TS

NOT MEASUREMENT SENSITIVE

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DOE STANDARD

INTERNAL DOSIMETRY

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For ^{220}Rn, F_{Tn} is defined as

$$F_{Tn} = \frac{0.913\, C_{Pb-212} + 0.087\, C_{Bi-212}}{C_{Rn-220}} \quad (32)$$

where

$C_{Pb\text{-}212}$ = the concentration of ^{212}Pb;
$C_{Bi\text{-}212}$ = the concentration of ^{212}Bi
C_{Rn220} = the concentration of ^{220}Rn (thoron)

To assess radon progeny exposure from a time-integrated measurement using a nuclear track detector, one must understand the measurement itself[4]. The fundamental result of a measurement with a nuclear track detector is an observed number of tracks per unit area. Nuclear track detectors typically have an area of 10 to 20 mm^2. The number of tracks per mm^2 is empirically related to a number of radioactive transitions (of radon) per unit volume of air that occurred during exposure, that is, a time integrated radon concentration. One commonly reported unit is picocurie-days per liter (pCi-d/L), where

$$\begin{aligned} 1\,\frac{\text{pCi}\cdot\text{d}}{\text{L}} &= \frac{37\,\text{Bq/m}^3}{\text{pCi/L}}\cdot\frac{86{,}400\,\text{s}}{\text{d}}\cdot\frac{1\,\text{transition}}{\text{Bq}\cdot\text{s}} \\ &= 3{,}196{,}800 \text{ radioactive transitions per cubic meter} \\ &= 3.1968 \text{ transitions per cubic millimeter,} \end{aligned} \quad (33)$$

where the numerical conversion factors are given to five significant figures to prevent round-off error. The average concentration and average equilibrium equivalent concentration, $\overline{C}$ and $EE\,\overline{C}$, during the exposure, uncorrected for background, can be calculated by knowing the exposure time, t_E (d), the number of transitions per unit volume, N_V, and the equilibrium factor using

$$\begin{aligned} \overline{C}\,(\text{pCi/L}) &= \frac{N_v\ (\text{pCi}\cdot\text{d/L})}{t_E\ (\text{d})} \quad \text{and} \\ EE\overline{C}\,(\text{pCi/L}) = F\cdot\overline{C} &= \frac{F\cdot N_v\ (\text{pCi}\cdot\text{d/L})}{t_E\ (\text{d})}. \end{aligned} \quad (34)$$

However, *PAEE* is directly proportional to N_V without the need for the intermediate step of calculating an average concentration:

[4]The commercial nuclear track detectors for radon are insensitive to thoron.

$$
\begin{aligned}
E(\text{WLM}) &= F \cdot N_v\,(\text{pCi}\cdot\text{d/L}) \cdot \left(\frac{1\ \text{WL}}{100\ \text{pCi/L}}\right)\left(\frac{24\ \text{h}}{\text{d}}\right)\left(\frac{1\ \text{Months}}{170}\right) \\
&= 1.4118 \times 10^{-3}\ F \cdot N_v\,(\text{pCi}\cdot\text{d/L}); \quad \text{and} \\
E\,(\text{WLM}) &= 5.6471 \times 10^{-4}\ N_v\ (\text{pCi}\cdot\text{d/L}) \text{ if } F = 0.4.
\end{aligned}
\tag{35}
$$

Committed effective dose is assessed directly from *PAEE* using

$$
\begin{aligned}
H_{E,50}\,(\text{rems}) &= PAEE\ (\text{WLM}) \cdot 1.25\ (\text{rems/WLM}) \\
&= 1.7647 \times 10^{-3} \cdot F \cdot N_v\,(\text{pCi}\cdot\text{d/L}); \quad \text{or} \\
&= 7.0588 \times 10^{-4}\ N_v\,(\text{pCi}\cdot\text{d/L}) \text{ if } F = 0.4.
\end{aligned}
\tag{36}
$$

From 10 CFR 835 one can infer a dose conversion factor of 0.5 rems per WLM, using the following equation:

$$
\left(\frac{5\ \text{rems}}{2000\ \text{DAC} - \text{hours}}\right)\left(\frac{1\ \text{DAC}}{5/6\ \text{WL}}\right)\left(\frac{2000\ \text{hours}}{12\ \text{Months}}\right) = 0.5\ (\text{rems/WLM}), \tag{37}
$$

ignoring the minor inaccuracy that the WLM is based on a 170-h occupational month, not a 166.6-hour month (2000 h/y). Another item that does not correspond exactly is that Appendix A to 10 CFR 835 states that all *DAC*s are based on a 5 μm AMAD. This is not the case for the short-lived progeny of radon and thoron.

On the basis of more refined dosimetry and in an effort to make the WLM and the sievert consistent on a risk basis, in 1994 the ICRP and IAEA adopted a dose conversion convention 5 mSv/WLM (that is, 0.5 rem/WLM) (ICRP 1993a; IAEA 1996). Thus DOE's implied dose conversion factor identical to that recommended in the international guidance, meaning that for the same exposure, the DOE rule would impute a larger dose. The dosimetry system specified by 10 CFR 835 does not include published refinements based on knowledge of equilibrium factor, unattached fraction, and particle size (James et al. 1988; James 1994; National Research Council 1991; NEA 1985). Under many circumstances, the dose for a given exposure, calculated using these refinements, would decrease. However, measurements of aerosol size, unattached fraction, and equilibrium factor are difficult to do in the workplace, making the refinements impractical.

Example 7.2. Minimum Detectable Dose for a Nuclear Track Etch Radon Detector

One commercial supplier of nuclear track radon detectors suitable for personnel dosimetry reports that the minimum detectable amount for time-integrated radon concentration is 30 pCi-d/L (9.59E7 transitions/m^3). This leads to a minimum detectable E_{50} of

$$E_{50}\ (\text{rems}) = (7.0588\times10^{-4})\ (30\text{pCi}\cdot\text{d/L})$$

$$= 0.021\ \text{rems if}\ F = 0.4.$$

This value of 21 mrems is for *each* monitoring interval. If detectors are changed 12 times per year, the minimum detectable dose is 252 mrem.

7.5.5 Calculating Dose to Lung, and Intakes and Identities of Radon, Thoron and Their Progeny

The lung is the only tissue significantly irradiated by radon and thoron progeny. Since workplace air measurements yield E_{50}, one must calculate $H_{50,\text{lung}}$ from that portion of the committed effective dose due to radon or thoron progeny using

$$H_{50,\text{lung}} = \frac{E_{50}\ (\text{due to Rn or Tn})}{0.12}, \tag{38}$$

where 0.12 is w_T for lung in 10 CFR 835[5]. While this is the opposite of the usual practice of calculating committed effective dose from the sum of committed equivalent dose to tissues multiplied by the weighting factor for those tissues, it is necessary because air concentration measurements lead to E_{50}, not to $H_{50,\text{lung}}$.

The 1993 ICRP-65 does not list an annual limit on intake for radon or thoron progeny. These values are more correctly termed Annual Limits on Exposure. The concept of intake for radon and thoron progeny, as explained in ICRP Publication 32 (ICRP 1981b), is expressed not in activity units (e.g., μCi or Bq), but in potential alpha energy units (MeV or joules, J). *Intake, I*, of radon or thoron progeny by a worker breathing at Reference Man's rate of 1.2 m^3 h^{-1} is given by

$$\begin{aligned} I(\text{J}) &= PAEE\ (\text{WLM})\cdot(1.30\times10^5\ \text{MeV L}^{-1}\ \text{WLM}^{-1})\cdot(1.6022\times10^{-13}\ \text{J MeV}^{-1}) \\ &\quad\cdot(170\ \text{h Month}^{-1})\cdot(1.2\ \text{m}^3\ \text{h}^{-1}) \\ &= PAEE\ (\text{WLM})\cdot(3.5408\times10^{-3}\ \text{J h m}^{-3}\ \text{WLM}^{-1})\cdot(1.2\ \text{m}^3\ \text{h}^{-1}) \\ &= PAEE\ (\text{WLM})\cdot(4.2490\times10^{-3}\ \text{J WLM}^{-1}). \end{aligned} \tag{39}$$

In Equation 39, it is acceptable to substitute the individual worker's actual breathing rate if it has been measured and documented doing identical or similar work.

When intake of radon progeny or thoron progeny is specified in joules, the identity of the radionuclides should be specified as "radon progeny" or "thoron progeny." When intake of radon gas or thoron gas is reported, units of μCi should be used, and the intake, *I*, in units of μCi of ambient radon

[5]Other values of w_T have been used in other contexts, e.g., 0.08 in NCRP Report 91 and 0.06 for each of two regions in ICRP Publication 32.

(μCi) should be converted to equilibrium equivalent intake, *EEI*, using

$$EEI\ (\mu\text{Ci}) \quad = \quad I\ (\mu\text{Ci}) \cdot F \tag{40}$$

Numerical conversions for ^{222}Rn and ^{220}Rn quantities are given in Table IV.

Table IV. Summary of Numerical Conversions for Radon and Thoron Quantities, Regardless of the Precision of Measurements

Multiply	In Units Of	By	To Obtain	In Units Of
Concentration, *C*	pCi/L	10^{-9}	Concentration, *C*	μCi/mL
Ambient ^{222}Rn or ^{220}Rn concentration, *C*	pCi/L	*F**	Equilibrium equivalent ^{222}Rn or ^{220}Rn concentration, *EEC*	pCi/L
^{222}Rn *EEC*	pCi/L	1/100 = 0.01	Potential alpha energy concentration, *PAEC*	WL
^{220}Rn *EEC*	pCi/L	1/(7.43) = 0.13459	*PAEC*	WL
^{222}Rn or ^{220}Rn progeny *PAEC*	WL	Exposure time, *t* (hours) ÷170	Potential alpha energy exposure, *PAEE*	WLM
Integrated ^{222}Rn concentration, N_V (ambient)	pCi·d/L	*F*× 1.4118E-3	*PAEE*	WLM
Integrated ^{222}Rn concentration, N_V (ambient)	pCi·d/L	5.6471E-4 assuming *F*= 0.4	*PAEE*	WLM
^{222}Rn *PAEE*	WLM	0.5	E_{50}	rem
^{222}Rn *PAEE*	WLM	2000/4 = 500	*Exposure*	*DAC*·h
^{220}Rn *PAEE*	WLM	$1/6 = 0.1666\overline{6}$	E_{50}	rem
^{220}Rn *PAEE*	WLM	$2000/12 = 166.6\overline{6}$	*Exposure*	*DAC*·h
E_{50} for ^{222}Rn or ^{220}Rn	rems	$1/0.12 = 8.333\overline{3}$	$H_{lung,50}$	rem
PAEC	WLM	4.2490E-3	Potential alpha energy intake, *I*, of ^{222}Rn or ^{220}Rn progeny	J
*For ^{222}Rn, $F_{default}$= 0.4; for ^{220}Rn, $F_{default}$= 0.04				

7.5.6 Possible Values of *DAC*s for Pure Radon and Thoron Gas

Neither the IAEA nor the EPA, NRC, or DOE have set standards for inhalation of pure radon or thoron such as may be found inside an air-purifying respirator. However, the ICRP in its 1981 Publication 32 did set such standards based on limitation of stochastic risk and on dosimetry. The 1981 ICRP *DAC* for ^{222}Rn without progeny is 1.5E5 Bq·m^{-3}, while that for ^{220}Rn + ^{216}Po (which are essentially in equilibrium due to the 0.145-s half-life of ^{216}Po) is 2.5E5 Bq·m^{-3}. These values are based in the same inferential system as the *ALIs* of 0.02 J and 0.06 J, respectively, for radon and thoron progeny. Since that system deduces values of 4.8 WLM and 14.4 WLM as ALEs for radon and thoron progeny, the concentrations shall be scaled by the ratio of 5/6 (= 4/4.8 = 12/14.4) to arrive at concentrations suitable for comparison to the DOE system. Furthermore, these *DAC*s are described as being exactly 100 and 500 times, respectively, larger than the equilibrium equivalent *DAC*s for radon and thoron. Thus, the *DAC*s in the DOE system become 8,333 pCi/L for pure ^{222}Rn and 9,325 pCi/L for pure ^{220}Rn (with ^{216}Po).

The 1993 UNSCEAR Report (Annex A, Table 24) has "effective dose" coefficients for radon and thoron gas (pure), both indoors and outdoors, in nSv per Bq·h·m^{-3}. These are given in Table V. The stochastic derived air concentration corresponds to 2.5 mrems per hour (i.e., 25 μSv.h^{-1} or 25,000 nSv·h^{-1}), so a "5-rems per year" *DAC* for pure radon or thoron gas can be calculated by dividing 25,000 nSv·h^{-1} by the effective dose coefficient. Note that these values, about 3,975 pCi/L and 6,143 pCi/L for radon and thoron, are comparable to the values derived above from ICRP Publication 32, even though the approaches are dramatically different and even the dose quantities are different.

Table V. Effective Dose Coefficients for Radon and Thoron Gas (Pure), Both Indoors and Outdoors

		Effective Dose Coefficient nSv per Bq h m^{-3}		*DAC*(Bq/m^3)	*DAC* (pCi/L)	*DAC* (μCi/cm^3)
		Gas	*EEC*	Gas	Gas	Gas
Radon	Outdoors	0.17	9	147059	3975	3.97E-06
	Indoors	0.17	9	147059	3975	3.97E-06
Thoron	Outdoors	0.11	10	227273	6143	6.14E-06
	Indoors	0.11	32	227273	6143	6.14E-06

7.5.7 Choice of and Use of Assigned Protection Factors for Respirators in Radon and Thoron Dose Calculations

Equilibrium factors inside respirators have not been measured. Clearly, for HEPA-filtered air-purifying respirators, the equilibrium factor would be close to zero, since virtually no particles pass through a respirator. However, radon and thoron are noble gases and will pass unimpeded through a particulate air filter in an air-purifying respirator. The use of activated carbon filters may impede the passage of 56-s thoron considerably, perhaps permitting some of it to decay. The use of activated carbon filters for radon is unlikely to be effective for prolonged exposures, since it will merely retard the passage of the radon. Using the rule-of-thumb observation that "one gram of carbon acts like 4 liters of air," a 50-g charcoal canister will act as if it were 200 liters of air, or about 10 minutes' worth of intake by a worker. Adsorbed radon will begin to desorb after a while and eventually radon will desorb as fast as it absorbs. Until there are measurements, it is not acceptable to use an assigned protection factor (*APF*) for radon gas or thoron gas greater than 1.

Radon and thoron gas concentrations may limit the *APF* for an air-purifying respirator.

Three options are available for determining *APF*s for radon, thoron, radon progeny, and thoron progeny, as summarized in Table VI. The first, best, and simplest option, is to accept the ANSI Z88.2-1992 *APF*s for radon progeny and thoron progeny, and to accept *APF*s of 1 for radon gas and thoron gas.

In the second option, regardless of the actual filtering ability of a respirator, an *APF* for radon and thoron progeny in combination with radon and thoron gas is the lesser of either the ANSI Z88.2-1992 (ANSI 1992) value or

$$APF \le 100 \cdot F_{\mathrm{Rn}} \ for\ ^{222}Rn, and$$
$$APF \le 500 \cdot F_{\mathrm{Tn}} \ for\ ^{220}Rn, \tag{41}$$

with the proviso that the *APF* cannot be less than 1.

Using the default values of 0.4 and 0.04 as examples, *APF*s can be no more than 40 for radon taken together with its progeny, or 20 for thoron taken together with its progeny, regardless of the respirator's performance for radon or thoron progeny.

The third option is to follow a recommendation by the National Institute of Occupational Safety and Health (NIOSH). NIOSH has recommended that an *APF* of no more than 10 be allowed for respirator use in underground mines due to the observation that workers do not use respirators more than 90% of the time (NIOSH 1987). Similarly, the ICRP has recommended a protection factor of no more than 10 in paragraphs 69 and 71 (ICRP 1986a), for practical reasons.

Table VI. Three Options for Assigned Protection Factors for Rn, Tn, and Their Progeny

Option	Radon	Radon Progeny	Thoron	Thoron Progeny
Measure Gas and Progeny	1	ANSI Z88.2-1992	1	ANSI Z88.2-1992
Eq. Factor, Gas Measurement	$1 \le APF \le 100 \cdot F_{Rn}$		$1 \le APF \le 500 \cdot F_{Rn}$	
NIOSH/ICRP	1	10	1	10

For airline supplied-air respirators, it is important to ensure that the intake air is filtered of radon progeny and free of radon gas. Bottled-air respirators in which the air has been aged for 30 or more days may be assumed to be free of radon and radon progeny.

7.5.8 Determination of Radon and Thoron Background

The background concentration used should be the best available estimate of the average concentration that would have existed without the activity or source. For distributed sources of radon, it is suggested that background be determined in accordance with DOE/EH-01737, *Environmental Regulatory Guide for Radiological Effluent Monitoring and Environmental Surveillance* (DOE 1991).

One method of determining background is through measurements made before the commencement of the activity or from measurements made in other unaffected parts of the same building (indoors) or from measurements made at least 400 m (≈1/4 mile) away from any known local source and/or up wind (outdoors). A site-specific background should be used whenever possible. However, if determination of the site-specific background is not feasible, a community-wide average may be used for up to one year until local measurements have been made. If neither of these is practicable, then background values of 0.006 WL for radon progeny and 0.002 WL for thoron progeny may be used indoors and 0.002 WL for radon progeny and 0.001 WL for thoron progeny may be used outdoors (see Table VII).

Table VII. Default Background *PAEC* Values

Location	^{222}Rn Progeny	^{220}Rn Progeny
Indoors	0.006 WL	0.002 WL
Outdoors	0.002 WL	0.001 WL

7.5.9 Correcting for Relatively High Background *PAEC*s

If the background radon progeny concentration is determined to be greater than 0.03 WL indoors or 0.01 WL outdoors, there is a significant probability that an unidentified source of radon exists. Therefore, if background is found to be greater than these concentrations, the cause of this elevated concentration shall be determined before using it as the background value in occupational radon progeny exposure calculations. If a previously unidentified radon source is discovered, then a background value shall be redetermined that is independent of any contribution from this source.

7.6 SIMPLIFIED METHOD FOR DOSE ASSESSMENT FOR SMALL INTAKES

When intakes can be established on the basis of bioassay data and are small (i.e., leading to doses below administrative control levels, or leading to $E_{50} < 100$ mrem), it is permissible to assign E_{50} values using Eq. (21), which amounts to using default assumptions. When doses approach limiting values for workers, it is often appropriate to refine dose assessments by using individual-specific parameters rather than default assumptions. The level of effort expended in dose assessment is generally in proportion to the projected dose.

7.7 UNCERTAINTIES

While internal dose assessments may be among the most accurate dosimetry available (e.g., following an intake of tritiated water or ^{137}Cs that occurs at a known time), in many cases uncertainties are very large (e.g., following a small intake of plutonium in an unknown chemical form at an uncertain time). Unlike external dose assessments, internal dose assessments change in many cases as information accrues over time. The availability of additional data may result in a reduction of uncertainty or a change in a point estimate of dose, or both.

Assessing doses starting from air activity concentrations and times requires more assumptions than does assessing doses from excreta measurements or in vivo count data. Thus, uncertainties are significantly larger for this method than they are from bioassay or in vivo counts. A summary of uncertainties and their relative impact on assessment of internal doses from in vivo and in vitro bioassay, and from air monitoring is given in Table VIII.

Assessing committed effective dose (E_{50}) from bioassay measurements is generally *more accurate* than assessing E_{50} from measurements of concentration of radioactive material in air and multiplying by stay time and breathing rate. There are numerous reasons why the latter procedure requires more leaps of inference than the former. However, for the case of plutonium and other actinides, air samples and stay times may be much more sensitive, that is, they may have much lower detection limits when expressed in terms of E_{50}. Furthermore, dose assessment based on air samples may also be *more precise*, even if far less accurate. Finally, for short-lived radionuclides (e.g., the decay products of radon), there may not be any bioassay procedure; the only available methods involve air monitoring.

Table VIII. Relative Importance of Various Sources of Uncertainty for Dose Assessment

Source of Uncertainty	In vivo	In vitro	Workplace Monitoring
The degree to which the contaminated air measurement represents the air actually breathed, including the effects of respiratory protection	-	-	high
The difference between actual and modeled breathing rate	-	-	high
Nose or mouth breathing	-	-	high
Degree of knowledge of particle size distribution	med	high	high
Aerosol transportability from lung into the transfer compartment, GI tract, and lymphatic system	med	high	high
Assumed aerosol deposition in the lung	-	high	high
Clearance rate from the lung	high	high	high
Cleared aerosol absorption from the GI tract and lymphatic system	high	high	high
Time course of intake(s)	high	high	high
Assumptions of present locations of radionuclides within the region near the detector (e.g., lymphatic system or lung)	high	-	-
Systematic uncertainty in calibration	high	low	med
Random uncertainty in measurement	high	low	med
Systematic uncertainty in the choice of an appropriate blank	med	low	low
Biokinetic model assumptions	high	high	high
Future time course of retention and excretion	high	high	high
Mass of target tissues or organs	high	high	high
Assumptions of present locations of radionuclides within the body (e.g., liver or bone)	low	high	High
Fraction of radionuclide excreted by route being sampled	-	high	-

Precision refers to how reproducible a measurement is. Bias or accuracy refers to how close the average of measurements is to a "conventionally true value." Precision and bias are independent, that is, measurements may be biased or unbiased without regard to their precision, and they may be precise or imprecise without regard to their bias.

Sensitivity, as used here, refers to the lowest E_{50} that can be distinguished from background. Technology shortfall, as defined in the Internal Dosimetry chapter of the *10 CFR 835 Implementation Guide* (DOE 2008b), occurs when the sensitivity of a dose assessment method is not adequate to meet the dose assessment requirements of 10 CFR 835. See (Carbaugh 2003c) for using the sensitivity of a bioassay measurement to assess the minimum detectable dose under a particular set of intake circumstances.

The best accuracy and precision for E_{50} assessment in the DOE is that for intakes of tritium when assessments are based on urinalysis bioassay results. Doses can be assessed to within 10% to 20% after only a couple of measurements over a couple of days. Even a site with a detection limit of 0.01 uCi of ^{3}H per liter of urine (10,000 pCi/L) can detect 0.04 mrems immediately after a tritium intake, and 22 mrems 90 days after a tritium intake. With an average tritium sampling frequency of every 14 days, one can detect a committed effective dose of 0.1 mrem, or about 1000 times less than the level at which a bioassay program is required by 10 CFR 835. Two cases are shown in Table IX, for effective clearance halftimes of 10 days (Reference Man) and 7 days (typical of a summer day). Dose numbers are higher for effective clearance half-times shorter than 10 days. Thus, for tritium, accuracy, precision, and sensitivity are no problem.

Table IX. Comparisons of Committed Effective Dose Detection Limits for Tritium Bioassay When 0.01 μCi/L of ^{3}H Is Observed, as a Function of Time since Intake

	E_{50} Inferred from 0.01 μCi/L of ^{3}H in urine (mrem)	
Days Since Intake	T_{eff} = 10 days	T_{eff} = 7 days
1	0.04	0.03
14	0.11	0.47
90	22	220

In the DOE, the worst accuracy for E_{50} assessments occurs for plutonium and actinides based on air monitoring data and worker's stay time. Such measurements, however, may result in assessed doses that are both more precise and far more sensitive than doses assessed on the basis of bioassay measurements. In the case of plutonium, there is a technology shortfall for doses assessed on the basis of routine urinalysis bioassay; such programs have such poor sensitivity that they may miss doses of several rems (thousands of millirems). Continuous air monitors for plutonium can readily detect 10 to 30 *DAC*-h under field conditions, corresponding to E_{50} values of 25 to 75 mrem. Lapel air samplers, for which air filters are measured in the laboratory, can do somewhat better.

Short-lived decay products of ^{222}Rn are found where there are radium-bearing residues of uranium ores. There is no practical method of bioassay for such decay products, so the only alternative is to use air monitoring results.

The results of the comparison of these three cases are shown in Table X.

Table X. Comparison of Methods of Assessing Dose from Intakes of Radionuclides

Method	Type	Accuracy	Precision	Sensitivity	Cost
^{3}H urinalysis	Bioassay	High	High	High	Low
^{239}Pu urinalysis	Bioassay	Moderate	Low	Very low	High
^{239}Pu air monitoring	Air monitoring	Very low	Moderate	Moderate	Moderate
Radon progeny air monitoring	Air monitoring	Moderate	Moderate	Moderate	Moderate

7.7.1 Uncertainties Associated with Preliminary Evaluations

Preliminary dose evaluations, when based on bioassay data obtained within the first few days of an intake by inhalation, may be very uncertain. It is not uncommon for such preliminary evaluations to be wrong by a factor of 10 either direction. It is thus very important not to overreact to initial dose assessments, which may be revised either upward or downward when bioassay data over a period of weeks or months become available.

7.7.2 Uncertainties Associated with Final Evaluations

Even when all bioassay data are consistent with a plausible biokinetic model, in many cases there are still significant uncertainties in doses assessed from bioassay data. This is especially true of intakes of actinides and doses from intakes of unknown time course and unknown physical and chemical form. For significant intakes, it is desirable, although not always feasible, to quantify and document the uncertainty associated with a final dose assessment.

8 INTERNAL DOSE MANAGEMENT

10 CFR 835 requires internal dose evaluation programs for assessing intakes of radionuclides and for maintaining adequate worker exposure records. The effective assessment of dose from intakes is highly dependent on individuals (staff, management, radiation protection, medical, etc.) taking appropriate action. 10 CFR 835 explicitly requires adding equivalent dose due to external irradiation to committed effective dose due to irradiation by internal sources. Optimization principles should be applied to maintain internal and external doses ALARA (ICRP 1978b, 1989a). This necessitates a close working relationship and cooperation between staff, management, medical, and radiation protection personnel. Each site shall have a plan that documents the dose management practices.

8.1 ROUTINE RADIOLOGICAL WORKER DOSE MANAGEMENT

Radiological workers should be requested to sign a statement concerning any prior work at a facility where radioactive materials or radiation generating machines were used. The signed statement should be available to the internal dosimetry group prior to a worker's being potentially exposed to radioactive materials. The internal dosimetry group should determine the existence or potential existence of a prior intake that provides current or future dose (e.g., exposure to short-lived radionuclides during the current or past exposure year or exposure to long-lived radionuclides). Radiological workers who indicate the existence or potential existence of an intake during previous work shall be prevented from having additional intakes until their cumulative *TED*, current retained quantities and current radionuclide excretion rates (if any) have been established. This action should be accomplished either through receipt of sufficient data from a previous employer(s) or by baseline bioassay measurements. If demands for the worker's services are immediate and great, the worker's signed estimate of prior dose can be used until official records are received.

8.1.1 Management of Dose from Previous Intakes (Work Restrictions)

In operation of programs for monitoring and controlling worker doses, consideration should be given to the reduced effectiveness of bioassay monitoring for workers that have internally deposited radionuclides (occupationally or medically derived). Special monitoring programs should be implemented as necessary to ensure that protection of these workers can be provided.

8.1.2 Compliance with Internal Dose Monitoring Requirements

Management shall require that radiation workers:

- comply with facility contamination control requirements
- participate in required bioassay measurements
- inform the health physicists, other radiation protection personnel, or their immediate supervisor as soon as an intake is suspected

Management should adopt additional administrative controls such as work restrictions for workers who do not meet the above requirements.

8.1.3 Control of Dose to the Embryo/fetus, Minors, and Students

Administrative controls should be established to protect the embryo/fetus for declared pregnant workers. This is necessary because of uncertainties in:

- distribution and retention of radioactive materials in the embryo/fetus
- dosimetry to embryo/fetus
- associated risk.

Example 8.1 illustrates sample dose management practices for declared pregnant workers.

Example 8.1. Dose Management Practices Regarding Internal Dosimetry Associated with Embryo/Fetus Dose Control

If a female radiological worker is on a routine bioassay schedule and submits a declaration of pregnancy, the appropriate bioassay is obtained from the female radiological worker as soon after the declaration as possible. This bioassay serves two purposes:

1) If the declared pregnant worker will no longer be exposed to possible intakes during the remainder of the gestation period, then this becomes an ending assignment bioassay and is used to document the embryo/fetal internal dose (usually none) for the period from conception to declaration.

2) Even if the declared pregnant worker continues her present work assignment, this declaration bioassay is reviewed using the embryo/fetal derived reference level, and serves either to show that no internal dose has been incurred to date or to document what internal dose has been incurred for the period of conception to declaration. The worker and her supervisor should have a good understanding of what dose has been received during the gestation period up to the time of declaration in order to make decisions about her work assignments for the remainder of the gestation period. The information gained from the declaration bioassay gives everyone a more complete dose status at the time of declaration. Finally, if the declared pregnant worker continues work where intakes are possible, a new bioassay schedule may be necessary for the remainder of the gestation period. At the very least, an attempt is made to obtain a bioassay after the pregnancy is concluded or as soon as the declared pregnant worker ceases work involving exposure. The gestation period is treated as a time separate from the declared pregnant worker's normal bioassay monitoring period.

Enhanced control of intake to minors and students should be exercised since the effective dose limits for these individuals are the same as for the general public.

8.2 DOSE LIMITATION

One acceptable method of limiting doses to workers involves the concept of administrative control levels as described in the RadCon Standard (DOE 2008a). The establishment of such dose levels below the limits provides reasonable assurance that limits will not be exceeded.

8.2.1 Interface and Coordination with the External Dosimetry Program and the Radiological Control Organization

Since the DOE limits *TED*, a two-way communication system is needed between the internal and external dosimetry programs. The two programs should develop a mechanism whereby the internal dosimetry program receives, in a timely fashion, notification of external doses received by workers that are a significant fraction of the applicable limits. Similarly, the external dosimetry program should be informed, by the internal dosimetry program, of workers who have experienced significant intakes. Together, the two programs must coordinate with the radiological control organization to prevent such workers from exceeding administrative control levels and dose limits.

In addition, when planning radiological work, workers who may be likely to receive both external irradiation and intakes of radioactive material should be identified by the radiological control organization, and this information communicated to the internal and external programs so that checks can be made of the dose status of workers for whom not all dose information is in the central records system. For example, workers for whom an intake is suspected but not yet confirmed should be permitted to engage in additional radiological work with significant potential for doses only if there is no indication that additional work would put the worker in danger of exceeding an administrative control level.

8.2.2 Lifetime Dose Control

Lifetime dose control has been recommended by the NCRP and described in the RadCon Standard. However, lifetime dose control is not required by 10 CFR 835 in any explicit way, and, in any case, is suggested only for radiological workers by the RadCon Standard (DOE 2008a). Because of differing practices in the past, it is problematic to determine doses adequate for today's dose quantities from historical bioassay and workplace monitoring data. Methods developed for epidemiological studies, such as of Oak Ridge Associated Universities, may be of some help (Crawford-Brown et al. 1989). For additional guidance refer to Section 9.4 for discussion of dose re-evaluations.

8.2.3 Doses Due to Intakes Prior to January 1, 1989

Prior to January 1, 1989, regulations in the DOE did not require computation of E_{50} and $H_{T,50}$ values from bioassay and workplace monitoring data. From January 1, 1989, sites were required to assess and record these values. Prior to 1989, records of intakes, if they exist, were likely to be expressed in fractions of a maximum permissible body burden (MPBB). There is no simple and straightforward general method to convert MPBB values to E_{50} values. Sites should consider whether it is feasible and cost-effective to attempt to historically reassess doses prior to 1989. The DOE position on prior years' exposures records does not address doses due to intakes prior to 1989 or intakes at non-DOE facilities.

8.2.4 Uncertainties

It is current practice in the DOE to use point estimates of dose and to ignore ranges of uncertainties when comparing doses to limits and administrative control levels. However, sites may

consider uncertainties when invoking work restrictions based on professional judgment. For example, an E_{50} value with a multiplicative (lognormal) uncertainty characterized as 1.5 rems (× or ÷ by 2) has a roughly 5% chance of actually exceeding 6 rems. This may exceed the "comfort level" of those responsible for dose management. While comparing point estimates of doses with limits and administrative control levels, sites may still consider using an upper confidence limit (such as the 95% upper confidence limit on a dose) for invoking work restrictions or other dose control practices.

8.3 DOSE CONTROL FOLLOWING ACCIDENTAL INTAKES

Unlike external irradiation, whose course cannot be altered after exposure, doses from retained quantities of radioactive materials can be influenced after intake occurs in some cases. While intervention following intake is usually a medical matter, it is necessary to involve the internal dosimetry program. Methods of reducing dose following an intake include enhanced decorporation ranging from washing to debridement, excision, blocking, chelation, and forcing fluids.

8.3.1 Incident Dose Management

Significant intakes of radionuclides usually occur as the result of accidents, not from routine, planned operations. A prompt response is needed following indication that an unexpected intake has occurred. The time interval and degree of urgency associated with the follow-up actions depend on several factors, including the possible significance of the exposure and the elapsed time from its occurrence to its detection.

8.3.2 Preparation for Incidents Involving Intake

Management at a facility should be prepared for an incident involving a worker receiving an intake of radioactive material even though the probability of an incident may be very small. Management shall have an emergency action plan for response to a potential or unplanned intake of radioactive material and be prepared to follow it. The amount of detail in the plan should be commensurate with the possible severity of an accidental intake.

An emergency action plan to deal with accidental internal intakes shall include: 1) plans for activating key response functions, such as internal dosimetry, analytical laboratory, and medical support, 2) the readiness of facilities, 3) the training of personnel, and 4) predetermined specifications for bioassay and other measurements.

The elements of this plan shall include the following:

- decision levels for determining when monitoring data or accident events necessitate emergency medical response
- responsibilities of the affected worker, the health physicist, medical staff, and management or supervisory personnel
- guides for immediate medical care, decontamination, monitoring, and the longer-term follow-up response
- provisions for periodically reviewing, updating, and rehearsing the emergency action plan.

Since the elements of this plan may be documented in various operating manuals, the overall

program, including the interrelationships, shall be summarized in one document with appropriate direction to the location of the various elements (e.g., use of a response tree).

The site occupational medicine personnel shall prepare a summary of the therapeutic measures, by radionuclide, that are maintained for the site and the targeted time from intake to treatment. These plans should be reviewed and updated as necessary.

In general, medical treatment (e.g., DTPA [diethylenetriaminepentaacetic acid] therapy) shall be available to internally contaminated individuals within a few hours of the detection of the exposure (see Section 10).

8.3.3 Internal Dose Control After an Incident

Before a worker is allowed to return to radiation work following a potential intake, the worker's exposure status shall be evaluated. This evaluation should include consideration of the uncertainty associated with early assessments of internal dose, the dose received from external exposures during the year, and the committed effective dose for the year from all prior intakes. Temporary restrictions or limitations from radiation work should be considered if the work could interfere with the internal dose assessment (e.g., if additional intakes of the radionuclide of interest could occur). Additional guidance is provided in Section 10.

9 RECORDS AND REPORTS

Internal dosimetry records are an important part of an internal dosimetry program, not only to demonstrate compliance with 10 CFR 835 and the DOE Orders, but also to support the on-going dose management of individuals following intakes. The minimum requirements for an internal dosimetry records program are specified in 10 CFR 835.702 and 703, with additional guidance in the Articles 523 and Section 7 of the RadCon Standard (DOE 2008a), and in the *Internal Dosimetry* chapter of the *10 CFR 835 Implementation Guide* (DOE 2008b). Prior dose assessments not compatible with committed equivalent doses shall be converted to provide committed organ/tissue and effective doses. However, constraints discussed in Section 8.2.3 may limit some reassessments.

Requirements for the annual reports to employees are given in 10 CFR 835.801, with additional guidance in the RadCon Standard Article 781.1 and the IDG.

The ANSI N13.6 standard on "Practice for Occupational Radiation Exposure Records Systems" (HPS 2010a) provides guidance for the systematic generation and retention of records relating to occupational radiation exposure.

In addition, Title 36 Code of Federal Regulations, Chapter XII, specifies policies for Federal agencies' records management programs relating to records creation and maintenance, adequate documentation, and proper records disposition.

As with all records containing sensitive data — such as individuals who are identified in radiological records by name, identifying numbers (e.g., Social Security Number or payroll number), or symbol — the Privacy Act of 1974 (as amended) shall be applied. That is, no information regarding an individual should be revealed to anyone other than the identified individual or DOE/DOE contractor personnel who have a need to know without advanced written consent of the individual, unless authorized by the Privacy Act. Records of deceased individuals are not covered by the Privacy Act, but are subject to the Freedom of Information Act.

9.1 WHAT TO RECORD - A GENERAL PHILOSOPHY OF RECORDS

The 10 CFR 835 regulation specifies particular items for which recording is required, including specific doses, combinations of external and internal doses, and nuclides of intake and their magnitude. In addition, records are required of pertinent data and information which resulted in the generation of the dose and intake information. There is a substantial amount of professional judgment needed in deciding what data to record and how to record it. The development of relational databases has eased much of the data storage capability but in the process has created some possible pitfalls. The interpretive keys and professional judgments used in evaluating data may not readily lend themselves to database formats. For this reason, an internal dose evaluation report consisting of discussion of assumptions and conditions unique to the individual worker and intake is suggested as the most effective means of documenting the assessment. The report may include the actual data used and calculations or computer outputs, or may reference the appropriate supporting documents and databases where the information and results can be found. Generally, the final doses are entered into a dosimetry database where they can be electronically summed with appropriate external doses to give the needed combinations.

A guiding philosophy for documenting cases is to imagine that 20 years after an exposure was evaluated, a knowledgeable health physicist is asked to independently review and critique that evaluation. The information available in the evaluation shall be adequate to lead that health physicist to a complete and unambiguous understanding of the original evaluator's thought processes in arriving at the intake and dose assessments. The advance of internal dosimetry and bioassay science in the intervening years might

lead the reviewing health physicist to completely disagree with the conclusions. However, there should not be any misunderstanding as to the approach and logic of the original evaluation.

9.2 REPORTING PRELIMINARY ASSESSMENTS OF UNPLANNED EXPOSURES

When an unplanned exposure occurs, an investigation and reporting system is set in motion to determine the severity of the event. A key item of information being sought is the magnitude of any dose likely to result from the intake. Pressure is often placed on the bioassay and internal dosimetry program to make immediate and precise assessments for categorizing the event. Unfortunately, bioassay measurement results upon which these assessments can be based are usually slow in coming and highly variable. Where the measurements can be obtained rapidly, it is often at a cost of analytical sensitivity, which can raise the minimum dose detectable by bioassay.

The early clearance patterns in the first few days after intake are the most uncertain parts of the biokinetic models, being highly affected by particle size, mode of intake, material transportability, and individual person-specific metabolism (Traub and Robinson 1986). If an intake is quite minor, then these issues are not particularly significant. This is because a conservative interpretation of early data using the standard biokinetic models resulting in a small E_{50} (e.g., below 100 mrem) is not likely to cause any major impact on classification of event.

High-energy photon-emitting radionuclides (e.g., fission and activation products such as ^{137}Cs and ^{60}Co) are easily and quickly measured using whole body counter systems. Because incidents involving these nuclides are usually small relative to the *ALI*, reasonably good early assessments of intake and dose can be obtained with a high degree of confidence.

Such is not the case when dealing with plutonium and americium mixtures. These nuclides are among the most difficult for which to provide confident early assessments. Errors in knowledge of the mixture can lead to significant variations (factors of 2 to 10) in assessed doses. In vivo measurements are relatively insensitive for plutonium mixtures. Likewise, early urine samples analyzed by a relatively insensitive radiochemical procedure are not well-suited for dose assessments but may be very valuable for initial determination of need for or efficacy of any dose reduction therapy. Large-volume urine samples and fecal samples will provide better assessments of intake but will likely require several days to produce results. The Hanford Site has described several example responses to potential intakes in the *Hanford Internal Dosimetry Project Manual* (Carbaugh 2003b), which specifically identifies the capability of response as a function of time following intake and measurements made. Table XI is an example of such an approach.

Preliminary assessments must be considered just that: It is not appropriate to place heavy reliance on the actual magnitude of the dose in the first few days following a suspected intake. It would not be unusual for a preliminary assessment of 10 or 20 rems committed effective dose derived from initial bioassay data for a plutonium intake to ultimately be lowered to 1 rem committed effective dose based on long-term follow-up data.

9.3 PRECISION OF INTERNAL DOSE ASSESSMENTS

Interpreting bioassay data generally involves making many assumptions which can vary between dosimetrists. Intercomparisons have been performed between DOE sites (Hui et al. 1994) and internationally (Gibson et al. 1992). These comparisons have shown that ranges between 30% and 50% of the mean value are not uncommon. In practical terms, this means that a factor of 2 to 3 variation between dosimetrists is not unreasonable. Similar results were demonstrated by intercomparison of one particular case (La Bone et al. 1992; La Bone and Kim 1993). A reassessment based on long-term data

increased the dose by a factor of 4 and also showed a factor of 2 variability around the mean assessment of dosimetrists.

Knowledge about the relative precision (or imprecision) of internal dose assessments does not relieve the site from making a precise conclusion about the dose to be assigned. It shall be the responsibility of the internal dosimetrist to decide on the best assessment of internal dose to be assigned for any confirmed intake. Peer review by another qualified dosimetrist is recommended, and is particularly important for assigned doses which exceed administrative control levels or dose limits.

Table XI. Inhalation of Aged 6% Plutonium Mixture, No DTPA Given at Worksite

Days Since Intake	Measurements	When Results Are Known	What Can be Said at What Point	Problems or Comments
Same day	3000-s chest count; second voiding spot urine; emergency processing	Same day or first thing next morning	Can say if E_{50} is more or less than 12 rems	If anything is detected, should administer DTPA
1	12-h urine, emergency processing; second chest count if first result detected activity	End of second day	If nothing in urine or chest, then E_{50} is Type M < 5 rems, or Type S < 10 rems	If nothing in urine or chest, then DTPA is not needed. If Pu alpha in urine > 2 dpm, then consider initiating DTPA.
2	24-h total urine, expedite processing	Morning of fifth day	If nothing in sample (and previous chest counts), then E_{50} Type M < 500 mrem, Type S < 5 rems	From bioassay data, still won't know inhalation material type.
1-3	Total fecal excretion for first 3 days after intake[a] Two processings by lab: 1) LEPD[b] expedited processing; 2) IPA[c] priority processing	LEPD[b] results: 6-7 days after intake IPA[c] priority: 16-17 days after intake	If nothing in LEPD analysis, then E_{50} < 500 mrem If nothing in IPA, then E_{50} < 100 mrem	-

[a]If more than one sample is produced in a day, the samples should be composited into a single sample before analysis.
[b]LEPD: Code for lab analysis, referring to non-destructive low-energy photon spectrometry; measures x rays from ^{241}Am.
[c]IPA: isotopic plutonium and ^{241}Am via alpha spectrometry.

9.4 GUIDANCE ON LONG-TERM REEVALUATION OF INTAKES

The purpose of long-term reevaluations is to verify the accuracy of projected bioassay patterns and thereby verify the accuracy of assigned intakes and doses. Since by their very nature long-term reevaluations are performed at long times after intake, there is little merit in reopening the administrative investigation of an intake based on a reassigned dose, regardless of whether or not the reassignment changes the original standing with regard to administrative control levels or dose limits. By the time a

reevaluation is completed, workplace actions appropriate to the events that caused the intake are usually long past. Thus, the reasons for updating a worker's dose assessment are to adjust the cumulative total effective dose and to update projected values of future bioassay results. Identifying and confirming subsequent intakes requires knowing the expected magnitude of future excretion rates and retained quantities.

It is a good practice for sites to use long-term reevaluations to update assessments of lifetime dose. The adjustments to lifetime dose from significant intakes of radionuclides (especially plutonium and americium) can affect the worker's status with regard to the RadCon Standard Lifetime Control Level.

It is suggested that long-term reevaluations be performed when the committed effective dose is likely to affect the lifetime control level or when projected long-term bioassay measurements indicate that there may be impairment of ability to detect new intakes due to an elevated baseline.

9.5 GUIDANCE FOR PRACTICAL REPORTING OF INTERNAL DOSES

The uncertainty associated with dose assessments suggests that some rounding of doses is reasonable. The decision to round to two significant figures is consistent with the accuracy associated with the biokinetic models and dose factors. However, this can lead to the issue of how to sum (for example) a 1.2-mrems tritium dose with a 3.1-rems plutonium dose. Most database recording systems will treat the results as integer values and end up reporting 3,101 mrem. From a technical standpoint, the tritium dose would certainly be insignificant relative to the plutonium dose; however, from the regulatory perspective, both must be considered absolute values suitable for direct addition. Thus, it is recommended that once a dose is assigned for an intake, it be treated as an absolute value, with all the significant figures implied. This is not meant to imply that individual intake assessments should be recorded to the nth decimal place. The suggested practice is to round an internal dose to two significant figures for assignment to a specific intake, unless the dose is less than 10 mrem, at which point it is reasonable to round to the nearest integer value.

9.6 GUIDANCE ON CUMULATIVE *TED*

CFR 835.702(c)(5)(iii) requires maintaining records of cumulative total effective dose for each radiological worker for intakes occurring after January 1, 1989. It is a good practice to keep additional, separate records of calculated doses that are not limited to intakes occurring after January 1, 1989, but includes *TED* contributions from intakes prior to this time. This lifetime *TED* is consistent with the guidance concept of lifetime effective dose contained in NCRP Reports 91 and 116 (1987 and 1993). It provides a more complete estimate of lifetime cumulative dose for comparing with the RadCon Standard lifetime control level. While determination of *TED* for intakes received prior to January 1, 1989, is recommended to improve the consistency of available information, such determinations may not be possible due to resource or data limitations. In such cases, all available dose and intake data shall be maintained in an individual's records.

9.7 RECORDS ASSOCIATED WITH BIOASSAY MEASUREMENTS AND THEIR INTERPRETATION

Guidance on the type and extent of records associated with both in vivo and in vitro bioassay measurements can be found in American National Standard "Practice for Occupational Radiation Exposure Records System" (HPS 2010).

9.8 DOCUMENTING, RECORDING, AND RETAINING OF *PAEC*, *PAEE*, INTAKE, AND E_{50} FROM RADON AND THORON

Since radon quantities and units differ from the traditional activity concentration (expressed in uCi/cm^3) and intake (expressed in μCi), records for exposures and doses from radon, thoron, and their short-lived decay products will be different. Record should include

- radon concentrations, if measured (pCi/L may be used for the time being, but units must be specified, never assumed)
- the value of F_{Rn} (if applicable) and whether it is assumed or measured
- worker exposure times or stay times (hours)
- assigned protection factors (*APF*) for respirators, if any
- potential alpha energy concentration, *PAEC* (WL)
- potential alpha energy exposure, *PAEE* (WLM)
- radon and thoron progeny intake, *I*, in J
- dose conversion factors (rems/WLM; these may change in the future)
- E_{50} and $H_{lung,50}$.

Each exposed worker must be unambiguously associated with the air sample result that represents his or her exposure, including the flow rate, filter type, start time, stop time, and date(s) of operation.

Calibration records for and the identities of active air samplers used for personnel monitoring must be accessible. Radiological work permits (RWPs) may be a convenient way to record this information. Archived procedure manuals must specify instructions for operation of active air samplers and the types of filters that are acceptable for use.

10 MEDICAL RESPONSE

10.1 NEED FOR MEDICAL RESPONSE

Medical intervention may be needed to reduce the committed doses from significant intakes of radionuclides. This intervention can take the form of prophylactic treatment (therapy administered before an intake has occurred or been confirmed) or treatment in direct response to identified intakes. Examples of prophylactic treatment include administration of potassium iodide to emergency response workers for prevention of radioiodine uptake, and immediate administration of a chelating agent following a suspected intake of certain actinides (e.g., plutonium, americium, curium) but before any confirming bioassay measurements. Treatment in response to identified intakes includes diuretics following tritium exposure, and use of adsorption agents to prevent gastrointestinal tract uptake from ingestion or inhalation exposures.

Example 10.1 provides three situations where medical treatment and associated internal dosimetry concerns occur simultaneously. These examples are intended to show the kinds of circumstances which should be addressed by the medical response action plan of Section 3.2.3.

Example 10.1. Situations Where Internal Dosimetry Actions and Medical Treatment Occur Simultaneously

1. A chemical (or steam) explosion results in severe contaminated lacerations, imbedded contaminated particles, and chemical (or thermal) burns. The worker requires emergency room medical treatment for physical trauma injuries. Contamination may be significant and raises some concerns for treatment staff.

2. While working in a plutonium glove box, a worker incurs a contaminated puncture wound in the index finger. Initial surveys of the wound site and blood smears indicate potential doses could exceed several times the allowable occupational limits. The worker has no other injuries and the wound itself is quite small (suitable for an adhesive bandage and a tetanus shot). However, dose therapy should consider tissue excision and DTPA chelation by appropriate medical staff.

3. Following exposure to tritium gas, a single void urine sample indicates a significant tritium oxide intake warranting diuresis as a therapeutic action. There are no physical injuries. Diuresis involves administration of diuretics and medical monitoring of blood chemistry for electrolyte control.

Each of these examples poses different questions for resolution in an action plan for medical response. Key points the action plan shall address the following:

- Identification of parties involved in response (facility, health physics support, initial medical response, emergency medical dispatch, hospital, etc.)
- Statement of authority & responsibilities for each party
- Identification of action levels, or reference to documentation of action levels

- Identification of policies, manuals, or procedures providing key details of response
- Notification and communication chains
- Guidance for actions, evaluations, work restriction
- Management approval by significant parties involved.

A common point of tension in combined medical emergency and radioactivity intake event is a question of priority of treatment. The general guidance is that medical treatment takes priority. Decontamination is of little immediate value in a major trauma emergency and is certainly of secondary concern to lifesaving activities. However, in many of the combined medical and radioactivity intake event, both insults are relatively minor. Under these circumstances, it is a good practice for both the health physicist and the physician to discuss their respective concerns with the potential intake and the injury and prioritize the treatment for the particular case at hand. Ultimately, the physician has responsibility for the treatment of the victim.

10.2 ROLE OF THE HEALTH PHYSICIST IN MEDICAL TREATMENT

Radiation protection and health physics expertise is rare in occupational medicine physicians and medical staff. Thus the health physicist will likely need to work closely with medical staff in dose reduction therapy. The decision to commence therapy for dose reduction is a medical decision which cannot be delegated to the health physicist. However, the health physicist can identify the circumstances under which therapy would seem appropriate, and advise the medical staff on the likely efficacies of treatment alternatives. Once therapy has commenced, bioassay measurements are required to determine the efficacy of therapy. The interpretation of those bioassay measurements will likely fall to the health physicist.

DOE facility health physics staff should establish contact with the cognizant medical staff prior to an emergency. Once a significant potential intake event occurs, the administrative and technical pressures associated with response and case management can become intense. Prior efforts to establish good communications will pay dividends.

10.3 TREATMENT CRITERIA - WHEN TO TREAT

Deciding when medical response is needed poses some real challenges. There is a practical need for field-identifiable criteria which can be interpreted as action points for initiating medical response. Such criteria may include *DAC*-h exposure to airborne radioactivity, nasal smear activity levels, personal skin contamination levels, wounds caused by contaminated objects, or special bioassay measurement results.

Developing specific field criteria to identify the need for medical response can be challenging. Inhalation intake estimates based on *DAC*-h exposure are straightforward and discussed earlier in this document. Early bioassay measurement levels corresponding to the action levels could be established. Examples of one site's approach are summarized in Table XII and Table XIII (Carbaugh 2007). Another method is to develop field observation criteria (e.g., nasal smear or skin contamination criteria) which might indicate an action level has been exceeded. This latter approach is highly subjective with any number chosen likely to be arguable. Knowledge of facility operations, material forms, and past experience will likely play a key role in development of such criteria.

LESSONS LEARNED
Plutonium Contamination of Workers
At the Zero Power Physics Reactor
November 11, 2011

On November 8, 2011, workers at the Zero Power Physics Reactor were packaging clad plutonium fuel plates in a material handling hood (NE-ID-BEA-ZPPR-2011-0001). Two of the fuel storage containers had atypical labels indicating potential abnormalities with the fuel plates. After management review of the situation, authorization was given to proceed. The fuel storage container was opened and the workers discovered a fuel plate wrapped in plastic and tape. When the workers attempted to remove the wrapping, there was a release of powder. High contamination levels were found on the inside of the fuel storage container. The workroom continuous air monitor (CAM) alarmed and workers exited the facility. Sixteen workers were exposed to airborne plutonium.

All sixteen affected individuals were counted for 30 minutes in the lung counter; two individuals had positive results. A second lung count the next day showed no detectable activity for one and a 40% decrease for the other. The committed effective dose was estimated to be less than 2.1 rem; the committed equivalent dose to the bone surfaces was less than 71 rem.

A number of causes were identified. The direct cause was the cutting and handling of the plastic wrapping around the degraded fuel plate. Personnel responsible for planning and executing the work did not recognize the potential degradation of the fuel element over the years. It had been eighteen years since the plates had been used; documentation and specifications had been lost. In addition, the marked-up labels should have been an indication of something wrong.

Table XII. Early Bioassay Measurement Results Corresponding to the Therapeutic Intervention Action Levels Used at the Hanford Site (Carbaugh 2007) (Part 1)

Isotope and Dose (E_{50})	Measurement	Result	Action	Possible Treatment
Tritium				
2 rems	Single-void urine 3-4 h after exposure	10^6 dpm/mL	Consider therapy	Fluids, diuretics
20 rems	Same	10^7 dpm/mL	Strongly recommend treatment	Fluids, diuretics
Mixed Fission Products				
2 rems (assumes 2:1 Sr/Cs ratio)	Whole body count, or urine/fecal for severe intakes	>2500 nCi uptake, or >40,000 nCi if no Sr present	Consider therapy	Prussian blue Ca,(Sr), ammonium phosphate, others
20 rems (assumes 2:1 Sr/Cs ratio	Same	>25,000 nCi uptake, or >400,000 nCi if no Sr present	Treatment strongly recommended	Same
^{90}Sr				
2 rems	Second-void spot urine	>200,000 dpm in spot	Consider therapy	Alginate, Ca

	or in vivo detection	urine, or >*MDA* in vivo		gluconate, Sr lactate, others
20 rems	Same	>2,000,000 dpm in spot urine, or >50 μCi in vivo	Treatment strongly recommended	Same

10.4 TREATMENT PROTOCOLS - HOW TO TREAT

Treatment can be considered to include both skin decontamination to prevent intake and intervention actions taken to reduce internal dose once an intake has occurred. Skin decontamination protocols beyond simple washing should be reviewed by appropriate medical authorities to ensure that skin integrity will not be breached. Therapeutic actions to reduce internal dose once an intake has occurred will likely require administration under the direction of competent medical authority.

Skin decontamination can generally be accomplished by simple washing with mild soap and water. This is frequently done with the assistance of a radiological control technician and, for intact skin, would not necessitate medical assistance. If contamination persists, an abrasive pumice soap, detergents, and commercial decontamination agents containing complexing agents such as EDTA (ethylenediaminetetra-acetic acid) may be effective. A final step in skin decontamination is the use of a saturated solution of potassium permanganate which is painted onto the skin with an applicator or cotton ball, followed by removal using a sodium bisulfite solution. The potassium permanganate/ sodium bisulfite procedure removes a thin layer of dead skin. Repeated applications of this method are cautioned because its overuse can result in epidermal irritation or burning, with possible loss of skin integrity and subsequent uptake. An extreme example of decontamination is the surgical debridement (aggressive cleaning) or excision (cutting out) of contaminated material from a wound. Details on skin decontamination methods can be found in NCRP Report 65 (NCRP 1980), IAEA Safety Series No. 47 (IAEA 1978b), the Radiological Health Handbook (Bureau of Radiological Health 1970), and the Health Physics and Radiological Health Handbook (Shleien 1992).

Table XIII. Early Bioassay Measurement Results Corresponding to the Therapeutic Intervention Action Levels Used at the Hanford Site (Carbaugh 2007) (Part 2)

Isotope and Dose (E_{50})	Measurement	Result	Action	Possible Treatment
Uranium, Soluble				
Potential Kidney toxicity	Chest count Second-void urine sample 12-hour urine sample	>*MDA*(14-21 mg) >0.1 mg >0.5 mg	Consider therapy	Na or Ca bicarbonate; intestinal adsorbents
Uranium Insoluble[(a)]				
2 rems	Chest count	>*MDA* for ^{235}U or ^{234}Th	Consider therapy	None recommended
200 rems	Same	100 x ALI	Treatment strongly recommended	Lung lavage
Plutonium or ^{241}Am				
2 rems	Chest count	>*MDA* for Pu or	Consider therapy	DTPA

	Early urine sample	^{241}Am >4 dpm when extrapolated to first day excretion		
[a] If soluble component is present, then urine sampling is appropriate. Use same action levels as above for soluble uranium.				

Therapeutic actions to reduce internal dose following the intake of radioactive material typically require medical administration of an agent to block, chelate, dilute, or purge the body of the radioactivity. Blocking agents are used to prevent gastrointestinal absorption through ion exchange processes (e.g., Prussian blue for cesium blockage) or adsorption (e.g., antacids or alginates for strontium). These may be coupled with stomach lavage, emetics, and purgatives or laxatives to accelerate removal or passage through the GI tract. Chelating agents, e.g., DTPA for plutonium or americium, are usually administered by intravenous injection and bind with ionic forms in the blood. They are then rapidly excreted in urine. Dilution of radioactivity can be accomplished by administering a relatively large dose of the stable form of the element, thereby reducing the likelihood of retention of the radioactive form (e.g., administration of stable potassium iodide in response to exposure to ^{131}I). Acceleration of normal metabolism to speed removal of radioactivity can be effective (e.g., diuretics to accelerate body water turnover to eliminate tritium). For extreme cases of insoluble particle inhalation, lung lavage may be an effective therapy.

Details concerning the effective methods of treatment and therapy for various radionuclide intakes can be found in the Guidebook for the Treatment of Accidental Internal Radionuclide Contamination of Workers edited by Gerber & Thomas (Bhattacharyya et al. 1992), NCRP Report No. 65 (NCRP 1980), IAEA Safety Series No. 47 (IAEA 1978b), IAEA Technical Report Series No. 184 (IAEA 1978a), and ICRP Publication 28 (ICRP 1978a). These documents shall be immediately available to health physics and medical personnel.

An additional resource for assisting with the medical management of radiation accidents is the Radiation Emergency Assistance Center and Training Site (REAC/TS), a service operated for the U.S. Department of Energy by the Oak Ridge Institute for Science and Education (ORISE). REAC/TS maintains a 24-hour emergency contact list, which can be reached by phone at (865) 576-3131 from 8 am to 4:30 pm Eastern Time and at other times, (865) 576-1005 (DOE Oak Ridge Operations Emergency Operations Center).

Sites with potential for intakes of transuranics shall have access to a supply of DTPA and a physician. DTPA is approved by the U.S. Food and Drug Administration and is available to physicians. For more information contact the REAC/TS DTPA program director Albert Wiley, MD, PhD, REAC/TS Director, 865-576- 3131 or albert.wiley@orise.orau.gov. The REAC/TS website page also has information on drug package inserts and other information which may be helpful (www.orise.orau.gov/REACTS). Radiological control personnel expected to support the medical personnel should take the time to not only get to know the medical staff, and vice versa, but to also familiarize themselves with what they may encounter.

10.5 IMPACT OF THERAPY ON DOSIMETRY

Most procedures and computer codes used for routine intake and internal dose assessment are based on standard ICRP assumptions for the biokinetics of radioactivity in the body. Dose reduction therapy can have significant impact on the validity of these assumptions. The nature of the impact depends on the type of therapy and the radionuclide of interest. There is no single rule for evaluating data

following dose reduction therapy. It is imperative that the dosimetrist understand the therapeutic processes involved and the impact on bioassay measurements. Some examples follow.

The use of diuretics to accelerate body water turnover effectively decreases the biological retention of tritium. Since tritium body water concentration can be easily measured by urinalysis, the actual biological half-time can be determined empirically for the affected individual, and appropriate modification made to dose calculations.

DTPA chelation therapy for transportable plutonium can create enormous uncertainty in the use of urine data for estimating intake. The DTPA can enhance urinary excretion of plutonium by a nominal factor of 10 to 100. Because therapy should be given as close to the time of intake as can be reasonably accomplished, there is little likelihood of identifying a pre-therapy baseline in urine. Methods for evaluating chelated data have recently been described by La Bone (La Bone 1994a, 1994b) and Carbaugh (Carbaugh et al. 1989). However, there is no standard approach. Historically, cases which were treated with DTPA were evaluated for uptake based on urine data obtained at times unaffected by chelation (e.g., 100 days after therapy) with the early data ignored. This approach gives an "effective" uptake estimate. Uncertainties will still exist in the fractionation and retention factors for organs and tissues as a result of chelation. Inhalation intake can still be assessed from early data on fecal excretion, which, compared to data on urinary excretion, are relatively unaffected by DTPA.

In vivo measurements can be used to monitor the effectiveness of therapy for removal of ^{137}Cs, ^{131}I, or other high-energy photon-emitters. These measurements can allow appropriate adjustment to be made to whole body or organ/tissue retention functions.

Bioassay measurements take on a dual role during dose reduction therapy. In addition to their use for dosimetry, their relative magnitude can be a valuable indication of the effectiveness of therapeutic actions. In some cases, crude measurements may be very valuable to indicate the efficacy of therapy; however, their value for the final intake and dose assessments may be quite limited.

Dose reduction therapy places great strains on an internal dosimetry/bioassay program. The dosimetrist must recognize the many potential impacts on bioassay measurements caused by therapy and factor these into the data interpretation. Where normal dosimetry would call for emphasis on a set of measurements which might be significantly affected by therapy, good practice suggests that estimates be obtained by as many alternate methods as reasonable and wise judgment exercised in final interpretation.

10.6 COUNSELING WORKERS

Counseling of workers who have incurred intakes of radioactivity should be performed to clarify the significance (or insignificance) of an intake and provide workers with the information needed to help resolve any concerns about medical or radiological effects. Such counseling is also an opportunity to discuss any needs for long-term follow-up bioassay measurements or dose reevaluations. Documentation of counseling may take the form of a memo to file, letter to worker, or simply a checklist of subjects discussed. Documented acknowledgment of the counseling session by the worker is desirable. However, the need for such acknowledgment does not justify any effort beyond that normally used for routinely reporting medical exam or bioassay measurement results.

11 QUALITY ASSURANCE

This section addresses quality assurance in general and independent review of dose assessments and computer software.

11.1 GENERAL NEEDS

Quality assurance needs for various aspects of internal dosimetry programs are described by the American National Standards Institute in published and soon-to-be published standards (HPS 2011a & 2000 and ANSI 1996). Berger has given an excellent general overview (Berger 1994). Accreditation through the U.S. Department of Energy Laboratory Accreditation Program (DOELAP) will include all of the quality assurance features needed for radiobioassay laboratories (DOE 1998b). The DOELAP program for radiobioassay laboratories follows many of the precedents set in the field of external dosimetry (DOE 1998b; McDonald et al. 1992).

11.2 INDEPENDENT REVIEW

When doses are large with respect to the *IL* and there is controversy over a dose assessment, an independent review shall be performed. The experience of one such review is provided by La Bone et al. (La Bone et al. 1992). Agreement within a factor of two among experienced dose assessors is probably the best that can be hoped for in difficult cases such as transuranic intakes with subsequent chelation. Easier, more straightforward cases result in better agreement during intercomparisons (Hui et al. 1994).

11.3 COMPUTER SOFTWARE QUALITY ASSURANCE

Computer software is an important tool in internal dosimetry. The software may include commercial dosimetry codes, site- or contractor-developed dosimetry codes, calculational algorithms incorporated into commercial application codes (e.g., spreadsheets), and database application software for management, manipulation, and reporting of data. Quality assurance activities involve configuration management, code testing, error correction, and security

Relevant requirements for software quality assurance are found in DOE Order 414.1D, *Quality Assurance.* Guidance is found in DOE Guide 414.1-4, *Safety Software Guide for Use with 10 CFR 830 , Subpart A, Quality Assurance Requirements, and DOE Order 414.1C*

11.3.1 Configuration Management

Dosimetry codes shall be subject to configuration management, including records of the version of the code, the user's manual, instructions for running the code, limitations of the code, hardware requirements, acceptance testing records, and a copy of the code itself.

11.3.2 Verification and Validation (Acceptance) Testing of Codes

Computer codes shall undergo a two-step verification and validation (V&V) process as acceptance testing before their routine use for dosimetry. This process shows that the code produces valid responses when used to analyze problems within a specific set of parameters and parameter values. *Verification* involves determining program requirements, range of program results that may be considered valid, or criteria to be used in evaluating the validity of results. *Validation* is the process of testing a computer program under a specific computing system and evaluating the results to ensure the compliance with specified requirements. Part of the testing should include running selected "benchmark" cases for comparison against an independent solution process (e.g., hand calculations, published tabulations of

reference man dose, results from other verified code, etc). Results of this testing should be maintained with the site or contractor internal dosimetry program records. This testing should be successfully completed before the code or algorithm is used for dosimetry calculations of workers.

"Existing software" is any software program that has been developed, put into operation and shown to possess desirable capabilities, but for which a formal V&V report is not available. Routine testing of this software shall be performed on a periodic basis utilizing corresponding nuclide doses and retention functions listed in the site or contractor technical basis documentation as models. The test of the software should follow the same procedure or process used for case assessments.

V&V shall be conducted according to a plan which specifies the following:

- application for which the program is to be utilized
- range of results that may be considered valid (i.e., acceptance criteria)
- user environment (hardware and operating system specifications, hardware user interface requirements, etc.).

V&V testing shall be peer-reviewed by a staff member other than the person who performed the test. A report of the V&V test should be recorded in the site or contractor internal dosimetry program records for each software application and include the following:

- identification of the program tested, scope of the test report
- description of the test environment - hardware configuration, software used
- description of the test results, copy of the test case log
- verification that all results are identical to previous results.

Occasional verification testing of infrequently used codes can be valuable to ensure that hardware and operating system changes have not affected the ability to use the code.

11.3.3 Corrections of Software Errors

In the case of errors with commercial software packages, the software system files should be reinstalled and a V&V test conducted to ensure correction of the problem. If errors continue, the next step is to contact the software vendor.

11.3.4 Software Security

Backup copies of all internal dosimetry software and data shall be kept in a secure place. Another copy should be stored at a different location for disaster recovery. Documentation of the procedure to install the software should be included with the backup copies.

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APPENDIX A. REVIEW OF MEASUREMENTS OF EQUILIBRIUM FACTORS FOR RADON AND THORON PROGENY

Values of radon progeny equilibrium factors have been published in the literature. Equilibrium factors depend on many variables, including whether measurements are made indoors or outdoors, whether there is smoke and dust in the air, the proximity of the radon source, and the rate of air exchange or wind speed.

A.1. MEASUREMENTS OF RADON PROGENY EQUILIBRIUM FACTORS

Fifteen results of outdoor F_{Rn} studies and three recommended values are summarized in Table XV. Observed values range from 0.01 to 1.00, with an average value of 0.39 and average ranges from 0.16 to 0.73. Since these measurements were made under very different circumstances, the wide range of values is not surprising. These results show that local characterization of F_{Rn} is advisable. Recommended values of 0.7 or 0.8 are higher than have been observed at the Uranium Mill Tailings Remedial Action (UMTRA) Project (Reif and Andrews 1992) and recently in the southeastern and southwestern USA in 240 measurements at 16 sites (Wasiolek and James 1995) and at the Fernald Environmental Management Project (FEMP) under several stability classes (Medora 1996).

Table XIV. Radon Equilibrium Factors Measured Outdoors

			Range				
Country	**avg**	**sd**	**min**	**max**	**Reference**	**Environment**	**Rec.**
USA	0.87				Cox et al. (1970)		
USA	0.79		0.57	0.89	George and Breslin (1980)		
USA	0.09		0.02	1.00	Reif and Andrews (1992)	at the source	0.1
USA	0.45		0.10	1.00	Reif and Andrews (1992)	upwind	0.4
USA	0.20		0.01	0.91	Reif and Andrews (1992)	downwind	0.2
USA	0.10		0.03	0.17	Borak (1983)		
USA	0.26				Schultz et al. (1994)		
USA	0.63		0.38	0.95	Wasiolek and James (1995)	varied	
USA	0.23	0.12	0.07	0.45	Medora (1996)	stabil. class A	
USA	0.22				Medora (1996)	stabil. class B	
USA	0.39	.20	0.22	0.63	Medora (1996)	stabil. class D	
USA	0.22	.04	0.17	0.25	Medora (1996)	stabil. class E	
Yugoslavia	0.25				Planinic and Faj (1990)		
West Germany	0.71				Jacobi (1972)		
West Germany	0.43		0.04	1.00	Keller and Folkerts(1984)		
AVERAGE	**0.39**		**0.16**	**0.73**	All Studies		
Std. Dev.	**0.25**		**0.18**	**0.33**			
Min of Minima	**0.01**						
Max of Maxima	**1.00**						
No. Studies	**15**						
recommendation					*UNSCEAR (1988) annex A para 93*		*0.8*
recommendation					*UNSCEAR (1993) annex A Table 24*		*0.8*
recommendation					*NCRP (1987)*		*0.7*

In Table XVI are 16 results of studies of FRn indoors, along with four recommendations for a default or assumed value (UNSCEAR 1988; UNSCEAR 1993; Porstendörfer and Reineking 1992; NCRP 1987a). Earlier data did not account for smoking, which is known to increase F_{Rn} and decrease the unattached fraction, f_p. In cleaner indoor air, lower values of F_{Rn} are observed (UNSCEAR 1993; Swedjemark 1983; NEA 1985). Observed values range from 0.04 to 0.97, with an average value of 0.43 and average ranges from 0.17 to 0.71. Since these measurements were made under very different circumstances, the wide range of values is not surprising. Most recommended values are 0.4, with one of 0.3. The ICRP has adopted 0.4 (ICRP 1993a).

Table XV. Radon Equilibrium Factors Indoors at Home

		Range					
Country	**avg**	**sd**	**min**	**max**	**Reference**	**Environment**	**Rec.**
Austria	0.60				Steinhausler et al. 1980		
Australia	0.32	0.09	0.17	0.49	Solomon and Ren (1992)		
Bangladesh	0.40	0.23	0.04	0.97	Farid (1993)		
Canada	0.35		0.17	0.65	McGregor and Gourgon (1980)	18 cities	
Canada	0.41				Scott (1983)		
Finland	0.47		0.30	0.63	Makelainen (1980)		
France	0.26		0.10	0.48	Tymen et al. (1992)		
Norway	0.50		0.30	0.80	Stranden et al. (1979)		
Sweden	0.44	0.12	0.10	0.80	Swedjemark (1983)		
Sweden	0.51				Jonassen and McLaughlin (1989)	smoker	
Sweden	0.46	.20			Jonassen and McLaughlin (1989)	nonsmokers	
USA	0.63	.04			George and Breslin (1980)	living areas	
USA	0.33				Israeli (1985)	living area	
West Germany	0.37		0.25	0.65	Wicke and Porstendorfer (1982)		
West Germany	0.34		0.10	0.90	Keller and Folkerts (1984)		
Yugoslavia	0.55				Planinic and Faj (1990)		
AVERAGE	**0.43**		**0.17**	**0.71**	All Studies		
Std. Dev.	**0.11**		**0.09**	**0.17**	All Studies		
Min of Minima	**0.04**				All Studies		
Max of Maxima	**0.97**				All Studies		
No. Studies	**16**						
recommendation					*UNSCEAR (1988) Annex A para 140*		*0.4*
recommendation					*UNSCEAR (1993) Annex A para 118*		*0.4*
Summary			*0.20*	*0.40*	*Porstendorfer and Reineking (1992)*		*0.3*
recommendation					*NCRP (1987)*		*0.4*

The workplace may have different aerosol characteristics from the home (either cleaner or dirtier). However, few measurements of F_{Rn} in the workplace are available. Two Japanese authors (Hattori and Ishida 1994) measured the equilibrium factor of ^{222}Rn in a pressurized water reactor auxiliary building for a year. In this clean, well-ventilated workplace, they observed a mean of 1,993

measurements of $F = 0.28 + 0.09$, with the lognormally distributed unattached fraction median $f_p = 0.069$ with a $GSD = 1.8$. In a boiling water reactor turbine building, they observed that the mean of 2,555 equilibrium factor measurements was $F = 0.32 \pm 0.10$ with a lognormally distributed unattached fraction median $f_p = 0.056$ with a $GSD = 2.0$. These workplace equilibrium factors (Table XVII) are lower than many of the home equilibrium factors given in Table XVI.

Table XVI. Radon Equilibrium Factors Indoors at Work

Country	avg	sd	min	max	Reference	Environment	Rec.
Japan	0.28	0.09			Hattori and Ishida (1994)	PWR Aux Bldg	
Japan	0.32	0.10			Hattori and Ishida (1994)	BWR Turb Bldg	
AVERAGE	**0.30**	0.10					
No. Studies	**2**						

In modern underground uranium mines, with their large ventilation rates, equilibrium factors are low (National Research Council 1991), as shown in Table XVIII. The average of three studies is 0.27. Such factors may apply to underground tunnel sites like the Waste Isolation Pilot Plant and the Yucca Mountain facility.

Table XVII. Radon Equilibrium Factors in Uranium Mines

Country	avg	sd	min	max	Reference	Environment	Rec.
USA	0.29				Kotrappa and Mayya (1976)		
USA	0.32				Holub and Droullard (1980)		
USA	0.19		**0.05**	**0.36**	George et al. (1977) All		
AVERAGE	**0.27**				All Studies		
No. Studies	**3**						
recommendation					UNSCEAR (1988)		0.3

Non-uranium mines may have lower ventilation rates, and radon equilibrium factors are likely to be higher, as shown in the four results listed in Table XIX. The average of four studies is 0.55.

Table XVIII. Radon Equilibrium Factors in Non-Uranium Mines

Country	avg	sd	min	max	Reference	Environment	Rec.
Norway	0.50				Stranden and Berteig (1982a, 1982b)		
Poland	0.30				Domanski et al. (1979)		
Sweden	0.70				Snihs (1977)		
UK	0.70				Strong et al. (1975)		
AVERAGE	**0.55**						
No. Studies	**4**						

A.2. MEASUREMENTS OF THORON PROGENY EQUILIBRIUM FACTORS

Because of thoron's short half-life, measurements of thoron progeny are generally made, rather than of thoron gas. Thus, equilibrium factors for thoron are less well known, and more research needs to be done (UNSCEAR 1993). Recommended values for indoors and outdoors are given in Table XX (UNSCEAR 1988; UNSCEAR 1993). The outdoor numbers, 0.02 (1988) and 0.01 (1993) are lower than the 0.04 default number given above, while the indoor numbers, 1/6 and 0.1, are higher than the 0.04 number. Because of this, it is good practice to measure thoron progeny directly when possible.

Table XIX. Thoron (^{220}Rn) Equilibrium Factors

[Country]	avg	sd	min	max	Reference	Environment	Rec.
Outdoors							
recommendation					UNSCEAR (1988)		0.02
recommendation					UNSCEAR (1993) Annex A Para 120		0.01
Indoors - Home							
recommendation					UNSCEAR (1988)		1/6
recommendation					UNSCEAR (1993) Annex A Para 120		0.10

APPENDIX B. DEFINITIONS OF OBSOLETE OR REVISED QUANTITIES USED IN HISTORICAL RECORDS

Various quantities have been used over the years in various DOE Orders and Regulations. Below are definitions of quantities that either have been revised or are no longer used but that appear in historical records and in the technical and scientific literature.

annual effective dose equivalent (AEDE): The sum of *effective dose equivalent* from both the internal and external irradiation of tissues and organs received in one calendar year. This definition is retained from the 1989 version of DOE Order 5480.11 because records from that period include this quantity.

annual limit on exposure (ALE): The limit for *potential alpha energy exposure* to the progeny of ^{222}Rn or ^{220}Rn, expressed in units of working level months (WLM) (ICRP 1981b). An implicit ALE for other radionuclides is 2000 *DAC*-h.

annual limit on intake (ALI): The derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. *ALI* is the smaller value of intake of a given radionuclide in a year by *Reference Man* that would result in a committed effective dose equivalent of 5 rems (0.05 sievert) or a committed dose equivalent of 50 rems (0.5 sievert) to any individual organ or tissue. 10 CFR 835.2 specifies that *ALI* values for intake by ingestion and inhalation of selected radionuclides are based on Table 1 of Federal Guidance Report No. 11 (Eckerman et al. 1988). (10 CFR 835.2)

> Note: *The ALI for ^{222}Rn and ^{220}Rn progeny is most correctly expressed in joules (J) of potential alpha energy (ICRP 1981b). Stochastic ALI (SALI) values and nonstochastic ALI (NALI) values result from different dose limits. Intake of 1 SALI results in 5 rems committed effective dose equivalent, while intake of 1 NALI results in 50 rems committed effective dose to the most highly exposed tissue or organ.*

committed dose equivalent ($H_{T,50}$): The dose equivalent calculated to be received by a tissue or organ over a 50-year period after the intake of a radionuclide into the body. It does not include contributions from radiation sources external to the body. Committed dose equivalent is expressed in units of rems (or sieverts). (10 CFR 835)

> Note: *For exposures to the short-lived radioactive progeny of ^{222}Rn and ^{220}Rn, see the definition of committed effective dose equivalent (below).*

committed effective dose equivalent ($H_{E,50}$): The sum of the committed dose equivalents to various tissues or organs in the body ($H_{T,50}$) each multiplied by the appropriate tissue weighting factor (w_T): that is, $H_{E,50} = 3w_T H_{T,50.}$ Committed effective dose equivalent (CEDE) is expressed in units of rems (or sieverts). (10 CFR 835)

> Note: *For exposures to the short-lived radioactive progeny of ^{222}Rn, committed effective dose equivalent is calculated directly from workplace measurements of*

potential alpha energy exposure using a dose conversion factor of 1.25 rems (0.0125 Sv) per working level month (WLM). For exposures to the short-lived radioactive progeny of ^{220}Rn, committed effective dose equivalent is calculated directly from workplace measurements of potential alpha energy exposure using a dose conversion factor of 5/12 rems (5/1200 Sv) per WLM. Since the lung is the only tissue significantly irradiated by radon and thoron, the committed dose equivalent to lung due to exposures to radon and thoron is calculated by dividing the committed effective dose equivalent from radon and thoron by the tissue weighting factor for lung ($w_T = 0.12$).

derived air concentration (*DAC*): For the radionuclides listed in Appendix A of 10 CFR 835, the airborne concentration that equals the *ALI* divided by the volume of air breathed by an average worker for a working year of 2000 hours (assuming a breathing volume of 2400 m^3).

Note: Prior to the 2007 amendment to 10 CFR 835 the Appendix A values were are based upon ICRP 26/30 methodology and used conversion factors from Federal Guidance Report No. 11.

Note: *The footnotes to Appendix A of 10 CFR 835 give some important information about the DACs listed. In particular, the right-hand column identifies the origin of each DAC–whether it was derived from the stochastic dose limit or the non-stochastic dose limit for a particular organ. Only DACs derived from the stochastic dose limit can be used to calculate committed effective dose directly from air sampling data.*

For radionuclides listed in Appendix C of 10 CFR 835, the air immersion *DAC*s were calculated for a continuous, non-shielded exposure via immersion in a semi-infinite atmospheric cloud. The values are based upon the derived airborne concentration found in Table 1 of Federal Guidance Report No. 11 (Eckerman et al. 1988). (10 CFR 835, RadCon Standard)

gastrointestinal (GI) tract model: A mathematical representation of the behavior of radionuclides in the contents of the human gastrointestinal tract.

minimum detectable (effective) dose: The minimum detectable committed (effective) dose equivalent associated with a bioassay program. Formerly called "missed dose."

respiratory tract model: A mathematical representation of the behavior of particles and gases in the human respiratory tract.

CONCLUDING MATERIAL

Review Activity:

DOE
NNSA
HSS
EM
NE
SC

Field Offices
AL
ID
SR

National Laboratories
BNL
LLNL
LANL
PNNL
Sandia

Area/Site Offices
Amarillo Area Office
Kirtland Area Office
Princeton Area Office
Y-12

Other DOE Facilities
ORAU

Preparing Activity:
DOE-HS-11

Project Number:
SAFT-0117