11.7	71.8	4.2	21.7	1.3	-	-
-do-	3.00	144.8	13.2	45.1	4.1	13.7	1.2	-	-
-do-	4.00	103.0	13.2	31.9	4.1	9.8	1.2	-	-
-do-	5.00	78.4	13.1	24.2	4.0	7.4	1.2	-	-
-do-	6.00	62.4	13.1	19.2	4.0	5.9	1.2	-	-
^{234}Th	-	-	-	-	1570.0	-	1440.0	-	-
^{230}Th	-	-	-	-	-	-	-	-	300
^{226}Ra	-	-	-	-	-	-	-	-	200
^{210}Po	-	-	-	-	-	-	-	-	200
^{210}Pb	-	-	-	-	-	-	-	-	80

- * Chemical toxicity is limiting. In all other cases radiotoxicity is the limiting factor.

Table 7. Derived Exposure Limits for Radon & Progeny

Parameter	Derived Level
Radon (EER)	1000 Bq.m ⁻³
Radon Progeny	0.3 WL

Table 8. Derived Surface & Personal Contamination Levels (Alpha)

Place of Contamination	DCL/DCL for uranium (Bq.cm^{-2})		
	Type S	Type M	Type F
General Working Areas (floors walls & in situ equipment etc.)	0.3	1.0	1.0
Personnel (Hands, skin)	3.0	3.0	3.0
Clothing (Work Clothes)	6.5	6.5	6.5
Clothing (Personal Clothes)	2.0	2.0	2.0
Footwear (Work Shoes)	8.0	8.0	8.0
Footwear (Personal footwear)	3.0	3.0	3.0
Equipment (in isolation)	1.5	1.5	1.5

DCL and DCL figures for beta activity may be taken to be same as that for alpha activity since they are in equilibrium.

Table 9. Derived Exposure Levels (External & Internal)

Parameter	Derived Level		
	γ	β	U in urine
External Radiation	1.0 mR.h^{-1}	25.0 mR.h^{-1}	
Internal			50 ug.l^{-1}

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Appendix I

Specific activity (Bq.mg⁻¹) of uranium
of different isotopic composition

²³⁵ U		²³⁸ U		²³⁴ U		Total Activity (Bq.mg ⁻¹)	Remarks
%	Sp. Act. (Bq.mg ⁻¹)	%	Sp. Act (Bq.mg ⁻¹)	%	Sp. Act (Bq.mg ⁻¹)		
0.40	0.317	99.597	12.273	0.003	5.223	17.813	Depleted U
0.72	0.571	99.275	12.233	0.005	12.233	25.037	Nat. U.
1.00	0.793	98.992	12.199	0.007	17.737	30.729	Enriched U
2.00	1.585	98.980	12.074	0.019	44.724	58.383	-do-
3.00	2.378	96.966	11.949	0.033	76.825	91.152	-do-
4.00	3.171	95.950	11.824	0.049	112.773	127.768	-do-
5.00	3.964	94.933	11.698	0.066	151.885	167.547	-do-
6.00	4.756	93.915	11.573	0.084	193.716	210.045	-do-

Note:

1. The specific activities listed above refer to the activities of the specified isotope in 1 mg of uranium as a whole and not 1 mg of the individual isotope.
2. The specific activities of enriched uranium are theoretically calculated for enrichment using gas centrifuges. The figures may be different for enrichment done using other processes.

Appendix II

Basic Limits (IC91)

Organ	Annual Dose Limit (mSv)	
	Occupational	Public
Whole Body	20	1
Skin, hands & feet	500	50

Appendix III

Classification of Uranium Compounds

Uranium Compound	Type	
	Inhalation	Ingestion
UO ₂	S	M
U ₃ O ₈	S	M
UO ₃	M	M
UF ₄	M	M
UCl ₄	M	M
MgU ₂ O ₇ (MDU)	M	M
(NH ₄) ₂ U ₂ O ₇ (ADU)	M	M
UF ₆	F	F
UO ₂ F ₂	F	F
UO ₂ (NO ₃) ₂	F	F

Appendix IV

Annual Limits of Intake (IC94)

Radio-nuclide	Type of Radiation	Annual Limit of Intake (Bq.y ⁻¹)				
		By Inhalation			By Ingestion	
		Type F	Type M	Type S	Type F	Type M
²¹⁰ Pb	β + γ	1.820x 10 ⁴	-	-	2.940x 10 ⁴	-
²³⁴ Th	β + γ	-	3.770x 10 ⁶	3.450x 10 ⁶	5.880x 10 ⁶	5.880x 10 ⁶
²³⁴ Pa	β + γ	-	3.640x 10 ⁷	3.450x 10 ⁷	-	1.960x 10 ⁶
²³¹ Pa	α	-	2.250x 10 ²	1.175x 10 ³	-	2.815x 10 ⁴
²¹⁰ Po	α	2.817x 10 ⁴	9.090x 10 ³	-	8.330x 10 ⁴	-
²²⁶ Ra	α	-	1.667x 10 ³	-	-	7.140x 10 ⁴
²³⁰ Th	α	-	7.140x 10 ²	2.777x 10 ³	2.300x 10 ⁵	9.520x 10 ⁴
²³⁴ U	α	3.125x 10 ⁴	9.520x 10 ³	2.940x 10 ³	4.080x 10 ⁵	2.410x 10 ⁶
²³⁵ U	α	3.333x 10 ⁴	1.111x 10 ⁴	3.280x 10 ³	4.350x 10 ⁵	2.410x 10 ⁶
²³⁸ U	α	3.448x 10 ⁴	1.250x 10 ⁴	3.500x 10 ³	4.550x 10 ⁵	2.630x 10 ⁶

1. The figures in above table refer to occupational exposure. Since the annual dose limit to members of public is 1/20th of the occupational dose limits, the ALI values for members of Public shall be 1/20th of the values shown in the above table.
2. The Daily Limit of Intake values are obtained by dividing the above values by 250 (no. of working days per year).

Appendix V

Typical Calculations

(1) Derived Air Concentration - Airborne uranium (Ref. Sec. 3.4.1)

Type S compounds :

Let the Annual Limit of Intake of composite uranium type 'S' by inhalation (mg.y^{-1}) be 'Ms'. The other terms used in equation [1] are:

$$S_1 (^{238}\text{U}) = 12.233, S_2 (^{234}\text{U}) = 12.233 \quad \& \quad S_3 (^{235}\text{U}) = 0.571$$

$$A_1 (^{238}\text{U}) = 3500, A_2 (^{234}\text{U}) = 2940 \quad \& \quad A_3 (^{235}\text{U}) = 3280$$

Substitute in equation [1],

$$\frac{12.233 M_s}{3500} + \frac{12.233 M_s}{2940} + \frac{0.571 M_s}{3280} = 1$$

$$\text{or } M_s = 127.7 \text{ mg.y}^{-1} \quad [\text{say } 125 \text{ mg.y}^{-1} \quad (3125 \text{ Bq.y}^{-1})]$$

$$D_s = 125 / (2000 \times 1.2) = 52.1 \text{ } \mu\text{g.m}^{-3} = 1.3 \text{ Bq.m}^{-3}$$

The DAC's in respect of type 'M' & 'F' uranium (D_M & D_F) are similarly worked out and are found to be 4.5 Bq.m^{-3} & 13.6 Bq.m^{-3} respectively.

(2) Derived Water Concentration (Ref. Sec. 4.3.1)

Type M compounds:

Let the Annual Limit of Intake of composite uranium type 'M' by ingestion (mg.y^{-1}) be 'Ws'. The other terms used in equation [8] are:

$$S_1 (^{238}\text{U}) = 12.233, S_2 (^{234}\text{U}) = 12.233 \quad \& \quad S_3 (^{235}\text{U}) = 0.571$$

$$A_1 (^{238}\text{U}) = 2.63 \times 10^6, A_2 (^{234}\text{U}) = 2.41 \times 10^6 \quad \& \quad A_3 (^{235}\text{U}) = 2.41 \times 10^6$$

Using equation [8],

$$20 \left[\frac{12.233 W_M}{2.63 \times 10^6} + \frac{12.233 W_M}{2.41 \times 10^6} + \frac{0.571 W_M}{.41 \times 10^6} \right] = 1$$

$$\text{or } W_M = 5018 \text{ mg.y}^{-1} \quad (1.25 \times 10^5 \text{ Bq.y}^{-1})$$

and the DWC, $DWC_M = 5018 / (1.48) = 3400 \text{ mg.m}^{-3}$

(3) Daily Limit of Intake (DLI) - (Table 3):

Assuming chronic uniform intake the Daily Limit of Intake (DLI) for occupational workers would be $(5018 \times 20/250) 401.44 \text{ mg.d}^{-1}$ or 10036 Bq.d^{-1} or say 10040 Bq.d^{-1} .

The DWC and DLI in respect of type 'F' uranium (DWC_F) are similarly worked out and are found to be 600 mg.m⁻³ & 1720 Bq.d⁻¹ respectively.

(4) DAC - Airborne uranium ore dust (Ref. Sec. 3.6.4)

Specific Activity of Ore Dust:

As a typical example, the Jaduguda uranium ore may be taken for calculating the DAC for airborne ore dust. Before computing the DAC for uranium ore dust, the bulk specific activity of air borne ore dust needs to be computed. The average ore grade is 0.052% U₃O₈ equivalent. Thus R' is 0.052 or the percentage P of uranium in the ore, is 0.044%. The specific activity of the air borne ore dust, which is very fine, is known to be a function of the particle size in an inverse proportion. Thus it is likely to exhibit a much higher specific activity than the bulk ore. Experiments have shown that the radium activity ' R_a , (Bq.g⁻¹)' in the uranium mill tailings (which is nothing but pulverised uranium ore) varies with the particle size according to the equation $R_a = 102 S^{-0.62}$ where S is the particle size (μm). While specific activity of radium in the bulk of the ore would be 5.4 Bq.g⁻¹ the value in 5 μm ore particles works out to 37.6 Bq.g⁻¹ using this equation. This implies a 7-fold increase in the specific activity. In other words the air borne ore dust would be equivalent to an ore of grade 0.308% instead of 0.044%. The bulk specific activity of the air borne ore dust is computed using equation [7]. The specific activities of the two parent uranium isotopes ²³⁸U and ²³⁵U are $S_1=12.233$ Bq & $S_2=0.571$ Bq per milligram of uranium respectively (from Appendix IV). The uranium ore in Jaduguda is known to be in secular equilibrium. Hence all the equilibrium fractions $F_{j,i}$ are equal to unity.

Using equation [7] the bulk specific activity ' S_o ' of the air borne ore dust is given by

$$S_o = 0.308 \left[\frac{12.233 \times 5}{100} + \frac{0.571 \times 2}{100} \right] = 0.192 \text{ Bq.mg}^{-1} \text{ or say } 0.190 \text{ Bq.mg}^{-1}.$$

DAC:

Let ' M_0 ' be the ALI by inhalation for the uranium ore dust (mg.y^{-1}). ' M_0 ' is computed using equation [5]. The other terms in the equation are

$$F_{11} = F_{12} = F_{13} = F_{14} = F_{15} = F_{21} = F_{22} = 1$$

$$S_1 (^{238}\text{U}) = 12.233, S_2 (^{235}\text{U}) = 0.571$$

$$A_{11} (^{238}\text{U type S}) = 3.50 \times 10^3, A_{12} (^{234}\text{U type S}) = 2.94 \times 10^3$$

$$A_{13} (^{230}\text{Th type M}) = 7.14 \times 10^2, A_{14} (^{226}\text{Ra type M}) = 1.667 \times 10^3$$

$$A_{15} (^{210}\text{Po type M}) = 9.09 \times 10^3,$$

$$A_{21} (^{235}\text{U type S}) = 3.28 \times 10^3, A_{22} (^{231}\text{Pa type M}) = 2.25 \times 10^2$$

$$P = 0.308$$

Substituting in Equation [5] and simplifying

$$M_0 = 6723 \text{ mg.y}^{-1} (1280 \text{ Bq.y}^{-1}) \text{ and}$$

$$\text{DAC } D_s = 6723 / (2000 \times 1.2) = 2.8 \text{ mg.m}^{-3} = 0.53 \text{ Bq.m}^{-3}.$$

Modifying Factors:

In addition to radioactivity from uranium and its decay products, the ore dust also contains other toxic substances notably free silica. From considerations of the contained free silica in the ore dust, the TLV for respirable ore dust for Jaduguda works out to 0.8 mg.m^{-3} [TLV = $30 / [\% \text{ free silica} + 3]$] [AC98]. Thus TLV based on considerations of exposure to free silica is more restrictive. Choosing the more restrictive of the two options the DAC for ore dust may be taken as 0.8 mg.m^{-3} (= 0.15 Bq.m^{-3}).

Experiment on water absorption by Beads

The beads were divided in 5 batches. 4 batches having 4 each and fifth one having remaining 2 beads. Beads in each batch were firstly dried at 110 degree C for 24 hours and weighed. Thereafter these were soaked in distill water for 7 days and removed these were firstly air dried at room temperature, weighed and then dried in oven at 110 degree C in oven for 24 hours and weighed. This was repeated upto 21 days and weights are given in Table below.

BATCH	INITIAL WEIGHT		WT AFTER 7 DAYS.		WT AFTER 15 DAYS.		WT AFTER 21 DAYS.	
	Air Dried	OVEN DRIED	OVEN DRIED	OVEN DRIED	OVEN DRIED	OVEN DRIED	OVEN DRIED	OVEN DRIED
1	1.1423	1.1376	1.1366	1.1392	1.1409			
2	1.1511	1.1466	1.1455	1.1478	1.1489			
3	1.2225	1.2189	1.2184	1.2179	1.2214			
4	1.1564	1.1511	1.1529	1.1537	1.1559			
5	0.5528	0.5511	0.5510	0.5520	0.5524			
TOTAL WEIGHT (gm)		5.2053	5.2044	5.2106	5.2195			
% Gain of initial wt.			-0.0173	0.1191	0.1710			

It will be observed that after first 7 days soaking in water the weight decreased by 0.017% but there after the weight increased by 0.119 % and 0.171 % of initial oven dried weight after 15 and 21 days soaking in water. The initial loss might have been due to loss of dust on putting in water.

05-Jul-99

(R.P.Gurg)

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