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NOT MEASUREMENT SENSITIVE

DOE-STD-1121-98
December 1999

Reaffirmation with Errata
May 2003

DOE STANDARD

INTERNAL DOSIMETRY

U.S. Department of Energy
Washington, D.C. 20585

AREA SAFT

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Changes to DOE-STD-1121-98, Internal Dosimetry

Section/page/para	Change
iii	Change "Office of Worker Protection Programs and Hazards Management (EH-52)" to "Office of Worker Protection Policy and Programs (EH-52)". Add to bottom "DOE is considering updating its requirements for assessing and recording internal dosimetry results. The update will make DOE's system consistent with more recent national and international consensus standards. Should these changes be made, this standard will subsequently be updated to reflect these changes."
2 1.4, 2nd para	Change "with potential" to "likely to receive".
4 1.5	Change to read: To the extent possible, this guidance document is written to be consistent with existing non-governmental standards for internal dosimetry, including: • ANSI N42.22-1999, *American National Standard–Traceability of Radioactive Sources to the National Institute of Standards and Technology (NIST) and Associated Instrument Quality Control* • ANSI N42.23-1996, *Measurement and Associated Instrumentation Quality Assurance for Radioassay Laboratories* • ANSI N323a-1997, *American National Standard for Radiation Protection Instrumentation Test and Calibration* • ANSI Z88.2-1992, *American National Standard for Respiratory Protection*. • HPS N13.1-1999, *Guide to Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities* • HPS N13.6-1999, *Practice for Occupational Radiation Exposure Records Systems* • HPS N13.12-1999, *Surface and Volume Radioactivity Standards for Clearance* • HPS N13.14-1994, *Internal Dosimetry Programs for Tritium Exposure, Minimum Requirements* • HPS N13.22-1995, *Bioassay Programs for Uranium* • HPS N13.30-1996, *Performance Criteria for Radiobioassay* • HPS N13.39-2000, *Design of Internal Dosimetry Programs* • HPS N13.42-1997, *Internal Dosimetry for Mixed Fission and Activation Products*.
20 3, last para	Delete " DOE Order N 441.1 (DOE 1995) or".

29 4, 1st para 2nd para 4.1	Change "DOE sites should strive to comply with Draft ANSI N13.39, "Design of Internal Dosimetry Programs - Minimum Acceptable Requirements" (HPS 1996c)" to "DOE sites should strive to comply with ANSI N13.39, "Design of Internal Dosimetry Programs " (HPS 2000)". Delete "Draft". Change "10 CFR 835.209(c)" to "10 CFR 835.209(b)". Change "based on representative air concentration values" to "based on air concentration values". Add to end "It may also include routine bioassay for isotopes with high dose per intake coefficients, such as many of the transuranics."
30 2nd para 4th para, end	Change "DOE Office of Worker Protection and Hazards Management has prepared" to "DOE Office of Worker Protection Policy and Programs (previously the Office of Worker Protection and Hazards Management) prepared". Delete extra quotation mark.
39 4.4.4, 1st sentence	Change "potential" to "likelihood".

43 4.4.6, 1st sentence	Change "Bioassay laboratories or service laboratories should participate in the U.S. Department of Energy Laboratory Accreditation Program (DOELAP) for radiobioassay laboratories (DOE 1998b). Bioassay laboratories or service laboratories should meet the requirements of HPS N13.30-1996, "Performance Criteria for Radiobioassay" (HPS 1996a). " to "10 CFR 835.402 requires internal dose monitoring programs implemented to demonstrate compliance with 10 CFR 835 be accredited by the U.S. Department of Energy Laboratory Accreditation Program (DOELAP) for Radiobioassay (DOE 1998a). Radiobioassay laboratories utilized by the internal dose monitoring programs will be evaluated against the requirements of the "Department of Energy Laboratory Accreditation Program for Radiobioassay" (DOE 1998b) which incorporates the recommendations of HPS N13.30-1996, "Performance Criteria for Radiobioassay" (HPS 1996a)."
Example 4.7 1st para	Change "N13.30-1966" to "N13.30-1996".
3rd para	Change "The radiobioassay laboratory (if required by 10 CFR 835) shall participate in any laboratory quality assurance programs that may be offered by the DOE" to "The radiobioassay laboratory shall participate in the DOELAP for Radiobioassay".
5th para	Delete "which complies with applicable sections of DOE Order 5700.6C (or current version)".
44 3rd para	Change "N13-30-1966"" to "N13-30-1996".
45 last para in example	Change "N13-30-1966"" to "N13-30-1996". 2 times
49 5,1, 1st sentence	Change "dose evaluation" to "dosimetry".
50 1st para	Change "putonium" to "plutonium".
53 Criterion 7	Change "There are two options available for record-keeping if bioassay is never obtained: 1) ignore annual accumulations less than 40 *DAC*-h, or 2) provide intake and dose assessments based solely on *DAC*-h." to "For this situation, sites should still provide intake and dose assessments based solely on *DAC*-h."
72 last line	Change "ICRP (1982a ¶120, 1988 ¶79)" to "ICRP (1982, 1988)".

74 After Eq 19.	Change “Figure 6” to “Figure 7”.
75	Top 2 lines from page 76 moved to bottom of page.
76 Last sentence	Change “10 CFR 835.403(a)(1)” to “10 CFR 835 Appendix A”.
91 8.2, 1st sentence	Delete “and in DOE N441.1 (1995)”.
92 8.2.2, 1st sentence	Delete “required by DOE N441.1 (DOE 1995),”.
95 9, 3rd para.	Change “(ANSI 1989)” to “(HPS 1999c)”.
98 9.7.	Change “(ANSI 1989)” to “(HPS 1999c)”.
104 3rd and 4th para.	Change “REAC/TS maintains a 24-hour emergency contact list, which can be reached by phone at (423) 576-3131 from 8 am to 4:30 pm Eastern Time and at other times, (423) 481-1000 (Methodist Medical Center switchboard; ask for REAC/TS staff person on call). ” “As of September 1996, physicians can register as IND co-investigators by contacting the REAC/TS DTPA program, Ronald E. Goans, M.D., Head of Medical Section, at (423) 576-4049.” to “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). ” “As of September 1996, physicians can register as IND co-investigators by contacting the REAC/TS DTPA program, Patrick Lowry, M.D., Head of Medical Section, at (865) 576-4049.”
106 11.1, 1st sentence.	Change “1996c” to “2000”.
107 3rd para, 2nd sentence.	Delete “, according to ANSI/ANS N10.4-1987 (ANSI 1987),”.

108-111	Delete 1st reference, rename and move 2nd and 5th as HPS standards with new dates to 111, revise date on 4th. Revise HPS Draft N13.12 and N13.39.

FOREWORD

1. This Department of Energy (DOE) standard is approved for use by all DOE Components and their contractors.

2. Constructive comments (recommendations, additions, deletions) and any pertinent data that may improve this document should be sent to

Office of Worker Protection Policy and Programs (EH-52)
U.S. Department of Energy
Washington, DC 20585

by letter or by using the self-addressed Document Improvement Proposal form (DOE F 1300.3) appearing at the end of this document.

3. DOE technical standards, such as this standard, do not establish requirements. However, all or part of the provisions in a DOE standard can become requirements under the following circumstances:

(1) they are explicitly stated to be requirements in a DOE requirements document; or

(2) the organization makes a commitment to meet a standard in a contract or in an implementation plan or program plan required by a DOE requirements document.

Throughout this standard, the word "shall" is used to denote actions which must be performed if the objectives of this standard are to be met. If the provisions in this standard are made requirements through one of the two ways discussed above, then the "shall" statements would become requirements. It is not appropriate to consider that "should" statements would automatically be converted to "shall" statements as this action would violate the consensus process used to approve this standard.

DOE is considering updating its requirements for assessing and recording internal dosimetry results. The update would make DOE's system consistent with more recent national and international consensus standards. Should these changes be made, this standard will subsequently be updated to reflect these changes.

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1.0 INTRODUCTION

Internal dosimetry is "...the scientific methodology used to measure, calculate, estimate, assay, predict, and otherwise quantify the radiative energy absorbed by the ionization and excitation of atoms in human tissues as a result of the emission of energetic radiation by internally deposited radionuclides" (Raabe 1994). Radiation protection requirements for U.S. Department of Energy (DOE) and DOE-contractor employees are given in DOE's *Occupational Radiation Protection*, Title 10, Code of Federal Regulations, Part 835 (DOE 1998a). In this Technical Standard this regulation will be referred to as "10 CFR 835." . Further, the *Radiological Control Standard* ("RadCon Standard;" DOE 1999e) contains provisions that apply to many contractors by virtue of being included in their contract. DOE's 10 CFR 835 and RadCon Standard require monitoring of the workplace, and monitoring of radiation workers who, under typical conditions, are likely to receive 0.1 rem (0.001 Sv) or more committed effective dose equivalent, and/or 5 rems (0.05 Sv) committed dose equivalent to any organ or tissue, from all occupational radionuclide intakes in a year. The regulation 10 CFR 835 also requires that measurements of internal radionuclides and the assessments of committed effective dose equivalent resulting from intakes of radionuclides be recorded, reported, and archived.

1.1 SCOPE

This document applies to the internal dosimetry aspects of all Radiation Protection Programs of DOE and its contractors as required by 10 CFR 835.101 for the conduct of radiological work. As such, it provides detailed technical guidance on internal dosimetry to DOE and DOE-contractor personnel in fulfilling the requirements of 10 CFR 835 and applicable provisions of the RadCon Standard, as elaborated in the *Internal Dosimetry Program Guide* (DOE 1999b) by clarifying the requirements and providing specific examples of practical methods for conducting an effective internal dosimetry program. Guidance is provided on organization, staffing, training, and facilities; documents and plans; design of and participation in the bioassay program; internal dose evaluation; internal dose management; recording internal doses and related information; reporting of internal doses; medical response; quality assurance; and guidance for monitoring in the workplace as it applies to internal dosimetry. Details are provided on internal dosimetry aspects associated with radon, thoron, and their long-lived progeny; applications of models to bioassay data; dose assessment techniques; use of significant figures; and a guide to the wealth of internal dosimetry information at the various DOE sites.

1.2 PURPOSE

This technical standard is created to provide a resource for those engaged in the science and practice of internal dosimetry within the DOE complex. This standard defines minimum levels of acceptable performance and provides basic procedural guidelines for evaluating the internal radiation dose equivalent that may be received by radiation workers from intakes of radionuclides. This set of defined internal dosimetry performance criteria meets the requirements set forth in 10 CFR 835 for monitoring the workplace, for assessing internal radiation doses to workers at DOE facilities, and for recording and reporting requirements as they apply to internal dosimetry programs.

1.3 USE

This standard is for use in implementing the specific parts of the radiation protection programs required by 10 CFR 835.101 that relate to internal dosimetry programs. DOE and DOE-contractor

personnel may use the specific methods and references in this standard as examples of acceptable means and methods to meet the internal dosimetry requirements of 10 CFR 835 and recommendations of the RadCon Standard, as elaborated in the *Implementation Guide for Internal Dosimetry Programs*.

The standard will be reviewed and updated by DOE when necessary. Technical advances in internal dose assessment may allow strengthening of the performance specifications. Additional improvements may be made to the standard as experience is gained through its use or application.

1.4 OVERVIEW

Internal dosimetry is a major component of nuclear safety for the approximately 100,000 radiation workers at DOE radiological or nuclear facilities. Workers who handle nuclear materials or who are involved in nuclear waste management are potentially at risk of inadvertent intakes of radioactive material. DOE policy and associated radiological control programs for limiting internal effective dose equivalents are based on containment of radioactive material to ensure (to the extent reasonably achievable) that radionuclides from work at radiological or nuclear facilities are not taken into the body. Most significant occupational intakes of radionuclides occur as the result of contamination incidents associated with either the inadvertent release of radioactive material in the workplace or the unplanned loss of containment.

DOE's 10 CFR 835 requires monitoring of employees likely to receive intakes of radionuclides that would result in committed effective dose equivalents at or above 100 mrem in a year. Monitoring programs in the workplace are designed to demonstrate that the requirements to limit exposure to 5 rems committed effective dose equivalent ($H_{E,50}$) in any year are being met. Radiation worker bioassay monitoring programs are designed to provide the data needed to assess organ and tissue dose equivalents from intakes of radioactive material. If exposures to radioactive materials are such that significant internal doses are received from intakes occurring during the year, they are most often assessed using biokinetic models.

In 1986, efforts were begun to develop a technically-based manual that would provide guidance on developing and operating internal dosimetry programs at DOE radiological or nuclear facilities that wold meet all applicable regulatory requirements. Input from internal dosimetry experts from DOE and various DOE contractors has been collected for well over a decade. This document, which resulted from that effort, attempts to assemble in one place information that will assist in meeting the requirements for conducting a internal dosimetry program within the DOE complex.

The intent of this guidance document is to provide a fairly complete, though not exhaustive, set of basic procedural guidelines for achieving minimum levels of acceptable performance in evaluating the internal radiation dose equivalent that may be received by radiation workers from intakes of radionuclides. The guidance provided here represents the collective wisdom of a diverse group with experience in internal dosimetry at DOE facilities. There has been a conscious effort to include examples from this group on the application of these guidance principles in the standard operations of their administered internal dosimetry programs.

Section 2 provides the definitions and abbreviations that are commonly used in the field of internal dosimetry.

Descriptions of documents and plans needed for an internal dosimetry program are provided in Section 3. These include internal dosimetry technical basis documentation, an internal dosimetry

procedures manual, a bioassay contingency plan for facilities having no routine monitoring program, a dose management practices plan, an action plan for medical response, and a quality assurance plan.

Section 4 provides guidance on the design of an individual monitoring program. It gives specific information on the investigation level *(IL)*, the derived investigation level *(DIL)*, methods of measurement, frequency of bioassay measurement, supplementing routine bioassay programs (where the *DIL* < the *MDA*), and performance specifications for a bioassay or service laboratory.

The different monitoring regimens of an individual monitoring program are discussed in Section 5. These include a baseline bioassay used prior to starting radiological work, routine bioassay monitoring conducted when workers are likely to receive 100 mrems committed effective dose equivalent in the workplace, special bioassay monitoring conducted following incidents with potential for intake, and bioassay monitoring conducted prior to termination of employment or end of potential for intake.

Section 6 contains the methods used to detect and confirm intakes of radioactive materials. The section explains the use of either bioassay data or workplace monitoring data to confirm an intake. Historically, workplace airborne radioactivity monitoring systems were put in place to detect inadvertent loss of containment. They were not intended to provide data for evaluating intakes by workers from exposures to airborne contamination. Thus, air monitors were located in areas with the highest potential for detecting loss of containment rather than in those areas most commonly occupied by radiation workers. Air monitoring data have not routinely been used to assess internal dose equivalent because of the poor correlation between concentration of radionuclides in the air sampled by monitoring equipment and the actual amount of radioactive material inhaled by workers. While bioassay monitoring data are used almost exclusively in internal dosimetry programs, there may be instances where workplace air monitoring data may be used to assess internal dose.

Following the confirmation of an intake of radioactive material, an evaluation of the resultant internal dose is necessary. A discussion of the calculation of internal dose from bioassay data, and recommendations on interpretation of the bioassay data and handling of statistical uncertainties are presented in Section 7.

Section 8 covers management of total effective dose equivalent (TEDE) and cumulative TEDE or lifetime occupational dose. Topics of discussion include routine occupational worker dose management, management of dose from previous intakes (work restrictions), compliance with internal dose monitoring requirements, control of dose to the embryo/fetus, minors, and students, and interface with external dosimetry. Guidance is provided on using and recording total effective dose equivalent, lifetime dose control, doses due to intakes prior to January 1, 1989, and statistical uncertainties. Also discussed are elements of an accidental dose control program, including incident dose management, preparation for incidents involving intakes, and internal dose control after an incident.

Section 9 presents a discussion of recommendations for recording and reporting internal doses. Guidance is provided on a general philosophy of records and record keeping, reporting of preliminary assessments of unplanned exposures, precision of internal dose assessments, long-term reevaluation of intakes, practical reporting of internal doses, minimum recordable doses, recording of significant organ and tissue doses, cumulative TEDE, and records associated with bioassay measurements and their interpretation.

Section 10 includes a recommended scheme for medical response following a potential intake of radioactive material. Guidance is provided on when and how to treat patients as well as the role of a

health physicist as an interface to medical treatment. The impact of therapeutic measures on the outcome of dosimetric evaluations is also discussed.

Quality assurance issues associated with bioassay measurements, evaluations of intake, and internal dose are presented in Section 11.

1.5 USE OF NON-GOVERNMENTAL STANDARDS

To the extent possible, this guidance document is written to be consistent with existing non-governmental standards for internal dosimetry, including:

- ANSI N42.22-1999, *American National Standard–Traceability of Radioactive Sources to the National Institute of Standards and Technology (NIST) and Associated Instrument Quality Control*
- ANSI N42.23-1996, *Measurement and Associated Instrumentation Quality Assurance for Radioassay Laboratories*
- ANSI N323a-1997, *American National Standard for Radiation Protection Instrumentation Test and Calibration*
- ANSI Z88.2-1992, *American National Standard for Respiratory Protection*.
- HPS N13.1-1999, *Guide to Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities*
- HPS N13.6-1999, *Practice for Occupational Radiation Exposure Records Systems*
- HPS N13.12-1999, *Surface and Volume Radioactivity Standards for Clearance*
- HPS N13.14-1994, *Internal Dosimetry Programs for Tritium Exposure, Minimum Requirements*
- HPS N13.22-1995, *Bioassay Programs for Uranium*
- HPS N13.30-1996, *Performance Criteria for Radiobioassay*
- HPS N13.39-2000, *Design of Internal Dosimetry Programs*
- HPS N13.42-1997, *Internal Dosimetry for Mixed Fission and Activation Products*

2.0 DEFINITIONS AND ABBREVIATIONS

The definitions below come from many sources, indicated in the definition itself, and many have been adopted from the compilation by (Traub 1994). In this section, RadCon Standard refers to the U.S. Department of Energy *Radiological Control Standard* (DOE 1999e); 10 CFR 835 refers to the DOE rule *Occupational Radiation Protection* (DOE 1998a); AMG refers to the DOE *Air Monitoring Guide* (DOE 1999d); and IDG refers to the DOE *Internal Dosimetry Program Guide* (DOE 1999b). Other definitions come from other DOE documents, national and international standards and recommendations; some definitions are new. Terms in italics are defined elsewhere in the definitions section.

2.1 DEFINITION CROSS-REFERENCE

Most of the terms commonly used in the field of internal dosimetry have been adequately defined in documents that are commonly available at DOE sites and facilities. Rather than repeat the majority of these definitions here, Table I cross-references these definitions to other documents. Where a definition is found to have more than one source, the definition that occurs in 10 CFR 835 (when applicable) should be taken as the official definition for that term. Definitions are given in Section 2.3, when they are not given in 10 CFR 835, the RadCon Standard, IDG, or AMG, or when it is useful to present additional clarifying information. In Table I below, italicized items are used as symbols for the quantity elsewhere in this standard.

Table I. Cross-Reference of Internal Dosimetry Terms

Term	10 CFR 835	RadCon Standard	IDG	AMG	Other*
activity median aerodynamic diameter (*AMAD*)			X		ICRP-66 (1994b)
activity median thermodynamic diameter (*AMTD*)					ICRP-66 (1994b)
administrative control level		X	X		
airborne radioactive material (or airborne radioactivity)	X	X			
airborne radioactivity area	X	X			
air monitoring				X	
air sampling				X	
ALARA Committee		X			
alpha (α) (as a probability)			X		
analyte			X		
annual limit on exposure (*ALE*)					ICRP-32 (1981)

Term	10 CFR 835	RadCon Standard	IDG	AMG	Other*
annual limit on intake (*ALI*)	X	X			ICRP-32 (1981) for ^{222}Rn and ^{220}Rn progeny
appropriate blank					HPS N13.30-1996
assessment		X			
assigned protection factor (*APF*)					ANSI Z88.2-1992
As Low As Reasonably Achievable (ALARA)	X	X			
background	X				synonymous with background radiation; HPS N13.30-1996
background radiation	X	X			
baseline bioassay			X		
becquerel (Bq)		X			
beta (β) (as a probability)			X		
bias					HPS N13.30-1996
bioassay	X	X			synonymous with radiobioassay
breathing zone air monitoring				X	
calibration	X	X			
company-issued clothing		X			
confirmed intake			X		
containment device		X			
contamination area	X	X			
continuous air monitor (CAM)		X		X	see “real time air monitoring”
contractor	X				DOELAP
controlled area	X	X			
conventionally true value of a quantity					HPS N13.30-1996
counseling		X			
critical mass		X			
critique		X			

Term	10 CFR 835	RadCon Standard	IDG	AMG	Other*
decision level (*DL*, L_c)			X		
declared pregnant worker	X	X			
decontamination		X			
derived air concentration (*DAC*)	X	X			
derived air concentration-hour (*DAC*-h)	X	X			
derived investigation level (*DIL*)			X		
deterministic effects					synonymous with nonstochastic effects
diagnostic examinations					HPS N13.30-1996
diagnostic measurment					HPS N13.30-1996
direct (in vivo) radiobioassay			X		HPS N13.30-1996
disintegration per minute (dpm)		X			
DOE activity	X	X			
DOELAP		X	X		
dose	X	X			
absorbed dose (*D*)	X	X			
collective dose		X			
committed dose equivalent ($H_{T,50}$)	X	X			
committed effective dose equivalent ($H_{E,50}$)	X	X			
cumulative total effective dose equivalent	X	X			
deep dose equivalent	X	X			
dose equivalent (*H*)	X	X			
effective dose equivalent (H_E)	X	X			
external dose or exposure	X	X			
internal dose or exposure	X	X			
lens of the eye dose equivalent	X	X			
quality factor	X	X			

Term	10 CFR 835	RadCon Standard	IDG	AMG	Other*
shallow dose equivalent	X	X			
total effective dose equivalent (TEDE)	X	X			
weighting factor (w_T)	X	X			
whole body	X	X			
dose assessment		X			
elimination			X		
embryo/fetus		X			
engineering controls		X			
equilibrium factor (F)					ICRP-32 (1981)
equilibrium equivalent concentration (EEC)					ICRP-32 (1981)
evaluation			X		
excretion			X		
exposure			X		ICRP-65 (1993a)
facility		X			
false negative			X		
false positive			X		
fixed contamination		X			
frisk or frisking		X			
gastrointestinal (GI) tract model			X		
general employee	X	X			
gestation period		X			
gray (Gy)		X			
high contamination area	X	X			
high radiation area	X	X			
hot particle		X			
hot spot		X			
indirect (in vitro) bioassay			X		
indirect radiobioassay					HPS N13.30-1996

Term	10 CFR 835	RadCon Standard	IDG	AMG	Other*
individual	X				
infrequent or first-time activities		X			
intake			X		
investigation level (*IL*)			X		
in vitro measurement					HPS N13.30-1996
in vivo measurement					HPS N13.30-1996
lifetime dose		X			
lifetime occupational dose					synonymous with lifetime dose
lower limit on detection					synonymous with *MDA*; HPS N13.30-1996
member of the public	X	X			
minimum detectable amount (*MDA*)			X		HPS N13.30-1996
minimum detectable concentration (*MDC*)					HPS N13.30-1996
minimum testing level (*MTL*)					HPS N13.30-1996
minor	X	X			
monitoring	X	X			
nonstochastic effects	X				
occupational dose	X	X			
person	X				
personal air monitoring				X	
personal protective equipment		X			
personnel dosimeters		X			
personnel monitoring		X			
planned special exposure		X			
potential alpha energy concentration (*PAEC*)					ICRP-32 (1981)
potential alpha energy exposure (*PAEE*)					ICRP-32 (1981)
prenatal radiation exposure		X			

Term	10 CFR 835	RadCon Standard	IDG	AMG	Other*
protective clothing		X			
qualification standard		X			
quality assurance					HPS N13.30-1996
quality control					HPS N13.30-1996
rad		X			
radiation	X	X			
radiation area	X	X			
radioactive material		X			
radioactive material area	X	X			
radioactivity		X			
radiobioassay					HPS N13.30-1996
radiological area	X	X			
radiological buffer area (RBA)		X			
radiological control hold point		X			
radiological work		X			
radiological work permit		X			
radiological worker(s)	X	X			
radon			X		ICRP-32 (1981)
real time air monitoring	X	X		X	replacement for “continuous air monitoring”
Reference Man			X		ICRP-23 (1975)
rem		X			
removable contamination		X			
respiratory protective equipment or device	X	X			
retention			X		
routine radiobioassay monitoring			X		
sievert (Sv)		X			
site		X			

Term	10 CFR 835	RadCon Standard	IDG	AMG	Other*
screening measurements					HPS N13.30-1996
service laboratory					HPS N13.30-1996
sealed radioactive source	X	X			
special radiobioassay monitoring			X		
state-of-the-art			X		
step-off pad		X			
sticky pad		X			
stochastic effects	X				
survey		X			
technology shortfall			X		
termination bioassay			X		
thermodynamic particle diameter (d_{th})					ICRP-66 (1994b)
thoron		X			ICRP-32 (1981)
Type I error			X		
Type II error			X		
very high radiation area	X	X			
visitor		X			
week	X	X			
whole body dose		X			
working level (WL)	X				See 10 CFR 835 App A Footnote 4
year	X	X			

* Definitions whose source is other than 10 CFR 835, the RadCon Standard, IDG, AMG, or DOELAP are presented in Section 2.3.

2.2 RADON AND THORON

The chemical element radon has two radiologically important isotopes that occur in nature: ^{220}Rn and ^{222}Rn. Following popular usage, this document refers to the former as "thoron" and the latter as "radon."

Radon and its short-lived progeny (decay products) are continuously produced by decay of ^{226}Ra, a member of the naturally occurring ^{238}U series. Airborne concentrations of radon's short-lived progeny

(^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po) are of interest due to their potential for deposition in the lung, leading to subsequent irradiation of lung tissue by alpha emissions from ^{218}Po and ^{214}Po.

Thoron and its short-lived progeny are continuously produced by the decay of ^{224}Ra, a member of the naturally occurring ^{232}Th series. Thoron and ^{216}Po have short half-lives: 56 s and 0.145 s, respectively. Lead-212 and ^{212}Bi are of interest due to the possibility of their being deposited in the lung and irradiating tissue with alpha emissions.

2.3 SPECIFIC DEFINITIONS

activity median thermodynamic diameter (*AMTD*): "Fifty percent of the activity (thermodynamically classified) in the aerosol is associated with particles of *thermodynamic diameter* (d_{th}) greater than the *AMTD*. A lognormal distribution of particle sizes is usually assumed." (ICRP 1994a)

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.*

appropriate blank: A sample, person, or phantom that is, ideally, identical in physicochemically and radiologically significant ways with the sample, person, or phantom to be analyzed. (HPS N13.30-1996)

assess: For purposes of this Standard, to officially assign or record a dose number.

assigned protection factor (*APF*): The expected workplace level of respiratory protection that would be provided by a properly functioning respirator or a class of respirators to properly fitted and trained users. (ANSI Z88.2-1992)

bias: The deviation of a single measured value of a random variable from a corresponding expected value, or a fixed mean deviation from the expected value that remains constant over replicated measurements within the statistical precision of the measurement (synonymous with deterministic error, fixed error, and systematic error). (HPS N13.30-1996)

bioassay: Another word for radiobioassay.

biokinetic model: A series of often empirically determined mathematical relationships formulated to describe the intake, deposition in respiratory tract (if applicable), uptakes by the transfer compartment from intake compartment(s), uptakes by tissues or organs from the transfer compartment, translocation, retention, and elimination of a radionuclide from the body.

censored data: Data that have been recorded as "less than" values rather than the observed numerical values (whether positive, zero, or negative).

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, $H_{E,50} = \Sigma w_T H_{T,50}$. 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 rem (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 rem (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 ($w_T = 0.12$).*

compartment: The smallest element in a biokinetic model for which a mathematical representation of a retained quantity is given. Compartments may be organs (e.g., lung, liver), tissues (e.g., bone marrow), or systemic (e.g., the transfer compartment).

critical level: Same as *decision level*.

compartment: The smallest element in a biokinetic model for which a mathematical representation of a retained quantity is given. Compartments may be organs (e.g., lung, liver), tissues (e.g., bone marrow), or systemic (e.g., the transfer compartment).

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: *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 CEDE 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)

decision level: The amount of a count (L_C) or a count rate (L_C') or the final instrument measurement of a quantity of analyte (D_C or D_C') at or above which a decision is made that the analyte is definitely present. (HPS N13.30-1996)

deposition fraction: The fraction of the amount of a material inhaled that is deposited in a particular region of the respiratory tract. For an aerosol, this fraction is a function of the aerodynamic or thermodynamic diameter.

detection level (L_D): This concept has been replaced by minimum detectable amount (*MDA*).

diagnostic measurements: Measurements performed to estimate the amount of radionuclide deposited in a person when an intake is known or is suspected to have occurred. (HPS N13.30-1996)

direct radiobioassay: The measurements of radioactive material in the human body utilizing instrumentation that detects radiation emitted from the radioactive material in the body (synonymous with in vivo measurement.). (HPS N13.30-1996)

equilibrium factor (*F*): The equilibrium factor *F* with respect to potential alpha energy is the ratio of the *equilibrium equivalent concentration* (*EEC*) to the actual activity concentration of radon in air.

equilibrium equivalent concentration (*EEC*): The *EEC* of a non-equilibrium mixture of short-lived radon progeny is that activity concentration of radon in radioactive equilibrium with its short-lived progeny that has the same potential alpha energy concentration as the non-equilibrium mixture to which the *EEC* refers.

exposure: (1) The general condition of being subjected to radiation, such as by exposure to radiation from external sources or to radiation sources inside the body. In this document, exposure does not refer to the radiological physics concept of charge liberated per unit mass of air. (IDG)

(2) The product of exposure time to a radioactive aerosol and the average concentration during exposure, divided by the value of the *DAC* for the radioactive material in question (expressed in *DAC*-h).

(3) Exposure (of an individual to radon progeny) is the time integral of the potential alpha energy concentration in air over a given period (expressed in WLM) (adapted from ICRP Publication 65, p.4).

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

indirect radiobioassay: Measurements to determine the presence of or to estimate the amount of radioactive material in the excreta or in other biological materials removed from the body (synonymous with *in vitro measurement*.) (HPS N13.30-1996)

intake compartment: One of four compartments from which systemic uptake can occur: the respiratory tract; the GI tract; a wound; or intact skin.

intake route: A pathway by which radioactive material enters the body. The main intake routes are inhalation, ingestion, absorption through the skin, and entry through injection or a cut or wound in the skin.

in vitro measurement: Synonymous with *indirect bioassay*.

in vivo measurement: Synonymous with *direct bioassay*.

lower limit of detection (*LLD*): Synonymous with *minimum detectable amount (MDA*).

minimum detectable amount (*MDA*): The smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability ß of non-detection *(Type II error)* while accepting a probability α of erroneously deciding that a positive (non-zero) quantity of analyte is present in an appropriate blank sample *(Type I error)*.

minimum detectable concentration (*MDC*): The *minimum detectable amount (MDA)* expressed in units of concentration. (HPS N13.30-1996)

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

minimum testing level (*MTL*): The amount of radioactive material that the service laboratory should be able to measure for participation in the performance testing program, assuming the samples are free of interference from other radionuclides unless specifically addressed. The *MTL*s should not be construed as being the appropriate *MDA* required for a specific internal dosimetry program, but rather an acceptable minimum testing level for radiobioassay service laboratories based on good measurement practice. (HPS N13.30-1996)

potential alpha energy concentration (*PAEC*): The kinetic energy potentially released in a unit volume of air by alpha particles emitted by the short-lived radioactive progeny of ^{222}Rn (i.e., ^{218}Po and ^{214}Po) or ^{220}Rn (i.e., ^{216}Po, ^{212}Bi, and ^{212}Po). *PAEC* is expressed in working levels (WL).

potential alpha energy exposure (*PAEE*): The average potential alpha energy concentration (*PAEC*) to which a worker is exposed, multiplied by the time of exposure in working months of 170 hours: that is, $PAEE = PAEC \times time$. *PAEE* is expressed in working level months (WLM).

quality assurance: All those planned and systematic actions necessary to provide adequate confidence that an analysis, measurement, or surveillance program will perform satisfactorily in service. (HPS N13.30-1996)

quality control: Those actions that control the attributes of the analytical process, standards, reagents, measurement equipment, components, system, or facility according to predetermined quality requirements. (HPS N13.30-1996)

radiobioassay: Measurement of amount or concentration of radioactive material in the body or in biological material excreted or removed from the body and analyzed for purposes of estimating the quantity of radioactive material in the body. (HPS N13.30-1996)

radon: For purposes of this DOE Standard, unless otherwise specified, the isotope ^{222}Rn.

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

retained quantity: The amount of material which, after being taken into the body by inhalation, ingestion, entry through an open wound, or absorption through the skin, exists in the whole body, a compartment, an organ, or a tissue at a specified time.

screening measurements: Measurements made to detect radioactive material under routine conditions, but not used to quantify the amount of a given radionuclide. (HPS N13.30-1996)

service laboratory: Laboratory performing direct and/or indirect radiobioassay measurements. (HPS N13.30-1996)

thermodynamic particle diameter (d_{th}): Diameter (in μm) of a spherical particle that has the same diffusion coefficient in air as the particle of interest. (ICRP 1994a)

thoron: The isotope ^{220}Rn, also symbolized by Tn. Thoron is a "trivial name" like tritium.

translocation: Movement within the body of a radioactive material after uptake, such as from bone to kidney.

working level (WL): is any combination of the short-lived radioactive progeny in one liter of air, without regard to the degree of equilibrium, that will result in the ultimate emission of 130,000 MeV of alpha energy (1 WL = 2.083 E-5 J/m^3). (10 CFR 835)

Note: WL is the unit of *potential alpha energy concentration (PAEC)*

working level month (WLM): The unit of *potential alpha energy exposure (PAEE)*, defined as exposure for 1 working month (of 170 hours) to an airborne concentration of 1 WL. (1 WLM = 1 WL × 170 hours = 0.00354 J·h/m^3).

wound compartment: The compartment in a biokinetic model whose retained quantity is the amount of radioactive material in a wound that has not moved to the transfer compartment.

2.4 ABBREVIATIONS, ACRONYMS, CODES, INITIALISMS, AND SYMBOLS

α	alpha
β	beta
ΔA_{min}	detection sensitivity
μ	prefix micro (10^{-6})
ACGIH	American Conference of Governmental Industrial Hygienists
AEDE	annual effective dose equivalent
ALARA	as low as reasonably achievable
ALE	annual limit on exposure
ALI	annual limit on intake
AMAD	activity median aerodynamic diameter
AMD	acceptable missed dose

AMTD	activity median thermodynamic diameter
ANSI	American National Standards Institute
APF	assigned protection factor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BEIR	Biological Effects of Ionizing Radiation
BZ	breathing zone
CAM	continuous air monitor
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
C_u	observed concentration of analyte in urine
D	absorbed dose
DAC	derived air concentration
DIL	derived investigation level
DL	decision level
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegration per minute
DRL	derived reference level
d_{th}	thermodynamic diameter
DTPA	diethylene-triamine-pentaacetic acid
E	exposure
ED	effective dose
EDTA	*ethylene-diamine-tetraacetic* acid
EEC	equilibrium equivalent concentration
EEI	equilibrium equivalent intake
EMSL	Environmental Measurements Standards Laboratory
EPA	U.S. Environmental Protection Agency
f	sampling frequency
F	equilibrium factor
FEMP	Fernald Environmental Management Project
f_p	unattached fraction
FR	Federal Register
GA	general area
GI	gastrointestinal
GM	Geiger-Müller
GSD	geometric standard deviation
Gy	gray
H_E	effective dose equivalent
$h_{E,50}$	committed effective dose equivalent per unit of activity
$H_{E,50}$	committed effective dose equivalent
H_T	tissue dose equivalent
$H_{T,50}$	committed dose equivalent
HPS	Health Physics Society
IDG	Internal Dosimetry Program Guide
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronic Engineers
IL	investigation level
IL_s	sample investigation level
IRF	intake retention fraction

IRF_u	intake retention function for urinary excretion
ISO	International Organization for Standardization
L_C	decision level (formerly the critical level)
L_D	detection level (use *MDA*)
L_{MR}	medical referral level (L_{DMR} is the derived medical referral level)
L_V	verification level (L_{DV} is the derived verification level)
LLD	lower limit of detection
MDA	minimum detectable amount (or activity)
MDC	minimum detectable concentration
MLE	maximum likelihood estimator
MPBB	maximum permissible body burden
MTL	minimum testing level
MWA	maximum working activity
N	all other modifying factors; age of worker in years
NALI	nonstochastic annual limit on intake
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute of Occupational Safety and Health
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
N_V	the number of transitions per unit volume
OSHA	Occupational Safety and Health Administration
PAEC	potential alpha energy concentration
PAEE	potential alpha energy exposure
PCs	protective clothing
Q	quality factor
QA	quality assurance
QAP	Quality Assurance Plan
QC	quality control
RBA	radiological buffer area
RCT	radiological control technician
RSO	radiation safety officer
RWP	radiological work permit
SA	specific activity
SALI	stochastic annual limit on intake
SI	International System (of units)
t	time
t_0, t_1, t_2	particular values of time
t_E	exposure time (d)
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
Tn	thoron (^{220}Rn)
V&V	verification and validation
V_u	urine excretion rate
UMTRA	Uranium Mill Tailings Remedial Action
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
WL	working level
WLM	working level month
w_T	tissue weighting factor

2.5 CONVENTIONS FOR ROUNDING AND SIGNIFICANT FIGURES

The need to distinguish between *exact* numbers, such as dose limits, and *imprecise* numbers, such as the results of measurements, leads to the use of some conventions in this DOE Standard.

Any legislated number, as well as any integer or ratio of two integers, is an exact number with no uncertainty[1]. Examples of exact numbers are the 5-rem annual TEDE limit, the *DAC* for radon progeny of 1/3 WL, and integral fractions and exponents (e.g., kinetic energy = 1/2 mv^2). Exact numbers may have tolerances, but when tolerances are not specified, the exact numbers must be treated as arbitrarily precise: a 5-rem limit is 5.000 000 000 rems.

Measurements are often uncertain and imprecise, and inferences of dose from measurements using calculational models with uncertain parameters are also uncertain and imprecise. Confusion sometimes results when comparing uncertain or imprecise numbers with exact standards. Furthermore, difficulty arises when exact numbers are converted from one set of units to another and the result is rounded. This difficulty becomes particularly acute for quantities and units associated with radon and thoron. Thus, the DOE has decided to derive all radon and thoron concentration values from 10 CFR 835, Appendix A, *PAEC* limits (or ICRP/IAEA *PAEE* limits for newer recommendations), rather than from 10 CFR 835, Appendix A, equilibrium equivalent *DAC*s, which give slightly different answers and lead to confusion (Strom et al. 1996).

Excellent, detailed guidance on significant figures and rounding for measurements is given by the ASTM (ASTM 1993). Unfortunately, ASTM E380-93 does not recognize the exact nature of regulatory limits nor address the problems of significant figures when converting exact numbers between unit systems. Also, it does not address radon and thoron quantities and units.

To minimize roundoff errors, it is recommended that all calculations be performed using numbers specified to at least "single precision" (six to seven significant figures) or as rational numbers if appropriate (ratios of integers, e.g., 1/3, 5/12, etc.) For reporting purposes, it should be acceptable to round to three significant figures or to the precision of the reporting field, whichever is less. For example, 3.84 mrem may be rounded to 4 mrem if only integral numbers of mrem can be reported in a given field. More detail on recording and reporting is given in Section 9.

Similarly, in this DOE Standard, all numbers that are simply unit conversions are expressed to 5 significant figures to prevent contradictions or inconsistencies.

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[1]Some irrational numbers are exact, such as π, $\sqrt{2}$, and e, the base of the natural logarithms. Sometimes, unit conversions are exact, such as 37,000,000,000 Bq/Ci.

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3.0 DOCUMENTS AND PLANS LISTED BY THE IMPLEMENTATION GUIDE

This section provides suggested contents for the various documents listed in the IDG and the RadCon Standard to provide technical guidance for implementing internal dosimetry programs. Possible organizational schemes for documents are presented in Example 3.1. The documents do not need to be distinct provided that all topics are covered. Other alternatives are acceptable.

Example 3.1. Organization of Documents at Hanford

The documentation of the internal dosimetry program at the Hanford Site is incorporated into several different documents.

- Technical Basis Manual - includes technical methods, supporting evidence, and reference information used to provide the technical foundation for the Internal Dosimetry Program
- Program Manual - includes a guide to the services and capabilities provided by the Internal Dosimetry Program including policies, recommendations for good practice, and general guidance to contractor dosimetry organizations
- Procedures Manual - includes procedures for the day-to-day operations of the Internal Dosimetry Program including records management, communications, data review, and exposure evaluation documentation.
- Incident Response Plan - incorporated as an appendix in the Program Manual.

Internal dosimetry documents and plans must be rooted in the requirements of 10 CFR 835 and other contractual requirements, for example the RadCon Standard. They should draw guidance from the IDG, this technical standard, and applicable non-government standards and draft standards, such as those listed in Section 1.5.

3.1 INTERNAL DOSIMETRY TECHNICAL BASIS DOCUMENTATION

This section summarizes all of the topics that appear throughout the IDG and the RadCon Standard (DOE 1999e) for the Internal Dosimetry Technical Basis Documentation. The RadCon Standard recommends the development of Internal Dosimetry Technical Basis Documentation that gives scientific information and other rationale explaining each element of the internal dosimetry program to support dose evaluation methods used therein.

The following information is suggested for inclusion in various sections of the internal dosimetry technical basis documentation:

3.1.1 Organization and Agreements

- letter(s) of agreement between contractors at a multiple-contractor site detailing the responsibilities, authority, and communication needs of the respective parties

- listing of arrangements between the internal dosimetry program and the bioassay measurements laboratory, including

  -needed turnaround times

  -*MDA*s for special and routine samples

  -priorities for classification of samples (e.g., routine, special, emergency)

3.1.2 Bioassay Program Design

- physical and chemical characteristics of radioactive materials encountered in the workplace
- establishment of the type and frequency of measurements to be used (RadCon Standard 522.3)
- derivation of decision levels
- default trigger levels
- preliminary actions to be taken for exposures to the different radionuclides present at a facility following suspected or confirmed intakes at various levels
- tailoring of investigations to a specific individual worker or exposure circumstances
- documentation of the derivation of *DIL*s
- established *DIL*s for each bioassay method applied for the analysis of all radionuclides to which workers are likely to be exposed
- if it is known or is likely that an individual has or could have intakes during the year from different sources that could result in doses above the *IL*, methods to use to derive an appropriately smaller *DIL*s
- methods of bioassay measurement and the rationale or justification for each
- the *MDA*s for the bioassays
- justification for the bioassay monitoring frequencies, including an evaluation of the largest internal dose (i.e., minimum detectable dose) from an intake (acute or chronic) that could go undetected with the chosen frequency
- documentation and justification of a planned supplementary approach for intake or dose assessment in case of technology shortfall
- the rationale for the formal action procedures following a bioassay result unexpectedly above the DL.

3.1.3 Participation in Bioassay Program

- rationale for selection of workers for bioassay monitoring

3.1.4 Detection and Confirmation of Intakes

- biokinetic models
- model parameters
- assumptions
- justification of the choices of default parameters used in deriving a *DIL*
- parameters and their associated default values used in dosimetric modeling and evaluation, such as

  -intake date

  -deposition probabilities (deposition fractions)

  -retention functions

  -organ masses

  -absorption fractions

  -facility-specific factors
- statistical methods for

  -evaluating bioassay data

  -identifying bioassay results above environmental background values

  -using appropriate blanks

  -analyzing trends

  -*MDA*s
- description of a procedure for evaluating doses if the time course of an intake cannot be plausibly established
- if *DAC*-h calculations are used to assess exposures to airborne radioactive materials, a description of any authorized adjustment(s) to such calculations to account for the use of respiratory protection

3.1.5 Internal Dose Evaluation

- method for evaluating internal doses from routine and special bioassay data, and where appropriate, from workplace monitoring data, including personal air samplers
- methods for calculating internal doses
- methods for evaluating dose equivalents from specific radionuclides, mixtures of radionuclides, and materials of differing chemical characteristics
- basis for the evaluation methods including recommendations given in ICRP Publications and NCRP Reports, which embody improvements and updates of the science of internal dosimetry
- justification for alternative approaches and assumptions used in dose calculations
- dose evaluation quality assurance
- biokinetic models
- model parameters
- assumptions
- individual-specific and facility-specific factors that are expected to change the dose calculations by a factor of 1.5 or more
- a description of the level of intake or committed effective dose equivalent detection achieved
- a basis for projecting a CEDE of one *IL* from bioassay results

3.1.6 Internal Dose Management

- action levels for administrative response to intakes of radionuclides by workers, including decisions reached among medical, management, and radiation protection staff (RadCon Standard 523.6)
- a description of the site policy for confirming intakes in instances of historical bioassay data prior to January 1, 1989, where follow-up bioassay samples were not required on positive bioassay samples or where documentation is lacking (counter efficiency, chemical recovery, minimum detectable amount/activity, etc.)
- methodology to account for the portion of a bioassay result that may be due to one or more prior confirmed intakes
- basis for work restrictions used during internal dose evaluation
- administrative controls to limit dose to declared pregnant workers, minors, and students

- description of the interface with external dosimetry
- methods for calculating TEDE
- determination of lifetime dose and specification of lifetime dose administrative control levels

3.1.7 Records and Reporting

- methods for documenting calculations
- recording and reporting practices for internal dosimetry
- a description of the configuration control of the internal dosimetry technical basis documentation, including
  - specific maintenance of the internal dosimetry technical basis documentation, including responsibilities for authorship, review, approval, and distribution
  - maintenance as a controlled document
  - periodic review of internal dosimetry technical basis documentation by the site radiation protection organization to ensure that the scientific bases are current and that the technical basis appropriately reflects changes in existing standards, anticipated changes, and new standards
  - external peer-review by qualified individuals on a periodic basis
  - retention as a radiological protection program record with copies of all previous revisions and changes retained for future program review

3.1.8 Medical Response

- description of accidental dose control methods

3.1.9 Monitoring the Workplace

- specification of continuous air monitor (CAM) alarm levels and justification of the levels chosen

3.2 INTERNAL DOSIMETRY PROCEDURES MANUAL

Written policies and procedures covering each step in the activities used to determine worker internal dose are an essential element of an acceptable internal dosimetry program. All elements of the internal dosimetry program should be specified in written procedures. These procedures should be consistent with 10 CFR 835, the RadCon Standard, the IDG, relevant DOE Orders, this document, and the internal dosimetry technical basis documentation. The internal dosimetry procedures should specify or identify the following:

- methods and requirements for measurement (bioassay) and evaluating and recording internal dose
- methods for consistent collection of workplace and personnel monitoring data, its evaluation, documentation of results, and records maintenance
- the components of the internal dosimetry program and the organizational structure to which it reports
- responsibilities of line management and members of the dose evaluation group
- elements of the workplace and radiological worker monitoring programs that are germane to internal dosimetry
- guidelines for prompt follow-up of worker intakes of radioactive materials, and appropriate follow-up response to intakes, including the medical management of workers with excessive intakes
- all relevant subcontractor procedures to be included in the historical record files of the DOE contractor
- the *MDA*s of the various bioassay measurement methods
- programmatic details, including:
  - method(s) of bioassay measurements (e.g., urinalysis, fecal analysis, or in vivo counting)
  - analytical methodology (e.g., chemical separation followed by alpha counting)
  - measurement parameters (e.g., counting time or instrument efficiency) to be used in each component of the bioassay program
- frequency of the routine bioassay program
- agreements with the bioassay measurements laboratory on needed turnaround times, *MDA*s for special and routine samples, and priorities for classification of samples (e.g., routine, special, emergency)
- factors to be considered by the internal dosimetry staff in determining the follow-up or confirmatory actions to be taken in response to positive bioassay results
- actions taken following a bioassay result unexpectedly above the DL
- personnel who will establish confirmatory bioassay requirements in cases not covered by the procedures

- trigger levels and preliminary actions to be taken for exposures to the different radionuclides encountered at the facility
- other methods that may be used for the evaluation of doses from intakes and their scientific basis
- action levels for administrative response to intakes of radionuclides by workers
- records to document the appropriateness, quality, and accuracy of monitoring methods, techniques, and procedures in use during any given period, pursuant to applicable requirements and standards
- documentation that all steps in the activities that control or evaluate worker internal doses by written procedures provide appropriate quality control and quality assurance

Radiochemical laboratories and in-vivo counting facilities whose measurements are used by internal dosimetry programs are expected to have written procedures that can be referenced by internal dosimetry programs.

The internal dosimetry program should receive periodic assessment by the site radiation protection organization to review dose assessment procedures as necessary to ensure that the program maintains the capability to stay abreast of scientific developments in internal dosimetry and provides a quality radiation protection service to workers. Paragraph 10 CFR 835.102 requires that an internal audit be done every 36 months. External peer-review by qualified individuals on a periodic basis is also recommended.

The procedures should be reviewed at least once every two years and updated as necessary. The needs for maintenance of procedures should be specified, including responsibilities for authorship, review, approval, and distribution.

3.2.1 Bioassay Contingency Plans

Some facilities with low potential for significant occupational intake of radioactivity may not have any routine bioassay program. Examples of such facilities are those where only sealed sources are handled, or the types, quantities, and frequency of dealing with radioactive materials does not support establishment of routine capability from a cost-effectiveness viewpoint. However, if quantities of unsealed radioactive material are handled infrequently or if accidents could happen causing intakes corresponding to 100 mrem CEDE, then it may be wise to have a contingency plan for obtaining bioassay measurements. Elaborate advance arrangements are not necessarily warranted. However, thought should be given to what types of bioassay measurements might be needed, and how and where they would be obtained. A good approach would be to identify the closest DOE facility with capability appropriate for the radionuclides and to have a letter of agreement or memorandum of understanding in place to obtain measurements on an as-needed basis. As a minimum, the radiation protection organization should know whom to contact for support, how long until data could be obtained, and what to do until data would become available.

A contingency plan for sites having routine bioassay is worth considering because of the possibility of losing one or more components of a bioassay program. Such loss could result from equipment or

facility failure, or from loss of vendor services. For instance, a site that relies on a contracted laboratory for radiochemistry analysis for bioassay samples could suddenly find itself in a crisis if the contracted laboratory were to close or the contract were canceled. Without timely support and re-establishing capabilities, site operations could be significantly impaired. A contingency plan with another DOE site to provide some limited, short-term support could allow normal site operations to continue.

The intent of this discussion of bioassay contingency plans is not to recommend establishment of a formally documented plan with implementing procedures, but that some clear thought be given to appropriate actions. The actual "plan" may be simply a paragraph or subsection in the technical basis manual or procedures identifying the contact point at another site for such support, and some indication of what would be needed (for example, a contract or inter-contractor order) to begin the support. A documented letter of agreement or understanding would be desirable.

3.2.2 Dose Management Practices Plan

The IDG describes a "Dose Management Practices Plan" (Section IV. H.). Since DOE's radiation protection program is based on total effective dose equivalent, dose management requires coordination between a site's internal dosimetry program and its external dosimetry program. For example, during an evaluation of an internal dose case, it may be important to restrict a worker's external dose. Similarly, if lifetime dose controls (as given in the RadCon Standard) were being used, a periodic reassessment of the internal doses could influence lifetime occupational dose decisions.

The dose management practices plan may be a part of the internal dosimetry procedures manual. Alternatively, the plan may be part of a higher echelon manual or contained in external dosimetry or other procedures.

3.2.3 Action Plan for Medical Response

This plan should describe the coordinated response when a medical injury is combined with potential internal dose concerns or when an intake may be sufficiently large to warrant therapeutic medical intervention for dose reduction. Medical response requires coordination between the radiation protection and medical organizations. The coordination can become even more complex when multiple contractors or subcontractors are involved and in situations where some medical services are provided by onsite personnel and some are provided by offsite sources. For example, onsite services usually include some kind of first aid response and may even involve nursing and medical doctor or physician's assistant staff. At the same time, emergency medical services (ambulance and medical trauma support) may be provided by offsite private or public organizations. A clear understanding and delineation of responsibilities and authorities in the treatment of contaminated injuries or for dose reduction therapy ought to be included in the action plan. This medical response action plan may be part of the internal dosimetry procedures or an element of other site documents.

Some examples of combined medical response and internal dose concern scenarios are provided in Section 10. Technical guidance for internal dosimetry efforts in support of medical response is also provided in Section 10.

3.2.4 Quality Assurance Plan

All steps in the activities that control or evaluate worker internal doses should be covered by written procedures that provide appropriate quality control and quality assurance. The quality assurance plan may be a section in technical basis documentation. More information is provided in Section 11.

4.0 DESIGN OF INDIVIDUAL MONITORING PROGRAMS FOR INTERNAL DOSIMETRY

In the context of internal dosimetry, individual monitoring includes routine bioassay (mentioned in 835.402(c)) and/or personal air sampling (not mentioned in 835.402(c)). The Implementation Guide on Internal Dosimetry Programs provides general guidance for the design of a bioassay program, but little guidance for air monitoring programs *as a basis for internal dosimetry*. In addition to considering all points in the IDG, DOE sites should strive to comply with ANSI N13.39, "Design of Internal Dosimetry Programs " (HPS 2000) when that standard is not in conflict with 10 CFR 835, the RadCon Standard, and the IDG, as appropriate.

There are at least two conflicts between ANSI N13.39 and the IDG where the IDG should prevail. The first is that the ANSI standard permits "censoring" of data in records, while the IDG forbids it. The second is in the definition of investigation level (see Section 4.2).

There is a wealth of information on design of bioassay programs in the technical basis documentation of many DOE sites (Sula et al. 1991; Hill and Strom 1993; Traub 1994; Baker et al. 1994; Inkret and Miller 1995; Calvo and McLaughlin 1995; Fauth et al. 1996). The reader is advised to consult this documentation for details of bioassay program design. Additional useful information on design can be found in works by Skrable (Skrable 1992) and Carbaugh (Carbaugh 1994); in element-specific standards (HPS 1996e; HPS 1994), in Regulatory Guides of the NRC (NRC 1992a, 1993a), in works of the ICRP (ICRP 1988) and NCRP (NCRP 1985a), and in the Hanford Internal Dosimetry Project Manual (Carbaugh et al. 1994).

Less information is available on design of personal air sampling programs as a basis for internal dosimetry. Readers should consult the DOE's *Air Monitoring Guide* (DOE 1999d) and documents of the NRC (NRC 1992a, 1992b, 1993a; Hickey et al. 1993). Guidance is given below on individual monitoring for the short-lived progeny of radon and thoron. As used in this DOE Standard, personal air monitoring refers to assigning specific air monitoring results to individual workers, regardless of whether the air monitoring was accomplished by general area sampling, breathing zone sampling, or individual personal (lapel) air samplers.

4.1 BIOASSAY COMPARED TO AND CONTRASTED WITH WORKPLACE AIR MONITORING

DOE's occupational radiation protection system is dose-based. 10 CFR 835.209(b) is the only requirement that addresses *methods* of internal dose assessment:

The estimation of internal dose shall be based on bioassay data rather than air concentration values unless bioassay data are:

(1) unavailable;
(2) inadequate; or
(3) internal dose estimates based on air concentration values are demonstrated to be as or more accurate.

10 CFR 835.209(b) does not *require* sites to use air monitoring data for internal dose assessment, but *permits* sites to use air monitoring data under certain conditions. "Inadequate bioassay," for compliance with 10 CFR 835.209(c), may be taken to pertain to radionuclides with effective half-lives too short to be feasible for routine or special bioassay. Such radionuclides include radon and thoron and their short-lived

progeny, as well as radionuclides such as ^{227}Th, ^{223}Ra, ^{225}Ra, and ^{225}Ac when separated from their long-lived parents. It may also include routine bioassay for isotopes with high dose per intake coefficients, such as many of the transuranics.

As defined in the IDG, a technology shortfall, such as for routine bioassay monitoring for Pu, does not preclude the use of routine bioassay monitoring nor force the use of air sample data for dose calculations. Rather, the IDG suggests that the capabilities of the bioassay program be stretched as far as reasonable, that workplace monitoring be enhanced, and that state-of-the-art techniques be used in general. Reliance should be placed on prompt detection of possible intakes in the workplace, and that special bioassay should be promptly initiated (usually the same day) when intakes are suspected. In vivo count times should be as long as reasonable, and *MDA*s should be as low as reasonably achievable, with an emphasis in both cases on "reasonable" as explained in the IDG. Air sample data may be used for initiating special bioassay without being used for dose assessment.

The DOE Office of Worker Protection Policy and Programs (previously the Office of Worker Protection and Hazards Management) prepared a "Radiological Control Technical Position" entitled "Technology Shortfalls and Dose Determinations for Radioactive Material Intakes" (Office of Worker Protection Programs and Hazards Management 1995a). This document states,

> By performing air sampling and documenting the results, in combination with an effective access control program, worker exposure measured in *DAC*-h can be tracked. Internal dosimetry programs typically base bioassay frequency and type on levels of actual or anticipated exposures to individuals. By tracking *DAC*-h for individuals, the type and frequency of needed bioassay measurements can be determined. For example, if a radiological worker receives less than 40 *DAC*-h (2 percent of an *ALI*) in a year with no respiratory protection, the individual would not be scheduled to participate in the bioassay monitoring program for that year. Additionally, participation of the individual in the bioassay monitoring program for the next year should be considered.
>
> In the case where bioassay measurements may not be available or their validity is questionable, internal dose assessments can be determined from the number of *DAC*-h tracked for that individual. When *DAC*-h are used for this purpose, any adjustments, such as protection factors for respiratory protection, must be documented.
>
> Air sampling and monitoring play an integral role in dose assessment for all isotopes, including those where the DIL is less than the detection capability. By tracking *DAC*-h, the expected magnitude of the exposure can be determined. *DAC*-h can be used to determine an individual's dose when necessary. Air monitoring provides early warning of an immediate and significant exposure hazard and provides indications of the need for special bioassay monitoring.

The monitoring criteria contained in 10 CFR 835.402(c) do not establish required levels of detection capability, that is, the minimum detectable dose. For example, it may not be feasible to actually confirm intakes that will result in 100-mrem $H_{E,50}$, particularly for bioassay measurements of some alpha-emitting radionuclides. Therefore, monitoring thresholds should not be considered requirements on the sensitivity of a particular measurement. Furthermore, workplace monitoring and occupancy factors should be considered, as appropriate, in evaluating potential exposures and monitoring needs.

10 CFR 835.402(d) requires that "internal dose evaluation programs" be capable of demonstrating compliance with the dose limits stated in 10 CFR 835.202 (e.g., 5 rems committed effective dose

equivalent in a year, or 50 rems committed dose equivalent to an organ or tissue other than the eye). In light of this requirement, there are three distinct situations for internal dosimetry programs:

1. ***Adequate technology.*** In this situation, routine bioassay measurements can show not only compliance with 10 CFR 835.202, but can be used to assess doses when $H_{E,50} \leq 100$ mrem (the investigation level). An example of an "adequate technology" situation, that is, where there is no technology shortfall, is a routine urinalysis program for ^{3}H or a routine in vivo counting program for ^{137}Cs: in each case, the *MDA* is less than the *DIL*.

2. ***Technology shortfall for routine bioassay.*** In this situation, the *DIL* is less than the *MDA* for practical routine bioassay, but special bioassay, triggered by workplace indicators, is available on short notice that can be used to show compliance with 10 CFR 835.402(d). An example of a "technology shortfall for routine bioassay" situation is a state-of-the-art internal dosimetry program for plutonium supplemented by vigorous workplace monitoring and controls.

3. ***No practical bioassay.*** In this situation, no bioassay method is available for the radionuclides in question, and no bioassay program, either routine or special, can show compliance with 10 CFR 835.202. An example of a "no practical bioassay" situation is routine worker exposure to the short-lived decay products of radon and thoron, in which no bioassay program can demonstrate compliance with the limits. In the "no practical bioassay" case, the only recourse in showing compliance with 10 CFR 835.202 is using representative air monitoring, tracking worker exposure in *DAC*-h or working level months (WLM), and performing dose assessments on the basis of the air monitoring results.

For the short-lived progeny of radon and thoron, worker stay times and measurements of potential alpha energy concentration (*PAEC*) can be converted to potential alpha energy exposure (*PAEE*) in WLM. Alternatively, worker stay times and radon concentration measurements, with knowledge or assumption of the equilibrium factor, can be converted to equilibrium equivalent *DAC*-h or to *PAEE* in WLM.

4.2 REFERENCE LEVELS AND DERIVED REFERENCE LEVELS

A reference level is a predetermined value of a quantity that triggers a specified course of action when exceeded or expected to be exceeded. Reference levels may be dose-based or intake-based. Derived reference levels are the measurement values for particular bioassay or air sampling results that correspond to a more general reference level under specifically defined circumstances. Some suggested reference levels are described below:

- Verification Level, L_V - The level of unexpected intake or dose at or above which an attempt should be made to determine if the intake is real. For example, this is the level at which special follow-up measurements should be obtained to confirm a high routine result. Below this level, it may be assumed, routine results are valid and default assumptions can be used to calculate and assign intake and dose.

- Investigation Level, *IL* - The level of intake or dose (specified in the IDG as 100 mrem) at or above which a bioassay or air monitoring result should be investigated. The intent of this level is to investigate the circumstances and, to the extent reasonable, to determine actual conditions and parameters for dose evaluation, rather than use default assumptions. An investigation may

involve special measurements, work history review, determination of material form, and modification of biokinetic parameters, and may culminate in a dose assessment.

- Medical Referral Level, L_{MR} - The level of intake or dose at or above which the medical staff shall be notified. The notification should be made as promptly as possible, but does not necessarily constitute an identified need for therapy.

Some suggested numerical values for these levels are shown in Table II. Additional discussion about the investigation level and derived investigation levels is provided in the following sections. This discussion is warranted by the definition of a 100-mrem investigation level in the IDG.

4.3 INVESTIGATION LEVEL AND DERIVED INVESTIGATION LEVEL

In the IDG, the investigation level is an $H_{E,50}$ of 0.1 rem (0.001 Sv) from intakes occurring in a year for general employees. Special *IL*s for minors, visitors, and the embryo/fetus of a declared pregnant worker should not exceed 50 mrem (0.5 mSv). Throughout this document, *IL* refers to the *IL* for the appropriate group unless otherwise specified.

In cases where it is practical, feasible, and affordable, internal dose evaluation programs should have a goal of assessing intakes of radioactive materials that occur in a year and that deliver a committed effective dose equivalent at the *IL*, that is, intakes of 0.02 stochastic annual limit on intake (*SALI)* for general employees and 0.01 *SALI* (or less) for declared pregnant workers, minors, and visitors.

With the exception of the *IL*, which is specified in the IDG on the basis of monitoring thresholds in 10 CFR 835.402, DOE sites are encouraged to consider using the alternative reference level quantities given in Table II.

Table II. Example Reference Level Magnitudes

Reference Levels (Amounts of Intake, Except for DOE *IL*)	General Employee, Except Declared Pregnant Worker		Minor, Visitor, Declared Pregnant Worker	
	Intake (*SALI*)	Corresponding $H_{E,50}$ (rem)	Intake (*SALI*)	Corresponding $H_{E,50}$ (rem)[1]
Verification Level, L_V	0.02	0.1	0.005	0.025
DOE Investigation Level, *IL*	0.02	0.1	0.01	0.05
[Alternative Investigation Level]	[0.1]	[0.5]		
Medical Referral Level, L_{MR}	1	5	1	5

[1]Note that in the case of a declared pregnant worker, the dose to the embryo/fetus is the dose to be considered, not the dose to the worker.

4.4 DERIVED INVESTIGATION LEVELS

Derived investigation levels (*DIL*s) are derived reference levels of routine individual monitoring results. Examples of *DIL*s are bioassay results, such as organ or body contents, or excreta concentrations or excretion rates, that indicate an intake resulting in a dose exceeding an *IL*. Other examples of *DIL*s are

workplace exposures, in stochastic *DAC*-h modified by a safety factor, that could lead to an $H_{E,50}$ greater than an *IL*. Internal dosimetry programs should establish *DIL*s for each individual monitoring method applied for the analysis of all radionuclides to which workers are likely to be exposed and document the derivation of such *DIL*s in the internal dosimetry technical basis documentation. The physical and chemical characteristics of the radioactive material which may be taken into the body should be taken into account in establishing *DIL*s. If an internal dosimetry program chooses to use Reference Man (ICRP Publications 23 and 30) default parameters in conjunction with modeling and assumptions recommended in ICRP Publications 30 and 54 in deriving a *DIL*, these choices should be identified in the internal dosimetry technical basis documentation. If one radionuclide is used as a tracer for a mixture of radionuclides, the *DIL* should be based on the dose from the entire mixture, not just the tracer radionuclide.

4.4.1 Factors Affecting the *DIL* for Bioassay

Factors such as significant clearance of a radionuclide in less than a year (e.g., tritium), the frequency of bioassay monitoring, and the likelihood of multiple exposures during a year (or under chronic intake conditions) should be considered in establishing a *DIL*. The *DIL* should be established so that a committed effective dose equivalent of one *IL* from all intakes in a year is likely to be detected by the monitoring program, i.e., the minimum detectable dose should be less than one *IL*. If a nonroutine or an unexpected intake of a radionuclide or group of radionuclides occurs, the minimum detectable dose may be calculated assuming a single intake that occurred on the date of the intake, if known, or the date that would result in the largest committed effective dose equivalent. If intermittent or chronic intakes are expected, the minimum detectable dose should be calculated assuming a chronic intake during the sample period.

For nonroutine or unexpected intakes, the *DIL* for each independent radionuclide or group of radionuclides ensures that a committed effective dose equivalent of not more than one *IL* would be missed in the year from intakes of that radionuclide or group.

If it is known or is likely that an individual has or could have intakes during the year from different sources that could result in doses above the *IL*, appropriately smaller *DIL*s should be determined and the basis for those *DIL*s included in the internal dosimetry technical basis documentation.

4.4.2 Calculating the Derived Investigation Level for a Given Sample Frequency

The IDG states that an *IL* of 100 mrem (0.001 Sv) of committed effective dose equivalent from all intakes occurring within a dosimetric calendar year should be used to establish *DILs*, and thus put an upper limit on the *MDA* for measurements. The desired value of the *MDA* may be further reduced by the need to confirm intakes by special follow-up bioassay: for rapidly clearing nuclides, a follow-up urine sample will generally contain a lower concentration of analyte than the initial unexpectedly high sample, but this lower concentration must still be detectable.

There are at least two approaches to calculating *DILs* as a function of sampling frequency. One acceptable alternative is to set a derived screening level based on an intake corresponding to some fraction of the *IL* (e.g., $H_{E,50}$ = 1/10 *IL* or 10 mrem for workers). The intent is to ensure that the reason and conditions of the intake are understood and that multiple intakes whose total would lead to an $H_{E,50}$ approaching the IL could not be missed. This derived screening level is for each intake, while the *IL* is for all intakes in a year. This simple approach is acceptable for exposures to multiple independent sources and is adequate for use by DOE sites.

A second acceptable alternative is to compute a *DIL* as a function of sampling frequency. With a sampling frequency of f samples per year (e.g., $f = 12$ per year for monthly samples), the goal of being able to detect 100 mrem of $H_{E,50}$ from all intakes in a year means that each analysis must be capable of detecting 100 mrem ÷ f. Thus, a yearly investigation level of 100 mrem results in a sample investigation level (IL_s) of (100 mrem/year)/(f samples/year). For example, for $f = 12$ per year, $IL_s = 100/12 = 8.3$ mrem per sample. Thus, there is a detection-level penalty for frequent sampling. The latter approach is especially important for radionuclides with short physical or biological half-lives such that multiple sampling in a year is essential. The screening level approach described above provides relief from complicated calculations by establishing the screening level *per intake*, below which a bioassay result can be disregarded, regardless of sampling frequency.

The sample investigation level is used to compute the *DIL*. Let $IRF_u(t)$ denote the intake retention function for urinary excretion at time t following a single acute intake (Bq per day excreted in urine per Bq of intake). Let $\dot{V}_u$ be the urine excretion rate for Reference Man, 1.4 liters per day. Let the effective dose conversion factor be denoted by $h_{E,50}$ (i.e., the committed effective dose equivalent per unit of activity of the radionuclide taken in by a specified route in Sv per Bq) tabulated in Tables 2.1 (inhalation) and 2.2 (ingestion) of Federal Guidance Report 11 (Eckerman et al. 1988), pages 10, 121, and 155. Let $C_u(t)$ denote the observed concentration of analyte in urine at time t. Then the committed effective dose equivalent is

$$H_{E,50} = I \cdot h_{E,50} = C_u(t) \cdot \frac{\dot{V}_u}{IRF_u(t)} h_{E,50} \,. \tag{1}$$

Rearranging the equation to solve for concentration, we have

$$C_u(t) = H_{E,50} \cdot \frac{IRF_u(t)}{h_{E,50} \cdot \dot{V}_u} \,. \tag{2}$$

This equation is used to determine the $DIL(f)$ for a given sampling frequency f by setting $H_{E,50}$ to the IL_s ($=IL/f$) and evaluating the $IRF_u(t)$ at t = (365 days per year) ÷ (f samples per year), that is, the longest period between a possible intake and bioassay:

$$DIL(f) = \frac{IL \cdot IRF_u(t = 365/f)}{f \cdot h_{E,50} \cdot \dot{V}_u} \tag{3}$$

To meet the performance objectives described in the IDG, the *MDC* or the *MDA* should be less than the $DIL(f)$. Use of Equation (3) is shown in Example 4.1.

For the more complicated case of several independent sources of radionuclides or groups of radionuclides, a more elaborate method may be needed. In many cases, the number of independent sources to which a worker will be exposed in a year is not known until the end of the year. Nonetheless, one can identify the formalism needed to calculate *DIL*s for many independent sources.

The concept of acceptable minimum detectable dose *(AMDD)* for each multiple independent source is introduced as a tool to help calculate *DIL*s. The *AMDD* is a dose value less than the *IL* by a factor that depends on the number and nature of independent sources a worker may be exposed to. To determine the

AMDD in a year for a given radionuclide or group of radionuclides *j*, it is necessary to consider the number of independent sources *n* to which an individual worker may be exposed, as shown in Figure 1. For each independent source *j*, a judgement must be made concerning whether intakes of that group are characterized as "rare, single" intakes or whether there is a possibility of multiple or chronic intakes. In the latter case, if the nuclide is rapidly clearing, then a dummy variable, p_j, is set to 1. For "rare, single"

Example 4.1. DIL for Class D Natural Uranium

Assume we have chosen a sampling frequency f = 12 samples per year. For class D uranium, IRF_u(30 days) = 0.0017 Bq per day per Bq of intake as calculated by the computer code CINDY using the ICRP 30 models. From p. 150 of Federal Guidance Report 11, $h_{E,50}$ for inhalation of class D ^{234}U = 7.37E-7 Sv/Bq, ^{235}U = 6.85E-7 Sv/Bq, and ^{238}U = 6.62E-7 Sv/Bq. Natural uranium is a mixture of these three isotopes. Since ^{234}U gives the highest dose per unit intake by a small margin, one may conservatively use the value for ^{234}U. Then, the *DIL* becomes

$$DIL(\text{inhal. Class D } ^{nat}\text{U}, f = 12/\text{year}) = \frac{0.001\ \text{Sv y}^{-1} \cdot 0.0017\ \text{d}^{-1}}{12\ \text{y}^{-1} \cdot 7.37 \times 10^{-7}\ \text{Sv Bq}^{-1} \cdot 1.4\ \text{L d}^{-1}}$$

$$= 0.137\ \text{Bq L}^{-1} \times 1\ \text{dpm}/0.0167\ \text{Bq}$$

$$= 8.2\ \text{dpm L}^{-1}\ (\text{total uranium } \alpha \text{ activity}).$$

Since the intake retention fraction decreases as the interval between samples increases (i.e., as f decreases), and the sample frequency f appears explicitly in the denominator of the *DIL* equation above, there is some optimum choice of f that requires the least detection capability. However, since annual cost is directly proportional to f, there are trade-offs between cost and detection capability.

intakes or for possible multiple or chronic intakes of slowly-clearing nuclides, $p_j = 0$. The *AMDD* (mrem per year) for each independent source then becomes

$$AMDD_j = IL \quad \text{if } p_j = 0, \text{ or}$$

$$AMDD_j = \frac{IL}{\sum_{j=1}^{n} p_j} \quad \text{if } p_j \neq 0. \tag{4}$$

In other words, *AMDDs* for rare intake radionuclides and slowly clearing multiple or chronic intake radionuclides are equal to the *IL*, and those for possible multiple intake or chronic intake groups that clear quickly are reduced by a factor of l/*k*, where *k* is the number of radionuclides or radionuclide groups *j* for which multiple or chronic intakes are possible.

A lower limit on the *DIL* for radionuclide group *j* as a function of sampling frequency can be determined. This limit is the detection sensitivity needed for a bioassay measurement, that is, the minimum *change* one would need to detect in each bioassay measurement to detect a series of small

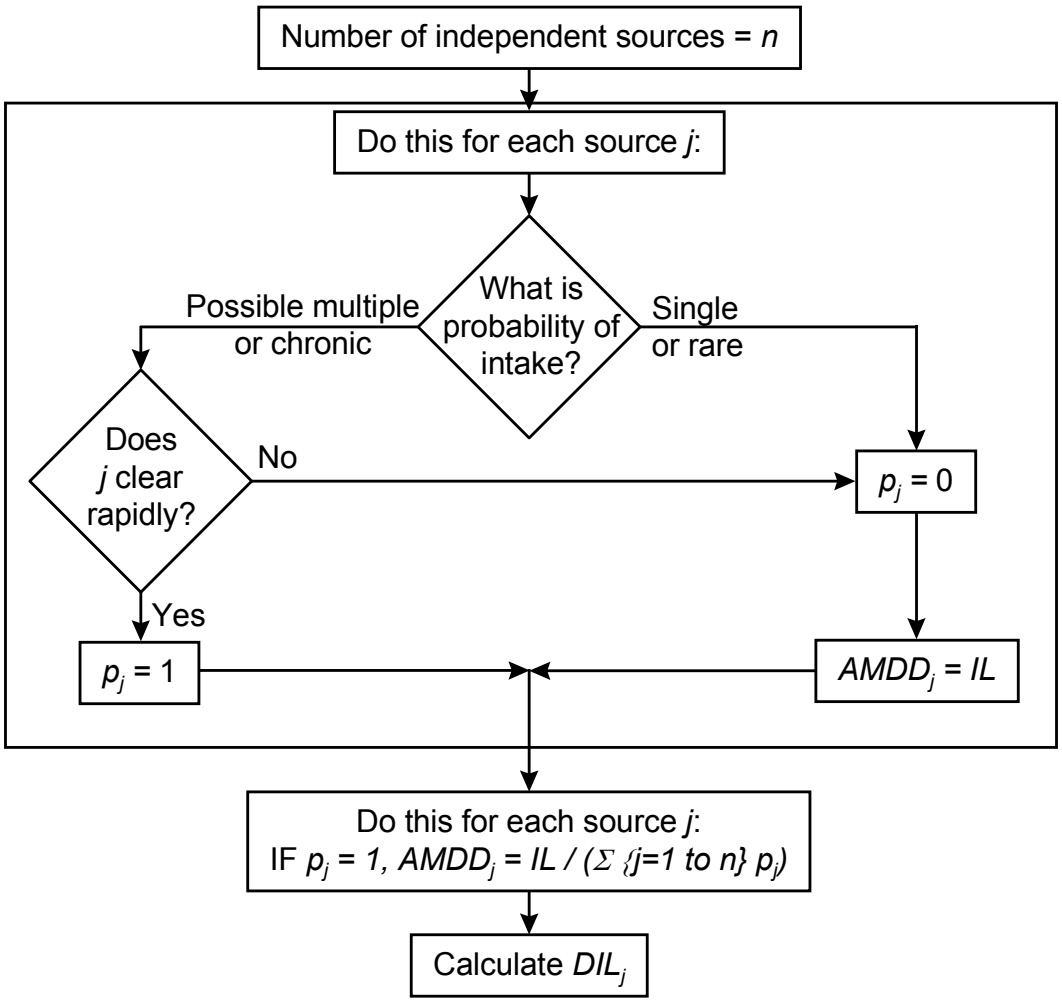

Figure 1. Alternative Logic Flow Chart for Determining the "Acceptable Minimum Detectable Dose" (*AMDD*) and *DIL* for Each Radionuclide or Group of Radionuclides When Exposure to Multiple, Independent Sources Is Possible

intakes resulting in the *AMDD* for group *j*. This detection sensitivity or minimum change in amount, ΔA_{min}, is given by

$$DIL_j(f) \geq \Delta A_{\min} = \frac{AMDD_j \cdot IRF_{\mathrm{u}}(t = 365/f)}{f \cdot h_{\mathrm{E},50} \cdot \dot{V}_{\mathrm{u}}} , \tag{5}$$

where $AMDD_j$ is substituted for the *IL*, and the other terms are as defined above. This formalism accounts for the problem of multiple independent sources.

The sampling frequency that makes minimum demands on analytical technology in terms of its detection sensitivity for analyte in bioassay samples is that frequency for which the ΔA_{min} is maximized. This sampling frequency can be found by setting

$$\frac{d}{df}(\Delta A_{\min}) = 0 . \tag{6}$$

Use of this equation can help determine the optimum sampling frequency for radionuclides for which the *MDA* is undesirably high. Example 4.2 shows this approach for tritium.

Example 4.2. Maximizing the Detection Sensitivity for Chronic Intakes of Tritium

To illustrate the dependence of the detection sensitivity on *f*, consider the $IRF_u(t)$ for ^{3}H:

$$IRF_u(t) = k e^{-0.693 t/(10 \text{ days})},$$

where *k* is a normalizing constant. Substituting 365/*f* for *t* and putting this in the $\Delta A_{\min}$ equation, we have

$$\Delta A_{\min} = \frac{AMDD \cdot k e^{-0.693 \cdot 365/(10 \cdot f)}}{f \cdot h_{E,50} \cdot \dot{V}_u} = \frac{AMDD}{h_{E,50} \cdot \dot{V}_u} \cdot \frac{k e^{-25.3/f}}{f}.$$

For the case of ^{3}H, the sampling frequency that makes minimum demands on analytical technology is

$$\frac{d}{dN}\left(\frac{k \cdot AMDD \cdot e^{-25.3/f}}{f \cdot h_{E,50} \cdot \dot{V}_u}\right) = \frac{k \cdot AMDD \cdot e^{-25.3/f}}{f^2 \cdot h_{E,50} \cdot \dot{V}_u}\left(\frac{25.3}{f} - 1\right) = 0 .$$

The solution to this is found by setting the term in parentheses to zero, giving

$$f = 25.3 \text{ samples per year} = 365 \text{ (days/year)} \cdot \lambda_{effective} \text{ (day}^{-1}\text{)}.$$

The *interval* between the samples is simply the *average* clearance time $\tau_{eff} = 1/\lambda_{eff} = 14.4$ days for ^{3}H.

A plot of the ^{3}H $\Delta A_{\min}$ for a constant total annual missed dose as a function of sampling frequency is shown in Figure 2. If sampling is done more often than once every τ_{eff}, a lower $\Delta A_{\min}$ (better analytical lab capability) is needed to see intakes resulting in the *AMDD*.

While use of the second method for establishing *DILs* may provide assurance that there is no possibility of missing intakes resulting in doses at or above the *IL*, it may be too complicated for practical implementation.

It may be possible to apply the averaging techniques of Strom and McGuire (Strom and McGuire 1993) as detailed in NUREG 1400 (Hickey et al. 1993) to improve the counting statistics, and thus reduce the *MDA* for a given bioassay technique, but this has been established only for air monitoring.

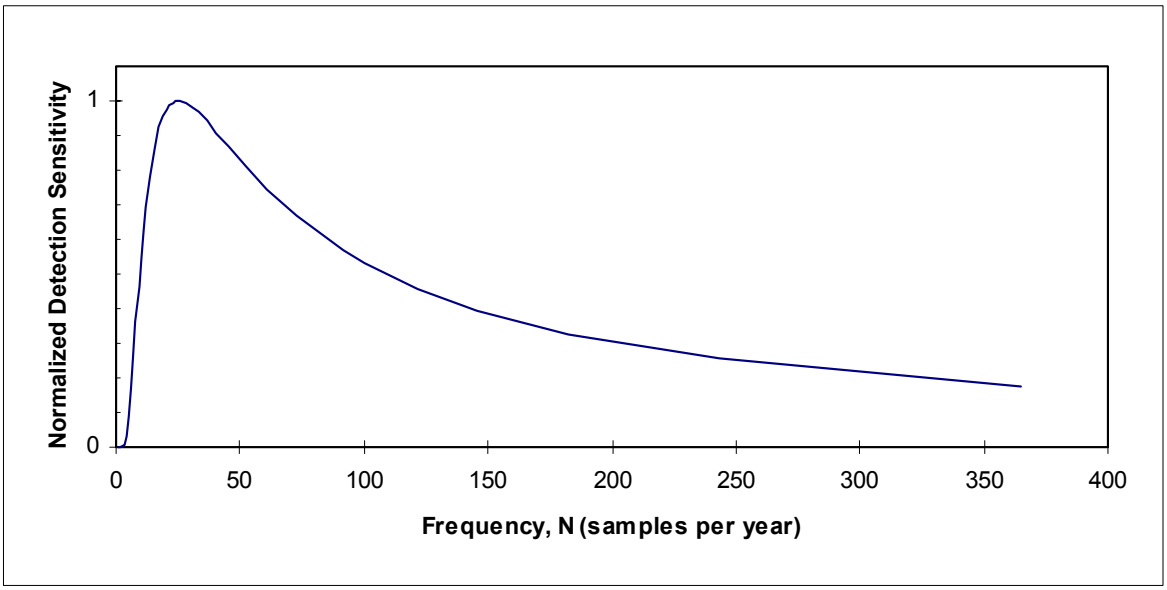

Figure 2. Plot of the Normalized Detection Sensitivity as a Function of Number of Samples per Year for ^3H

4.4.3 Factors Affecting the *DIL* for Air Sampling

A given air monitoring result may indicate a concentration higher or lower than that in the air breathed by a particular worker or workers. How well an air sample reflects the concentration actually inhaled by a worker is called "representativeness." Bioassay results, which are specific to individual, do not have this property. Efforts to correlate bioassay measurements with workplace air monitoring have shown that intakes predicted on the basis of general area (GA) air monitoring results may have limited correlation with intakes based on bioassay results. Breathing zone (BZ) air samples are more representative.

Air monitoring results, depending on where the sampler input is located, may underestimate intakes due to the "Pig Pen" effect[2], in which air is more contaminated near a worker than at some distance away. The explanation for the Pig Pen effect is simply that the worker is generating the aerosol. It is important because it impacts the degree to which an air sample represents the concentration breathed by a worker, and it leads to the need to consider a safety factor when formulating a *DIL* for air monitoring.

For an *IL* of 100 mrems of $H_{E,50}$,

$$DIL = \frac{40\ \mathrm{DAC_s\text{-}h}}{\textit{Safety Factor}\ \text{(to allow for poor representativeness)}}, \tag{7}$$

where the subscript "s" denotes "stochastic." Depending on the location of the air sampler with respect to the worker's breathing zone, the value of *Safety Factor* may be in the range of 1 to 10, based on NUREG-1400 (Hickey et al. 1993) and Caldwell's work (Caldwell 1972). Caldwell showed that, for plutonium work, fixed station air samplers tended to dramatically underestimate intakes assessed from fecal samples,

[2]Named after the Charles Schultz character in the Peanuts™ comic strip who walks around in a cloud of dust and debris.

and that lapel-type breathing zone air samplers more accurately corresponded to intakes predicted using the 1966 ICRP lung model and fecal data. He also showed wide variability between breathing zone air results and general area air results, with median BZ/GA ratios between 3 and 8, and 90 percentile ratios from 9 to 26.

4.4.4 Supplementing Routine Bioassay Programs When *DIL* < *MDA*

DOE's 10 CFR 835.402(c) requires that, with a likelihood for 0.1 rem of $H_{E,50}$, a worker must be on a dose evaluation program. There is no requirement that the program be able to detect 0.1 rem of CEDE, only that it has to detect 5 rems of CEDE, as in 10 CFR 835.402(d).

To gain insight on the question of detection capability, one may examine requirements for external irradiation. There is the same 0.1-rem threshold for external monitoring, but an additional requirement that external dosimeters be accredited by the U.S. Department of Energy Laboratory Accreditation Program (DOELAP). Since 10 CFR 835 is a requirements document, then the standards in the DOELAP manual (DOE 1986) are requirements. Thus, personnel dosimeters must be able to detect 0.03 rem in several categories of radiation exposure. The practice for external irradiation is to require not only detection capability at 30% of the monitoring threshold, but also fairly precise and accurate detection capability at that level. By analogy, one might consider it desirable for an internal dosimetry program to be capable of detecting $H_{E,50}$ values in the same range. However, this is not always practical or even feasible.

There is technology shortfall for routine bioassay programs when the *DIL* is lower than the *MDA*. When a bioassay program has *DIL* < *MDA*, BZ or personal air monitoring may be implemented to supplement the routine bioassay program, as illustrated in Example 4.3.

Personal air samples are often more representative that fixed samplers. However, personal air samplers have a lower flow rate than most fixed air samplers. Example 4.4 shows how averaging of periodic results can be used to lower the MDC.

4.4.5 A Potential Technology Shortfall for Breathing Zone Air Sampling

Breathing zone air sampling may not be adequate in facilities where ^{238}Pu or another high specific activity alpha emitter is processed. High specific activity radionuclides usually have shorter half lives than lower specific activity isotopes. The problem with high specific activity radionuclides arises from the fact that a small number of particles can be significant from a dose standpoint as illustrated in Example 4.5. However, as shown in Example 4.6, a similar concern does not exist for isotopes with lower specific activity, such as ^{239}Pu. Thorough discussions of the problems with detecting and quantifying intakes using personal air samplers, including accounting for slip correction, are given by Birchall et al. (1985, 1986, 1987, 1991) and by Scott et al. (1997).

There is historical precedent for a BZ or personal air monitoring program supplemented by an aggressive fecal sampling program in NRC-licensed plutonium facilities. A facility operated in the 1960s and 1970s in Parks Township, Pennsylvania, by NUMEC, ARCO, and most recently by Babcock & Wilcox (Caldwell 1972), which processed reactor-grade plutonium, did not have significant trouble with the "countable number of particles" problem discussed in Example 4.5.

Example 4.3. Use of Breathing Zone Air Samples to Supplement Routine Bioassay for Plutonium

To illustrate the detection capability of breathing zone air monitoring, consider the *DAC* for class Y ^{239}Pu of 6E-12 µCi/mL (10 CFR 835, Appendix A). Multiplying by 2.4E9 mL/year breathed by Reference Man, one derives DOE's nonstochastic annual limit on intake (*NALI*) for class Y ^{239}Pu as 1.44 E-2 µCi = 14.4 nCi (533 Bq). The complementary "5-rem" stochastic annual limit on intake *(SALI)* from ICRP 30 is 16.2 nCi (600 Bq). Then, 2% of a *SALI* (that is, the intake that would result in a $H_{E,50}$ of 100 mrem) is 324 pCi (12 Bq), or 720 dpm of Pu. (The *SALI* for class W material is about 3 times lower.)

Suppose a worker was exposed to an atmosphere in which, breathing at Reference Man's rate of 20 liters per minute, he would experience an intake of 2% of a *SALI*. A BZ or personal air sampler operating at 20 L/min would collect this same 720 dpm of Pu activity. A lapel air sampler operating at 1.8 L/min would accumulate about 64 dpm (1.1 Bq). Thus, for a single air sample, there is no difficulty (in the sense of counting statistics problems) achieving detection capabilities comparable to those that the DOE requires for external radiation monitoring using BZ or personal air samples *for a single filter*.

Personal air sampler filters are likely to be changed every day, or 250 times in a year. Thus, the 720 dpm, which is 2% of the *SALI*, could be on one filter or spread among many or all. The minimum detectable intake for uniform, chronic exposure based on 250 samples is higher than the minimum detectable intake for a single, acute exposure. See Example 4.4.

Another benefit of BZ air monitoring programs is that they give workers feedback about work practices. The experience at Apollo, Pennsylvania, showed that workers develop better radiological control habits based on BZ air sample results.

It is well known that bioassay is much more accurate than BZ or personal air monitoring when bioassay results are available and adequate. However, when bioassay methods are not adequate or unavailable, BZ or personal air monitoring should be used. (See 10 CFR 835.209(b).) When there is technology shortfall for routine bioassay, DOE sites should consider using BZ or personal air monitoring programs to supplement their routine bioassay programs. Such use should be tempered with an understanding of the limitations described in Example 4.5.

4.4.6 Performance Specifications for a Bioassay Laboratory

10 CFR 835.402 requires internal dose monitoring programs implemented to demonstrate compliance with 10 CFR 835 be accredited by the U.S. Department of Energy Laboratory Accreditation Program (DOELAP) for Radiobioassay (DOE 1998a). Radiobioassay laboratories utilized by the internal dose monitoring programs will be evaluated against the requirements of the "Department of Energy Laboratory Accreditation Program for Radiobioassay" (DOE 1998b) which incorporates the recommendations of HPS N13.30-1996, "Performance Criteria for Radiobioassay" (HPS 1996a). In addition, they may wish to consider the requirements of ANSI N42.23-1996, "Measurement and Associated Instrumentation Quality Assurance for Radioassay Laboratories" (ANSI 1996). Additional specifications for the bioassay or service laboratory should be negotiated between the site and the laboratory. Example 4.7 gives performance specifications for a radiobioassay laboratory.

Example 4.4. Improving Detection Capabilities of Air Sampling Using Averaging

The minimum detectable average concentration for repeated BZ or personal air samples over a year or other period of time can be reduced by averaging the original raw data, as described in the Appendix to NUREG-1400 (Hickey et al. 1993a, Strom 1993). The simplest case is when independent activity measurements are made of a sequence of samples for which large numbers of counts (i.e., more than 50) are collected and for which the following remain identical between samples: background count times and rates, sample count times, and counting yields. In such a case, the *MDA* for the sum of n samples is larger than that for a single sample: $MDA(n) = \sqrt{n} \cdot MDA(1)$. Conversely, the minimum detectable average concentration $(MD\bar{C})$ for n samples is smaller than the minimum detectable concentration *(MDC)* for a single sample: $MD\bar{C}(n) = MDC(1)/\sqrt{n}$, when sample volumes or masses are all equal, equal sample collection times are used, and collection efficiencies are equal. Although the *MDA* for such pooled samples increases by $\sqrt{n}$, the volume or mass in which this activity is found increases by a factor of *n*, resulting in a net decrease in $MD\bar{C}$ by a factor of $\sqrt{n}/n = 1/\sqrt{n}$. In general, samples may have varying count times, background count rates, counting efficiencies, collection efficiencies, and collection times. Exact time-weighted formulas for $MD\bar{C}$ and decision level (*DL*) are given for the general case in the Appendix to NUREG-1400, and exact formulas are provided for both large and small numbers of background counts (Hickey et al. 1993a). This methodology is useful in situations where daily, weekly, or monthly concentration measurements must be compared to an annual limit. It is also useful in determining the detection capabilities of a measurement program. This work shows the importance of reporting measurements and their standard deviations as observed, rather than "censoring" them by reporting them as "less than" values.

An alternative to averaging is to *physically* combine air filters containing long-lived material. For example, if a worker had 200 separate personal air sample filters during a year, they could be combined and the composite analyzed as a single sample. If the material were a penetrating photon-emitter, the ensemble of filters could be counted directly by gamma spectroscopy. If the material were an alpha-emitter, radiochemistry would be necessary.

4.5 MEASUREMENTS OF WORKPLACE RADON AND THORON CONCENTRATIONS, POTENTIAL ALPHA ENERGY CONCENTRATIONS, AND MEASUREMENTS OF (OR ASSUMPTIONS ABOUT) EQUILIBRIUM FACTORS

4.5.1 Measurements

There are two objectives of radon/radon progeny monitoring and hence two sets of standards for these measurements. The two monitoring objectives are 1) to characterize in real time the concentrations that workers might be exposed to while in an area and 2) to establish the exposure of record that each worker actually receives. In the *Air Monitoring Guide* (DOE 1999d), these two types of monitoring are respectively referred to as air monitoring and air sampling. It will generally be found that meeting both objectives is best achieved using two different types of instruments.

Instruments used for both purposes should measure either airborne radon or radon progeny concentration. If materials containing thorium-232 or its progeny are known to be present in the area, the instruments should also be capable of measuring airborne thoron progeny concentrations.

Example 4.5. Potential Technology Shortfall for Breathing Zone Air Sampling of High Specific Activity Alpha Emitting Nuclides

For high specific activity alpha emitters, a single large particle on an air sampler filter may give erroneous results, a phenomenon that can be described as the "countable number of particles problem." In facilities where ^{238}Pu is processed, it may be difficult to use BZ or personal air monitoring to control intakes near the level of 2% of a *NALI*. Using the methods in ICRP Publication 66, and a density of 11 g/cm^3 for plutonium oxide (ICRP 1994a and p. 1.7 of Faust et al. 1988), the table below was calculated. The equivalent physical diameter also accounts for slip correction and thermodynamic effects, both important at small particle sizes. The table shows that one particle with an aerodynamic diameter of 5 µm is approximately 2% of a *NALI*.

With Monte Carlo analysis, Scott et al. (1997) show that calculated average intake of high specific activity alpha emitters, in *DAC*-h, is not an operationally useful quantity. They used a light activity breathing rate of 1.5 m^{3}h^{-1}, a density of 10.0 g cm^{-3}, an *AMAD* of 5µm, and a *GSD* of 2.5 and calculated the intakes of 10,000 workers exposed. In an 8 *DAC*-h exposure, 9,831 had no intake, 4 had intakes greater than one *ALI* (that is, 2,000 DAC-h or 600 Bq of ^{238}Pu), and 165 had intakes ranging from a fraction of a *DAC*-h to nearly 2,000. All intakes resulted from inhaling a single particle of $^{239}PuO_2$. Thus, the *average* intake computed for the group of workers, would both overestimate the intakes of the vast majority of individuals and seriously underestimate intakes of the more highly exposed individuals.

Aerodynamic diameter (µm)	Equivalent physical dia. (µm)	Volume (cm^3)	Mass per particle (g)	Activity per particle (Bq)	Number of particles per *NALI*	Number of particles per 0.02 *NALI*
0.1	1.2E-02	8.1E-19	8.9E-18	5.0E-06	1.5E+08	2.9E+06
0.2	2.9E-02	1.2E-17	1.3E-16	7.5E-05	9.6E+06	192,551
0.3	5.0E-02	6.7E-17	7.3E-16	4.1E-04	1.8E+06	35,224
0.5	1.0E-01	5.6E-16	6.2E-15	3.5E-03	207,805	4,156
0.7	1.6E-01	2.1E-15	2.4E-14	1.3E-02	54,698	1,094
1	2.5E-01	8.1E-15	8.9E-14	5.0E-02	14,530	291
2	5.5E-01	8.6E-14	9.5E-13	5.3E-01	1,361	27
3	8.5E-01	3.2E-13	3.5E-12	2.0E+00	367	7.3
5	1.4E+00	1.6E-12	1.8E-11	9.8E+00	74	1.5
7	2.1E+00	4.5E-12	5.0E-11	2.8E+01	26	0.52
10	3.0E+00	1.4E-11	1.5E-10	8.3E+01	8.7	0.17
20	6.0E+00	1.1E-10	1.2E-09	6.8E+02	1.05	0.021
30	9.0E+00	3.8E-10	4.2E-09	2.3E+03	0.31	0.0062
50	1.5E+01	1.8E-09	2.0E-08	1.1E+04	0.066	0.0013
70	2.1E+01	4.9E-09	5.4E-08	3.0E+04	0.024	0.00048
100	3.0E+01	1.4E-08	1.6E-07	8.8E+04	0.0082	0.00016
200	6.0E+01	1.1E-07	1.3E-06	7.0E+05	0.0010	0.00002
300	9.0E+01	3.9E-07	4.3E-06	2.4E+06	0.00030	6.1E-06

Example 4.6. The Number of Particles for Breathing Zone Air Sampling of a Lower Specific Activity Radionuclide

In the previous example, it was shown that breathing zone air sampling in a facility that handled ^{238}Pu might not be useful. For lower specific activity material, there is no similar problem. For example, "6%" plutonium (Table 9.2 in Sula et al.1991), aged 14.4 years, has 50% ingrowth of ^{241}Am (Rittmann 1993), a specific activity of 3.44E9 Bq/g of α-emitters, and an α-*NALI* of 458 Bq (bone surfaces). Using the same density for plutonium oxide and calulational approach as in Example 4.5, gives the table below. It shows that the number of particles corresponding to 2% of a *NALI* does not create is a problem until the *AMAD* > 10 µm. It is important, however, to minimize accidental filter contamination by even one "large" particle. One 10-µm particle corresponds to an $H_{T=bone\ surfaces,50}$ of about 60 mrem and an $H_{E,50}$ of about 5 mrem.

Aerodynamic Diameter (µm)	Equivalent Physical Dia. (µm)	Volume (cm^3)	Mass per particle (g)	Activity per particle (Bq)	Number of particles per *NALI*	Number of particles per 0.02 *NALI*
0.1	0.012	8.1E-19	8.9E-18	3.1E-08	1.5E+10	3.0E+08
0.2	0.029	1.2E-17	1.3E-16	4.6E-07	9.9E+08	2.0E+07
0.3	0.050	6.7E-17	7.3E-16	2.5E-06	1.8E+08	3.6E+06
0.5	0.10	5.6E-16	6.2E-15	2.1E-05	2.1E+07	429,056
0.7	0.16	2.1E-15	2.4E-14	8.1E-05	5.6E+06	112,937
1	0.25	8.1E-15	8.9E-14	3.1E-04	1.5E+06	30,000
2	0.55	8.6E-14	9.5E-13	3.3E-03	140,494	2810
3	0.85	3.2E-13	3.5E-12	1.2E-02	37,912	758
5	1.4	1.6E-12	1.8E-11	6.0E-02	7,590	152
7	2.1	4.5E-12	5.0E-11	1.7E-01	2,676	54
10	3.0	1.4E-11	1.5E-10	5.1E-01	895	18
20	6.0	1.1E-10	1.2E-09	4.2E+00	109	2.2
30	9.0	3.8E-10	4.2E-09	1.4E+01	32	0.64
50	15	1.8E-09	2.0E-08	6.7E+01	6.8	0.14
70	21	4.9E-09	5.4E-08	1.8E+02	2.5	0.050
100	30	1.4E-08	1.6E-07	5.4E+02	0.85	0.017
200	60	1.1E-07	1.3E-06	4.3E+03	0.11	0.0021
300	90	3.9E-07	4.3E-06	1.5E+04	0.031	0.00063

Instruments used for air monitoring should be real-time monitors that continuously measure and display results for periods of one hour or less. They should be placed to measure the highest concentrations to which workers in the area are likely to be exposed.

Example 4.7. Example of Performance Specifications for a Bioassay Laboratory

The radiobioassay laboratory shall meet the contractual minimum detectable amounts, as defined in HPS N13.30-1996, as listed in [DOE site to provide specific list].

Control sample results shall, as a minimum, meet the criteria concerning relative bias statistics within -0.25 to +0.50 and the relative precision statistic shall be less than or equal to 0.4. At the levels to be used in spikes, the bias and precision should normally be smaller than the limits in HPS N13.30-1996. The radiobioassay laboratory shall verify that these limits are met.

The radiobioassay laboratory (if required by 10 CFR 835) shall participate in the DOELAP for Radiobioassay. The radiobioassay laboratory shall achieve satisfactory results for all appropriate test categories. In addition, the radiobioassay laboratory should participate in traceability-testing for bioassay sample matrices offered through NIST's Radiochemistry Intercomparison Program (NRIP). (Note: non-bioassay matrices are not good indicators of bioassay laboratory performance) The radiobioassay laboratory shall furnish the DOE site with all intercomparison data annually and/or upon request.

The radiobioassay laboratory shall furnish the DOE site with all internal quality assurance and quality control (QA/QC) data upon request.

The radiobioassay laboratory's quality assurance program shall be implemented through an established documented plan. The QA program must also satisfy HPS N13.30-1996.

The radiobioassay laboratory will prepare and analyze reagent blanks and spiked urine and fecal samples for internal quality control. The number of QC spiked samples shall be at least 5% of the total samples analyzed and a reagent blank shall be analyzed with each set of samples. The reagent blanks will be used by the radiobioassay laboratory and the DOE Site, during audits and review of bioassay reports, to verify that all detection levels comply with the Contractual Detection Levels specified above. (The correct equation for verification of detection level is documented in HPS N13.30-1996.)

The radiobioassay laboratory must satisfy initially and on a continuing basis certain quality control factors specified below concerning yields, resolution, contamination and control standards or a "stop work order" may be enforced until the problem(s) is resolved. The radiobioassay laboratory will report internal quality control results to the DOE Site Procurement Manager when requested.

The DOE site may send, from time to time, blind spiked and/or blank samples to the radiobioassay laboratory. These sample results will be compared to the in vitro performance criteria documented in HPS N13.30-1996 and will be used in conjunction with the radiobiossay laboratory's in-house quality control results to determine if the radiobioassay laboratory is meeting the Contractual Detection Levels. (Note: A limited number of blanks are not a good indicator of the true MDA. It is better to use the lab's QC results).

(continued)

Example 4.7 (continued)
Example of Performance Specifications for a Bioassay Laboratory

Other factors that should be negotiated and put into the statement of work include turnaround time(s) for analytical results, especially for special bioassay; the need for prompt notification of results that exceed certain levels; and length of storage time for unused portion of samples or final analyzed preparation of samples (e.g., counting planchet) to allow for reanalysis or recounting of samples, if necessary.

The radiobioassay laboratory is required to maintain a QA manual that outlines responsibilities and also provides requirements for data control, document control, maintenance/test equipment calibration and checks, procedures, training, corrective action in the event of noncompliance, and traceability to standardizing bodies such as the National Institute of Standards and Technology (NIST) (when available).

All instruments used for the analysis of the radionuclides in the bioassay program shall be properly "response"-checked before being used to analyze the DOE site's samples. The results of the response checks shall be documented for each instrument that requires calibration (e.g., radiation detectors, scales, balances, etc.). All radiation detection instruments used for analysis of the radionuclides in the bioassay program shall be calibrated at least annually using NIST-traceable standards when they are available. A NIST certificate for all standards (when available) shall be retained by the radiobioassay laboratory and shall be made available to the DOE site for review.

Additional Quality Control Factors

Yields: The average yields determined for plutonium and strontium separated from urine and feces shall be at least 50% without restrictions, and at least 25% if it is determined that contractual minimum detectable amounts can be met. For americium and uranium, the average yields shall be at least 40% and 20%, respectively.

Resolution: The resolution of α-particle spectrum energy peaks shall be less than 100 keV full width at half maximum.

Contamination: The results of the reagent blanks shall be at least low enough to allow meeting the minimum detectable amounts. Any trend or sudden change towards increase in activity in blanks or their standard deviations that may cause the contractual minimum detectable amounts to be exceeded should be investigated and the cause eliminated.

Contamination of the final fraction of one element with the nuclide of another element becomes important in alpha-particle spectrometry, particularly when it involves nuclides with alpha energies that cannot be resolved (energy peaks within one full width at half maximum of each other). Whenever potentially interfering foreign nuclides appear in the final fraction of any element, the cause for the contaminations should be identified and eliminated. If the magnitude of the contamination adversely affects the result, work shall be stopped until the problem is solved. However, work stopage is not warranted for an isolated suspected contamination event.

(continued)

Example 4.7 (continued)
Example of Performance Specifications for a Bioassay Laboratory

Quality Control Spikes: HPS N13.30-1996 requires that control sample results, as a minimum, have a relative bias statistics within -0.25 to +0.50 and a relative precision statistic of less than or equal to 0.4. At the levels to be used in spikes, the bias and precision should normally be smaller than the limits in HPS N13.30-1996. The radiobioassay laboratory shall verify that these limits are met.

Instruments used for air sampling should be continuous instruments that make either time-averaged or real-time measurements. They should be placed so as to measure as nearly as is practicable the concentrations to which workers are exposed. In areas with large gradients of concentration or equilibrium (e.g., outdoors), individual personnel monitors should be used for each worker.

Several good references are available for radon and thoron measurements (NEA 1985; NCRP 1990; Fortmann 1994). Sheets gives a recent review of indoor thoron with many references (Sheets 1993).

4.5.2 Equilibrium Factors

If radon measuring instruments are used, radon progeny concentration should be inferred by application of an appropriate equilibrium factor. In general, equilibrium factors should be measured under a representative set of circumstances and for a representative time frame.

If it is not practical to measure equilibrium factors, a default ^{222}Rn equilibrium factor of 0.4 (ICRP 1993a; UNSCEAR 1993) may be used for indoor areas with normal ventilation rates and outdoor areas with radon sources no closer than 400 m (≈1/4 mile;Table III). Average indoor equilibrium factors increase with increasing particle concentration in air, and decrease with increased air exchange rate (James et al. 1988; James 1994; National Research Council 1991; NEA 1985; UNSCEAR 1993). For outdoor areas with local sources of radon and highly ventilated indoor areas, the appropriate equilibrium factor should be determined by concurrent radon and radon progeny measurements made over a set of conditions that present the range of equilibrium factors to be encountered when workers are present. These measurements and the rationale for their application to inferring radon progeny concentration should be documented in the internal dosimetry technical basis documentation.

Table III. Acceptable Default Equilibrium Factors for Radon (F_{Rn})

Location/ Environment	Default Equilibrium Factor (F_{Rn})
Indoors, normal ventilation	0.4
Indoors, unusual ventilation	Measure
Outdoors - no local radon sources	0.4
Outdoors with “local” radon sources	Measure

Appendix A contains a review of measurements of radon progeny equilibrium factors on which Table III is based. Appendix A also contains a brief review of published values of thoron progeny equilibrium factors.

4.5.3 Performance Criteria for Instruments Used at Doe Sites to Characterize Airborne Radon and Thoron and Their Progeny

The American National Standards Institute provides performance specifications for instruments for the measurement of radon, radon progeny, and thoron progeny in air (ANSI 1994a, 1994b). All instruments should be operated using standardized approved operating procedures. All operators should be trained on these procedures prior to performing field measurements.

4.5.3.1 Air Monitoring

Instruments used for air monitoring should have the following characteristics:

- a response rate that is limited only by radon progeny ingrowth (i.e., a full-scale response time of about 4 hours; does not apply to thoron),
- a sensitivity to environmental influences that complies with the applicable parts of ANSI N42.17A-1994 and ANSI N42.17B-1989,
- a coefficient of variation of no more than 15% when making one-hour measurements of constant, normal background concentrations, and
- a calibration bias of no more than 10%.

Enclosing the instrument in a protective housing may be necessary to limit environmental influences.

4.5.3.2 Air Sampling

Instruments used for air sampling should have the following characteristics:

- a sensitivity to environmental influences that complies with the applicable parts of ANSI N42.17A-1994 and ANSI N42.17B-1989,
- a coefficient of variation of no more than 15% when making 170-hour measurements of constant, normal background concentrations, and
- a calibration bias of no more than 10%.

4.5.4 Participation by DOE Sites in an Intercomparison Program for Radon Instrument Calibration, Precision, and Accuracy

Compliance of the measuring system(s) with the above performance specifications should be demonstrated by subjecting a representative sample of instruments to periodic (annual if possible) radon and/or radon progeny comparison exercises, if and when such exercises are conducted by Department of Energy laboratories. If and when the DOE Laboratory Accreditation Program (DOELAP) offers a radon measurements program, DOE sites should participate in the DOELAP.

4.5.5 Calibration and Quality Control for Radon, Thoron, and Progeny Instrumentation

All instruments should be recalibrated at least annually. The lack of stability of some instruments may require that they be calibrated more frequently. Calibrations should be performed in a controlled atmosphere which is monitored with instruments whose flow rate and detection efficiency have been

determined by reference to standards traceable to the National Institute of Standards and Technology, if such standards are available.

Periodic functional tests should be performed at a frequency dependent on the performance history of the instrument. As a minimum, these tests will include checks of the airflow rate and detector efficiency. Replicate pairs of measurements should also be performed on a rotating schedule that covers all instruments at least once every two months.

4.5.6 Use of Engineering Controls for Management of Exposures to Radon, Thoron, and Their Short-Lived Decay Products

The use of engineering control methods for radon and thoron should be based on cost-benefit analyses because they can be expensive to implement. Engineering controls for new building construction may be significantly cheaper than for existing construction. Guidance and model standards are available from the U.S. Environmental Protection Agency for reducing radon levels in existing construction (EPA 1989, 1991a, 1992, 1993, 1994b, 1994c). Such methods may be appropriate when the radon is due to DOE "activities" as defined in 10 CFR 835. Engineering controls for contaminated sites with elevated radon levels due to DOE activities may not be cost-effective, and personnel protective equipment or other radiation protection measures such as limiting stay times, performing work at times of the day when radon progeny levels are lower, etc., may be needed.

All new construction at DOE facilities that will be occupied for significant periods of time should be "radon-resistant" construction. References for radon-resistant construction methods are available from the U.S. Environmental Protection Agency and ASTM (EPA 1991b, 1994a; ASTM 1992). Making new structures radon-resistant generally adds little to the cost of construction.

5.0 INDIVIDUAL MONITORING FOR INTERNAL DOSIMETRY

5.1 SCOPE OF PARTICIPATION IN INDIVIDUAL MONITORING PROGRAMS FOR INTERNAL DOSIMETRY

Workers considered likely to have intakes resulting in an $H_{E,50}$ in excess of 100-mrem are required by 10 CFR 835.402(c) to participate in an "internal dosimetry program." Measurements from individual monitoring programs are needed as input to an internal dosimetry program. In the context of internal dosimetry, individual monitoring includes routine bioassay (mentioned in 10 CFR 835.402(c)) and/or personal air sampling (not mentioned in 10 CFR 835.402(c)). This section gives criteria for participation in individual monitoring programs, which include baseline, routine, special, and termination or task-ending bioassay and personal air sampling programs.

Most radiation protection programs should be capable of preventing intakes through rigorous application of engineering and administrative controls. Under such controls, a good argument can be made that no one is likely to have an intake resulting in a $H_{E,50}$ of 100 mrem. This may reduce the need for participation in a routine bioassay program (meaning scheduled periodic measurements) but does not eliminate the need for confirmatory or special bioassay monitoring. Likewise, the need for an internal dosimetry program is linked more to the potential for intake than the likelihood of intake. If sufficient quantities of radionuclides are present or handled at a facility that accidental intakes resulting in 100-mrem $H_{E,50}$ cannot be ruled out, an internal dosimetry program must be available.

5.2 BASELINE INDIVIDUAL MONITORING: BIOASSAY

Baseline monitoring involves determining the worker's bioassay status at the start of employment or potential exposure, and obtaining appropriate baseline measurements. Internal dosimetry programs that must of necessity be based on air sampling have no analog for baseline bioassay monitoring.

The concept of establishing a baseline does not necessarily mean that baseline bioassay measurements be obtained. Administrative review of the worker's history can lead to the conclusion that baseline measurements are not needed because the expected results are readily predictable (e.g., no detectable activity). Such a review can constitute acceptable baseline monitoring.

If baseline measurements are needed, they should be completed before performing work requiring routine bioassay. Baseline measurements are appropriate for any of the following circumstances: (1) the worker has had prior exposure to the pertinent radionuclides and the effective retention in the body might exceed the screening level, (2) the exposure history is missing or inconclusive, or (3) the worker will be working with radioactive material which may be potentially detectable in bioassay due to non-occupational sources. Illustrations of baseline bioassay scenarios are given in Example 5.1.

Example 5.1. Baseline Bioassay Scenarios

- A new employee in a plutonium facility would not require a baseline bioassay measurement if there was no prior potential occupational exposure to plutonium. However, a new employee at a plutonium facility who came from another facility where plutonium was a nuclide of concern should undergo baseline measurements if a termination bioassay was not performed by the former employer or work history information is absent.

- Workers with potential exposure to uranium should receive baseline uranium urinalyses due to the ubiquitous and highly variable occurrence of uranium naturally and its possible presence in urine.

- Workers with potential exposure to ^{137}Cs should receive baseline whole body counts because of environmental ^{137}Cs, present from worldwide atmospheric fallout, can be present in low-levels in certain food products.

5.3 PARTICIPATION IN ROUTINE INDIVIDUAL MONITORING PROGRAMS: BIOASSAY AND/OR PERSONAL AIR SAMPLING

Workers considered likely to receive intakes which could result in $H_{E,50}$ values in excess of 100 mrem or who are at risk for such intakes should participate in a routine individual monitoring program that includes bioassay and/or personal air sampling. Those workers are identified using criteria based on knowledge of the radionuclides, facilities, processes, and anticipated work. Criteria may be expressed in many forms, including quantity and form of material handled, type of work, or category of worker. There is no single method that is most cost-effective and technically correct for identifying those workers. Example 5.2 presents criteria for determining the need for routine participation in a bioassay and/or personal air sampling program and sample applications of those criteria. Example 5.3 gives instances in which personal monitoring is not needed..

The ICRP (1988) recommends the order of preference for bioassay program data interpretation to be (1) direct in vivo measurement of body content, (2) excreta analysis, and (3) personal air sampling. However, the radionuclide or element being monitored and its characteristic radiations usually establish the type of monitoring performed.

Participation in routine individual monitoring programs may be discontinued when sufficient facility history and experience is available to indicate that there is no need for a routine program. However, in such cases, a confirmatory monitoring program (see Section 5.7) may be of value.

5.3.1 Exposure Monitoring Thresholds for Radon and Thoron Progeny

Since there is no practical bioassay for radon and thoron, exposure monitoring is required when individuals have the potential to be exposed in excess of the dose levels given in 10 CFR 835.402(c) requiring monitoring. It is important to emphasize that the radon and thoron exposure monitoring thresholds are exposure-based (WLM or *DAC*-h) versus concentration-based thresholds because of the dynamic nature of radon concentrations.

Example 5.2. Criteria for Participating in Individual Monitoring Programs

Criterion 1: Quantity of radioactive material in process

This criterion establishes a maximum working activity (MWA) or a mixture specific activity above which individual monitoring is recommended. The MWA is a quantity calculated using the nuclide stochastic *ALI*, and factors for such considerations as physical form of the material, containment or confinement methods, dispersibility based on the processing being performed on the material, occupancy, and a special form factor for DNA precursors. Examples of such formulations are provided in NUREG-1400 (Hickey et al. 1995) and the Hanford Internal Dosimetry Project Manual. Recent discussion among some health physicists suggests that the factors used in NUREG-1400 may be too liberal (i.e., too few people would be monitored), and this issue may be addressed in a future ANSI standard. The mixture specific activity approach is described by Carbaugh et al.1995 and applies to situations where radioactivity is essentially uniformly mixed with a large volume or mass of inert material (e.g., contaminated soil).

Criterion 2: Worker training and tasks

Workers with Radiation Worker II training and who work with radioactive materials may be scheduled for routine bioassay and/or routine personal air sampling. This is a very broad-scope practice, giving rise to large programs. While it is easy to implement, it is likely to result in requiring personal internal dosimetry measurements of workers who are not likely to exceed 100 mrem of $H_{E,50}$. The cost of the unnecessary measurements is a tradeoff for less scrutiny of actual worker assignments.

Criterion 3: After-the-fact determination of bioassay need based on actual work performed (does not apply to air sampling)

An aggressive program with continuing checks on worker potential exposures (e.g., entries into contamination areas or under specific radiation work permits) may be able to retroactively determine the need for bioassay based on actual work. Such a program might review a worker's actual activities during the course of the last routine bioassay interval (e.g., one year) and determine that no potential for exposure occurred. Under such circumstances, the bioassay measurement which might otherwise be routinely obtained could be omitted. This practice calls for close review of an individual worker's activities. The cost savings for omitted bioassay must be weighed against the cost of work history review to determine the net cost savings.

Criterion 4: Use of respiratory protection to limit intake and dose

When respiratory protection is used to limit intake of radioactive material, 10 CFR 835.403(a)(2) requires that air monitoring be done, as necessary, to characterize the hazard. In addition this Technical Standard recommends that workers participate in routine bioassay monitoring if respiratory protection is used to limit the intake of radioactivity (i.e., respiratory protection factors are being used to limit the estimated intake of radioactivity). Routine bioassay may be omitted if respirators are used as a matter of conservative protocol without any actual indications of airborne radioactivity, or air sample results indicate that the worker would not have been at risk of exceeding the 100-mrem investigation level without respiratory protection. Workers who use positive

(continued)

Example 5.2 (Continued)

pressure suits should undergo routine bioassay or have provisions for breathing zone air sampling within the suit.

Criterion 5: Long-term chronic exposure to air concentrations exceeding 2% of the DAC

This condition can apply to facilities that have low-level airborne radioactivity but do not meet the criteria for posting as airborne contamination areas. **Caution**: just because an area does not require posting as an airborne contamination area (that is, 10% of the *DAC*) does not mean that individual monitoring is not needed. Where routine air concentrations never exceed posting requirements but exceed 2% of the *DAC*, the need for individual monitoring should be based on potential stay times in those areas. Continuous (or significant) occupancy over a year would suggest individual monitoring is needed.

Criterion 6: Short-term chronic airborne exposure, or multiple acute airborne exposures

Criteria 6 and 7 may be particularly useful for addressing supervisory, walk-through, and inspection staff who do not actually handle or process radioactive material. The derived concentration threshold for individual monitoring (C_{air}, in terms of fractional *DAC*) can be calculated using an exposure fraction for the worker (f_w), as follows:

$$C_{air} = \frac{0.02 \cdot DAC}{f_w}$$

$$\text{where } f_w = \frac{\text{number of estimated exposure hours per year}}{\text{2000 hours per working year}}$$

Criterion 7: Tracking individual exposure in DAC-h

Individual work assignments and concentrations are tracked to determine cumulative exposure in *DAC*-h. Once a worker exceeds 40 *DAC*-h, bioassay should be performed (if feasible). This method implies the use of a *DAC*-h tracking log. Such a log might be continued for a worker over the course of a year and then zeroed out at the start of a new year. One issue to be resolved by the facility is what to do with *DAC*-h if the total never exceeds 40. For this situation, sites should still provide intake and dose assessments based solely on *DAC*-h.

Example 5.3. Circumstances Not Requiring Routine Individual Monitoring

- Radioactive materials are in a sealed source or special form.
- Radioactive materials are packaged in accordance with Department of Transportation specifications.
- Quantities of radioactive material in process are less than 2% of an *ALI*.

The requirement in 10 CFR 835 is that monitoring be provided for workers who are likely to receive a potential alpha energy exposure (*PAEE*) *above background* that would lead to a committed effective dose equivalent $H_{E,50}$ of 100 mrem in a year. The corresponding exposures are 0.08 WLM for ^{222}Rn progeny and 0.24 WLM for ^{220}Rn progeny. Monitoring would normally include breathing zone air sampling using lapel air samplers or etched-track detectors, or fixed air monitors with records of stay times.

Because of compelling special circumstances, a few contractors have been able to get regulatory relief under 10 CFR 820.62. The problem arises from the inability at these sites to distinguish between natural and occupational sources of radon and thoron exposure. At these sites, monitoring is provided for workers who are likely to receive a *PAEE including background* that would lead to a committed effective dose equivalent $H_{E,50}$ of 500 mrem in a year: this is 0.4 WLM for ^{222}Rn progeny and 1.2 WLM for ^{220}Rn progeny. Monitoring would normally include breathing zone air sampling using lapel air samplers or etched-track detectors, or fixed air monitors with records of stay times.

Both approaches to a monitoring are based on exposure, which includes both air concentration and amount of time breathing the air. It is important to point out that workers may be permitted to work in significant concentrations of potential alpha energy for short periods of time with no personnel monitoring, providing they don't exceed the likelihood of receiving an $H_{E,50}$ of 100 mrem or 500 mrem (depending on whether the site has obtained an exemption from 10 CFR 835 for radon).

5.4 SPECIAL BIOASSAY PROGRAM

Special bioassay should be initiated when off-normal conditions occur or there are indications that an intake needing assessment may have occurred. Criteria for identifying those conditions typically can include personal contamination, high air sample results, uncontrolled spread of contamination, or expressed worker concerns. The response to these conditions is the performance of bioassay measurements outside the envelope of routinely performed baseline, scheduled periodic, and termination or ending work measurements. The reason for and interpretation of these special measurements should be clearly identified. The role of special measurements is to confirm or rule out the initial indication of an intake, to determine the radiological significance of confirmed intakes, to indicate the need for work restriction or dose reduction therapy, and to begin the dose assessment process.

Special bioassay measurements may include the same types of measurements as those performed for routine monitoring (e.g., in vivo measurement, urinalysis) and may also include additional types of measurements (e.g., fecal analysis, wound counting).

Some criteria for initiating special bioassay can be found in the RadCon Standard (DOE 1999e). Typically, site technical basis documents or programmatic manuals provide additional guidance. The criteria of Example 5.4 should be considered more as qualitative guidelines than quantitative requirements. The decision to select any particular contamination level as a criterion for initiating special bioassay is highly subjective. For example, a hot particle on a shoe cover would not necessarily warrant special bioassay, even though the contamination level may exceed the alpha or beta/gamma contamination level shown above. Likewise, a single spot of contamination on the side of the face would be less likely to warrant special bioassay than substantially lower levels of contamination covering the mouth and nose area. While it is certainly conservative to perform bioassay when any of the listed criteria are exceeded, an excellent internal dosimetry program will factor in the unique aspects of each occurrence and exercise good professional judgment in prescribing special bioassay.

There are potential pitfalls in relying on some indicators as a basis for not performing special bioassay. For example, no detectable activity on nasal smears following a suspected inhalation does not necessarily mean that no intake occurred - a worker who has nasal congestion or is a mouth breather would not necessarily show activity detectable by nasal smears following an inhalation intake. Wounds involving alpha-emitting nuclides need special attention because the contamination could be completely shielded by overlying skin, tissue, blood, or serum moisture. Blood smears should be dried before counting with an alpha detector.

In some cases, workplace detection methods can be adequate to moderate the need for immediate bioassay measurements. For example, high-energy beta or photon emitters such as ^{90}Sr, ^{137}Cs, and ^{60}Co can be readily detected using portable Geiger-Müller (GM) survey meters. The typical sensitivity of such instruments is sufficient to determine the relative severity of a potential intake by a wound. If contamination is not detectable by these instruments at the time of the injury, then it is highly unlikely that there is any significant wound intake. This knowledge can permit a more relaxed approach to special bioassay, rather than precipitate a crisis response. Deciding the duration and extent of a special bioassay program also calls for professional judgement. It should be recognized that early excreta bioassay (collected earlier than 1 to 2 hours following the intake) will not necessarily reflect sufficient equilibrium to allow an accurate assessment of intake. Urine collected earlier than 1 hour after intake is likely to reflect the pre-intake condition. Likewise, feces voided within a few hours of an inhalation intake may be too early to have permitted passage of radioactivity through the gastrointestinal (GI) tract. In vivo measurements made shortly after intake may also reflect rapidly changing clearance. Residual external contamination on an in vivo bioassay subject is sometimes a problem near the time of intake. Thus, multiple bioassay measurements over several days following an intake provide a better tool for quantifying the magnitude than a single sample. These may include longer term measurements at weeks, months, and even years after an intake to accurately characterize the biokinetics and provide accurate intake and dose assessments.

5.5 TERMINATION AND ENDING-TASK BIOASSAY PARTICIPATION

When a worker completes an assignment requiring routine bioassay, an ending-task bioassay measurement is used to indicate the worker's status when the potential for further exposure has ended. Ideally, this measurement should be made as soon as the work assignment is completed. If the measurement is not made until employment is ended, then the measurement is actually an employment termination measurement and documents the status of the worker when no further occupational exposure under that employer will occur. Ideally, the termination measurement would be performed on the last day of employment. The need for both ending-task and termination samples is a matter of company policy. If ending-task measurements are performed and the cognizant radiation protection organization is confident

Example 5.4. Criteria for Commencing Special Bioassay

Special bioassay should be initiated if any of the following criteria are met (Fauth et al. 1996, Carbaugh et al. 1994a):

- Nasal or mouth smears, nose blows, or sputum samples that indicate above background levels of radioactivity
- Any contaminated wound
- Contamination on protective clothing in excess of 10,000 dpm-alpha or 100,000 dpm-beta/gamma per 100 cm^2 if no respiratory protection is in use
- Unplanned spread of contamination on accessible surfaces in excess of 1500 dpm-alpha or 15,000 dpm-beta/gamma per 100 cm^2 if no respiratory protection is in use
- Any detectable general facial contamination in excess of 200 dpm-alpha or 4,000 dpm-beta/gamma per 100 cm^2
- Detectable contamination on the skin, other than the facial area, in excess of 1000 dpm-alpha or 100,000 dpm-beta/gamma per 100 cm^2
- Detectable contamination inside a respirator after its removal
- Acute exposure to 40 *DAC*-h after incorporating any respiratory protection factor
- Any unplanned suspected intake

that no further potential for intake existed, then an employment termination bioassay is probably not needed. For practical purposes, the ending-task measurement may be considered the release of a worker from requirements for further bioassay.

5.6 BIOASSAY FOR DECLARED PREGNANT FEMALE WORKER

DOE has published an implementation guide on *Evaluation and Control of Radiation Dose to the Embryo/Fetus Guide* (DOE 1999a). All relevant parts of this document should be used in design and operation of the parts of a bioassay program that apply to declared pregnant female workers. This Technical Standard does not summarize the recommendations of that implementation guide but does note a few points about internal dosimetry. The dose limits for a declared pregnant worker's embryo-fetus is substantially more restrictive than those for radiological workers, except for the fact that the 500-mrem limit applies to the dose equivalent for the nine-month gestation period, and not the committed dose equivalent for 50 years following intake. The maternal uptakes that would cause a 500-mrem gestation period dose to the embryo-fetus are in the nominal microcurie range (e.g., approximately 1 μCi for ^{238}Pu, 10 μCi for ^{137}Cs, and 50 μCi for ^{90}Sr, based on Fauth et al. 1996). Routine bioassay programs designed to monitor workers should be easily adequate to demonstrate compliance with the embryo-fetus dose limits. As a verification, it may be desirable to obtain a special bioassay upon receipt of a pregnancy

declaration, with a follow-up special bioassay at the conclusion of pregnancy if the worker continues to be exposed to possible intakes. Sikov and Hui (Sikov et al. 1996) provide methods for embryo-fetus internal dosimetry.

5.7 CONFIRMATORY BIOASSAY PROGRAM

A confirmatory bioassay program involves limited surveillance of workers to provide verification that routine bioassay is not required. As described by ICRP Publication 54 (ICRP 1988), confirmatory monitoring programs are qualitatively useful to show that results are as expected. Any unexpected results warrant special investigation and may suggest the need for a routine monitoring program. A confirmatory bioassay program for a work group having low potential for significant intake may involve sampling a small fraction (e.g., 10%) of the group at a relatively constant rate over a 1-year period. Confirmatory bioassay programs should not be interpreted in terms of minimum detectable dose. This type of program is particularly suited for radionuclides which are easily detected at low levels relative to levels of concern.

5.8 TIMELY RECEIPT OF BIOASSAY RESULTS

Bioassay measurement results should be provided in a manner timely to the purpose for which they are obtained. Factors to consider in determining timeliness include:

- use of results to implement or determine efficacy of dose reduction therapy
- use of results for preliminary assessments for rapid reporting to the worker and management and for determining appropriate follow-up activities
- need to confirm a suspected intake based on a high routine measurement before detection capability is lost due to normal biokinetics
- trade-offs in sensitivity (due to analytical short-cuts and reduced counting times) for rapid results.

Because in vivo measurement data is usually available almost immediately upon completion of the measurement, the response times discussed in this section will generally apply to excreta bioassay measurements.

Confirmatory bioassay measurements are not expected to show any significant detection of nuclides of concern. Since the purpose of these measurements is merely to provide general information that significant intakes are not occurring and that radiological controls are effective, the time between obtaining a bioassay sample (or measurement) and receipt of the results need not be rapid. Likewise, where routine periodic measurements are not likely to show significant intakes with regard to dose control and work administration, a 1- or 2-month analytical response time is not likely to have any significant impact. Generally speaking, a 1-month turnaround time for routine excreta sample analysis does not pose serious problems for either analytical laboratories or worker monitoring programs.

Special bioassay measurements should have much faster response time. This is particularly important if the results are being used to determine need for, or efficacy of, dose reduction therapy. Rapid availability of special results is also needed for preliminary intake and dose assessments used to classify intakes for reporting purposes. It is suggested that some kind of preliminary bioassay measurements should be available within 24 to 48 hours following intake. The need for precision and

accuracy in these early assessments is much less than for the measurements which will be used for the final dose assessment.

Provisions for assuring that a worker has received the appropriate in vivo measurements or has provided the scheduled excreta sample should not be overlooked in designing a program. A reasonable grace period is appropriate to deal with workers who forget to submit excreta samples or who are unable to meet the schedule. For some routine sampling frequencies, a grace period of 30 or 60 days may be appropriate. However, administrative actions (e.g., work restriction) may be appropriate for a worker who is substantially overdue for measurement.

Ideally, results of new-hire or baseline measurements should be available before a worker commences the work requiring the bioassay. This prevents loss of baseline information if a sample is lost during analysis. However, loss during analysis tends to be a rare occurrence, and it is an acceptable practice to begin work once the sample has been collected but prior to receipt of results.

Where air sample results form the basis for identifying intakes and making preliminary dose assessments, some kind of initial results (e.g., gross alpha or gross beta concentration) should be available within a few hours of obtaining the sample. This is particularly important for samples used to monitor for unknown or changing work conditions. Routine air samples for well-established processes and facilities may have longer turnaround times (e.g., as much as a few days), provided they are not the sole method of detecting off-normal workplace conditions.

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6.0 DETECTION AND CONFIRMATION OF INTAKES

Two fundamentally different kinds of signals may indicate the possibility that intakes of radionuclides have occurred. Most often, possible intakes may be indicated by workplace monitoring results (CAM alarms, survey and frisking results, air sample results) or observations (an accident, explosion, spill, leak, equipment failure). Possible intakes are more rarely signaled first by unexpected, elevated bioassay results. In some events, there is no question that intakes were possible, so special bioassay procedures and investigations are initiated to confirm or rule out intakes. In situations where the possibility is less clear, the suspicion of an intake should be investigated, that is, efforts should be made to confirm or rule out intakes, if preliminary results indicate the possibility for a significant dose. If preliminary results do not indicate the possibility of a dose above the IL of 100 mrem, then a dose may simply be assigned without investigation.

A suspected intake based solely on workplace monitoring data cannot be confirmed in the same sense that repeated bioassay measurements can confirm an intake of radionuclides that can be detected by bioassay. There are, however, some checks that can be used to help validate the result. This is particularly important for larger predicted intakes. For example, one can look at coworker BZ data, evidence of concomitant external contamination, job-specific air monitoring information, and results of nasal smears. None of these sources provides confirmation, but collectively they can sometimes help flesh out the details of the exposure.

6.1 USE OF WORKPLACE MONITORING DATA FOR DETECTING AND CONFIRMING INTAKES

The identity of radionuclides inadvertently taken into the body and the amount of intake may be inferred using workplace monitoring data (e.g., airborne contamination concentration measurements, nasal-smear activity measurements, application of resuspension factors to measured surface contamination levels, etc.). Airborne radioactive material concentration data may be used as a direct indication of intake, especially if information on particle size distribution can be obtained. Evaluation of other workplace indicators proved to be useful in identifying possible intakes. However, there is no generally accepted quantitative method for correlating such indicators with intake. Heid and Jech (1972) concluded from review of several plutonium inhalation cases that the amount of activity on a nasal smear collected shortly after intake was about the same as the amount deposited in the deep lung for nose breathers and about half the deposited activity for mouth breathers. Brodsky (1980) suggested that a resuspension factor could be applied to surface contamination levels to assess the corresponding airborne contamination levels. Due to the provisional acceptance of dose assessments based on workplace monitoring data, detailed methods are not described here. Where use of such data appear to be appropriate for dose assessment, the facility should establish a protocol for their use as part of the internal dosimetry program, and document it in the technical basis documentation.

6.2 USE OF BIOASSAY DATA FOR DETECTING AND CONFIRMING INTAKES

According to the *Internal Dosimetry Program Guide* (DOE 1999b), intakes of radioactive materials that are suspected on the basis of a single bioassay measurement must be confirmed by one of several means. "False alarms" based on erroneous bioassay results carry a heavy penalty in terms of cost, paperwork, and public relations for both DOE and its contractors. The decision to confirm an intake based on bioassay measurements currently uses a statistical comparison of one or more results with an

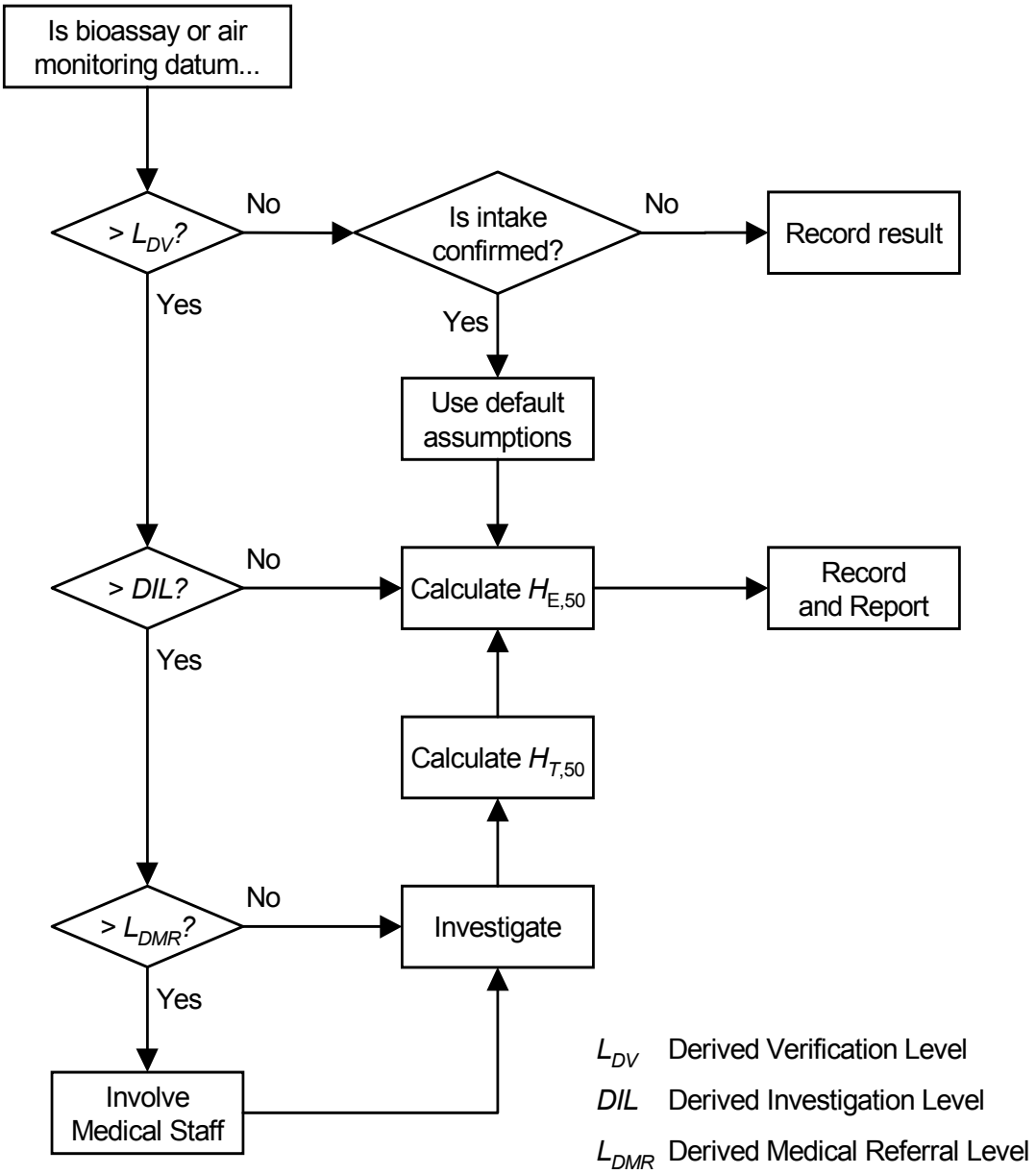

Figure 3. Reference Levels for Interpreting or Responding to Intake Monitoring Results

appropriate blank. Guidance from a variety of sources (including HPS N13.30-1996) uses the concept of an appropriate blank for comparison with analytical measurements such as those that form the basis for bioassay measurements. In fact, however, two distinct decisions are confounded by the current method: the first is the decision whether radioactivity above background levels is present, and the second is a decision whether any radioactivity that is present is above that which would be expected from non-occupational exposures, as explained in the IDG. For example, it is well known that environmental exposures to natural uranium occur, and that these have been mistaken for occupational exposures.

6.2.1 Decisions Based on Individual Monitoring Data

Examples of actions taken following acquisition of a result from an individual monitoring program are shown in Figure 3. This approach is useful to consider, with individual sites determining the values of their respective reference levels.

6.3 STATISTICAL METHODS FOR CONFIRMING THAT AN INTAKE HAS OCCURRED

Beyond the methods described in the IDG, at least two other statistical methods exist for confirming that an intake has occurred. The first is to simply pool the *n* bioassay results statistically to achieve the $1/\sqrt{n}$ improvement in the decision level (Hickey et al. 1993).

The second is to employ Bayesian statistical inference (Miller et al. 1993, 1995), based on Bayes' theorem (Lindley 1972, 1980, 1985; Martz and Waller 1982; Calvin 1989; Press 1989). The Bayesian formalism is attractive because it incorporates prior knowledge in addition to the results of a given measurement, and it results in a distribution of likely outcomes rather than merely a point estimate with an uncertainty. However, the method has been criticized as being too subjective. At present, the DOE and this Technical Standard have taken no official position on the use of the Bayesian method. The appropriateness of Bayesian methods must be decided on a case-by-case basis.

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7.0 INTERNAL DOSE EVALUATION

Radiation protection guides are expressed in terms of limiting values of dose to workers. As summed with deep dose equivalent, 10 CFR 835.202 limits committed effective dose equivalent for individuals and committed dose equivalent for their organs and tissues. Committed effective dose equivalent and committed organ dose equivalent are calculated for intakes in specific calendar years to evaluate conformance with limiting values for occupational exposure and for reporting doses to workers. A committed effective dose equivalent is calculated (1) to evaluate conformance with limiting values for control of the workplace, (2) to measure the effectiveness of the facility's radiation protection programs, and (3) to provide a summary to the worker of the dose equivalent that may be received in subsequent years as a result of any intake during the calendar year. The need may also arise to calculate doses over other time periods such as from the date of intake to the first year following the intake, to the date when the person would turn age 75 (i.e., "the expected lifetime dose"), and to the date of death.

There are three conceptually distinct methods to assess internal dose:

- assessment of intake directly from air samples or other workplace data, followed by the assessment of dose from intake
- assessment of intake from bioassay data and biokinetic models, followed by the assessment of dose from intake
- direct assessment of dose time-integrated retention from bioassay data, with assignment of a putative intake that is consistent with the dose.

Assessments of internal dose using mathematical biokinetic models should be based, as appropriate, on

- direct, in vivo measurements of a radionuclide(s) in various source organs of the body; or
- indirect, in vitro measurements on excreta.

If bioassay data are not available or are of questionable value, assessments of inhaled radionuclides should be based on workplace data, preferably on air sample measurements. The initial assessment of a radionuclide intake or retained quantity may be based on air monitoring or other workplace measurement data as well as available bioassay measurement data. However, assessments based only on workplace monitoring data should be regarded as provisional and should be updated if and when bioassay measurement data of sufficient quality become available. Evaluations of dose equivalent resulting from an intake of a radionuclide proceed from an assessment of the amount of the radionuclide in organs and tissues of the body as a function of time. The radionuclide distribution and retention depends on the physical and chemical forms of the radionuclide, its radiological properties, the physiological characteristics of the individual, the route(s) of intake, and the magnitude of intake(s).

Except for radon, thoron, and their short-lived progeny, internal dose equivalent is defined in terms of the energy imparted to target tissues from the radiations emitted by radionuclides in source organs and tissues of the body. The purpose for analyzing radionuclide intake and retained quantity as a function of time is to identify the organs and tissues in which the radionuclide is deposited and to evaluate the cumulated activity (e.g., transitions in Bq·s or μCi-days; 1 μCi-day = 3.1968E9 transitions) in source organs. Since it is often difficult to precisely determine the cumulated activity in all source organs directly from bioassay measurements, biokinetic models have been developed to describe empirical relationships between intake, number of transitions, and bioassay measurement values.

Bioassay and other supporting data can often require considerable expense and effort to obtain. It is neither necessary nor cost-effective to assess all intakes using the same level of effort; rather, it is more reasonable to employ a graded approach to bioassay collection and dose assessment whereby the level of effort expended on the assessment increases with the magnitude of the anticipated dose. Minor exposures may be assessed using generalized biokinetic models for a reference individual and conservative (or default) assumptions regarding the nature of the exposure and characteristics of the contaminant. The generalized model and assumptions should be based on previous experience or supporting studies at the facility or models recognized by ICRP or NCRP. The facility should document the default models and assumptions and when these are appropriate for use. For projected doses of increased magnitude, sufficient bioassay and source characterization data should be obtained to enable adjustments to be made to the generalized models, as appropriate, to account for the specific behavior of the radionuclide(s) in the body. The facility should establish and document specific dose levels which require enhancement of data collection and individual specific dose assessment efforts.

7.1 DOSES TO BE ASSESSED

10 CFR 835 requires that the following doses be calculated:

- committed effective dose equivalent from intakes occurring during the year
- committed dose equivalent to tissues of concern from intakes occurring during the year
- total effective dose equivalent
- cumulative total effective dose equivalent.

The RadCon Standard also recommends the calculation of "lifetime occupational dose," which is taken to be the same as cumulative total effective dose equivalent.

7.1.1 Committed Effective Dose Equivalent

All confirmed occupational intakes, above the decision level, should be assessed. Based on each assessment, the committed effective dose equivalent $H_{E,50}$ should be calculated for each intake during the calendar year.

Where there are multiple intakes or where several radionuclides are involved, each facility may establish a per-radionuclide or a per-intake minimum assessment value so that the intent of the above recommendation is met. In practice, it is not necessary to record the contribution from a radionuclide or a specific group of radionuclides (when their respective source terms are independent and the measurement system provides discrimination) that contributes less than 1-mrem committed effective dose equivalent.

7.1.2 Committed Dose Equivalent to Tissue of Concern

Each facility should identify the tissues of concern relative to radionuclides at the facility and should justify and document the selection in the technical basis document. Specific requirements for identifying the organs of concern are given in the definition of weighting factor in 10 CFR 835 . Wound site tissue and associated lymph nodes should be excluded from committed dose equivalent calculations (Nénot and Stather 1979; National Research Council 1988).

The committed dose equivalent $H_{T,50}$ to the tissue(s) of concern should be calculated for those years where a committed effective dose equivalent is calculated.

For exposures to the short-lived progeny of radon and thoron, $H_{lung,50}$ may be calculated as $H_{E,50}$ divided by the tissue weighting factor for lung, $w_{lung} = 0.12$.

7.1.3 Total Effective Dose Equivalent

Total effective dose equivalent should be calculated in cooperation with the site's external dosimetry program and records program pursuant to 10 CFR 835. Total effective dose equivalent includes all occupational doses: internal, external, and those received at other sites.

7.1.4 Cumulative Total Effective Dose Equivalent

Committed doses from intakes prior to January 1, 1989, may be included in lifetime dose calculations. Including such doses gives a more accurate estimate of the lifetime accumulation and is consistent with the recommendations of NCRP Reports 91 and 116 (1987 and 1993). However, to demonstrate compliance with 10 CFR 835 requirements, these doses should be kept separate from the cumulative total effective dose equivalent from intakes occurring after January 1, 1989.

7.2 DATA NEEDS AND DEFAULT ASSUMPTIONS

Generally, the more data available, the more precise the dose determination. However, practical considerations generally limit the amount of data available. Internal dosimetry programs should commit resources in proportion to the magnitude of potential doses. For doses below the *IL*, it is acceptable to use default assumptions as described in the technical basis documentation.

7.3 INTERPRETATION OF BIOASSAY DATA

Selection of methods for bioassay interpretation plays an important role in the design of the bioassay program. For example, in cases where either the intake scenario or the biological retention cannot be well known, more bioassay data are needed to adequately arrive at the dose estimation. Conversely, if the intake, uptake, and retention models are well characterized and apply to the exposure scenario, one bioassay measurement which confirms a previous result may be sufficient for dose assessment. Since there is normally sufficient uncertainty in both the bioassay data and the biokinetic models, the use of multiple data points and fitting to the model may be necessary. Facility-specific and radionuclide-specific decisions about bioassay interpretation methods should be documented and should dictate a significant part of the overall bioassay and internal dosimetry program.

The derivation of intakes and retained quantities from bioassay data may be the critical step in the dose assessment process. Evaluations of exposure to internal radionuclides should account for all possible sites of retention and their associated retention times (if known) in the body. Generalized biokinetic models, suitably modified to account for experience or studies at the facility, may provide a starting point for the initial assessment of an intake and for determining the specific needs for follow-up bioassay measurements. All organs contributing to the effective dose equivalent, calculated with the weighting factors given in 10 CFR 835, should be considered rather than only those organs in which the radionuclide can be readily measured.

7.3.1 Direct Estimation of Retained Quantity

When thorough bioassay histories are attainable, and good confidence can be placed on organ and whole body radionuclide content evaluations, it is possible to explicitly derive the retained quantity and retention history of an exposure without resorting to use of default parameters. In some cases the uncertainties associated with the biokinetics are much greater than the uncertainty in the direct assessments of intake and retained quantity.

A tritium exposure with sufficient urine assay data to document the biological excretion rate is an example of using excretion history. Due to uncertainty in the route of intake (e.g., skin absorption versus inhalation) and in the biological clearance rate (which depends on water consumption), the tritium excretion history provides the best assessment of the number of transitions and, thus, the dose equivalent. Similarly, a radioiodine exposure, well documented in time and monitored by in vivo thyroid counting, can be assessed directly from the bioassay result. In both cases, discrete or parameterized methods of summation of transitions in the well-known source organs will provide sufficient information for dose assessment.

Where direct uptake and retention history are used for dose assessment, the method for converting data to dose equivalent should be documented as part of the dose assessment. However, if the bioassay data are insufficient for a thorough assessment of retained quantities, or are of such poor quality that whole body or pertinent organ content cannot be directly derived, then biokinetic models should be used.

7.3.2 Biokinetic Modeling

A biokinetic model is a time-dependent mathematical representation of the relationship between intake, uptake, retention, translocation, and excretion for radionuclides taken into the body. Models differ in their scientific approach and mathematical formalism. Some models, such as systemic uptake excretion models, are empirically derived from studies of radionuclide behavior in humans or animals. Other models are derived from considerations of the fundamental physiological and biochemical processes of the body.

The application of biokinetic models for internal dosimetry has been described by the NCRP (NCRP 1985a, 1985b), the ICRP (ICRP 1968, 1969, 1973, 1979a, 1979b, 1980a, 1980b, 1981a, 1981b, 1982a, 1982b, 1982c, 1986b, 1988, 1989b, 1993b, 1994a, 1994b, 1995), and many distinguished authors (Avadhanula et al. 1985. Cabello and Ferreri 1993; Calvo and McLaughlin 1995; Carbaugh et al. 1989; Chang and Snipes 1991; Fauth et al. 1996; French et al. 1996; Hill and Strom 1993; Inkret and Miller 1995; Johnson and Carver 1981; Johnson and Myers 1981; Lawrence 1978; Lessard et al. 1987; Skrable et al. 1994b; Sula et al. 1991).

7.3.2.1 Selection of Biokinetic Models

Biokinetic models at a facility should be documented and used consistently. If an exception to the documented model is appropriate, the alternative method should be justified and documented in the dose assessment. Normally, models developed or endorsed by the DOE, ICRP, NCRP, or ANSI should be used. Limitations of these models should be recognized, and the models should be used for their intended purpose. (An example may be the use of biokinetic models in the ICRP Publication 30 series that describe the retention of radionuclides in the body. The ICRP models generally employ linear first-order kinetics to simplify the mathematical representation, ignoring recirculation between organs and the systemic compartment. Models that have been developed from empirical excretion functions or those that

incorporate feedback from organs to the systemic circulation may be more appropriate for interpreting excretion data.)

Biokinetic models used for intake, uptake or retention assessment should be appropriate for the following conditions:

- intake mode (e.g., by ingestion, inhalation, or injection)
- duration of the intake (e.g., acute, continuous, or intermittent)
- time period of interest
- sites of uptake and retention
- workplace conditions
- intake radionuclide and its progeny.

Biokinetic models should relate well to the available bioassay data, should account specifically for the chemical and physical characteristics of the contaminant, and should account for the influence of decorporation therapy if used. Models, default assumptions, methodologies, and computer codes used for assessments of doses from intakes should be recorded and maintained.

7.3.2.2 Selection of Intake Default Assumptions

Many different factors influence the resulting distribution, retention, and excretion of radionuclides following an intake. The following default assumptions should be applied to assessments of intakes and their resulting doses unless more appropriate values are available. As discussed in the introduction to Section 7, there are levels of intake and dose that make it more appropriate to determine values or parameters more accurately and realistically.

Entry Pathway and Duration of Intake. If the intake mode is not known, acute inhalation should be assumed. Acute inhalation represents the most common type of occupational intake. This assumption will tend to maximize the committed effective dose equivalent evaluated from bioassay data.

Time of Intake. If it is not reasonably possible to establish the time of an intake identified by a routine bioassay measurement, it may be assumed that the intake occurred at the midpoint of the period during which it could have occurred (ICRP 1982), or the time at which the expectation value of intake would have occurred (see Section 7.4.1.3). The midpoint is usually the date halfway between the sample from which an intake was detected and the previous routine bioassay measurement. If no prior sample exists (baseline result) or if a baseline bioassay measurement exceeds the decision level, effort should be expanded to examine the person's previous work history, in an attempt to assign an intake date.

Particle-Size Distribution. The particle-size distribution influences the probability of aerosols depositing in the nasopharyngeal, tracheobronchial, and pulmonary (parenchymal) regions of the lung. Particle size also influences the relationship between lung deposition, retention times, and excretion rate. Therefore, assessments of quantities retained in lung and assumptions regarding lung clearance should be determined using direct lung counting, wherever possible.

When lung counts cannot be used to determine the activity retained in lung, assessments may be made from urinalysis or fecal analysis data. The ICRP Publication 30 (1979) model for the respiratory tract shows that deposition in the pulmonary region will vary by a factor of about 3 over a range of activity median aerodynamic diameters (*AMADs)* between 0.3 - 3.0 μm. In the absence of specific information on particle size, a particle size distribution with an *AMAD* of 1 μm and the associated deposition fractions should be assumed. If the newer respiratory tract model is used (ICRP 1994a),

particle sizes may need to be characterized by their activity median thermodynamic diameter *(AMTD)* for small diameters for which diffusion behavior predominates. Note that use of respiratory tract models other than ICRP 30 must be justified in the program's technical basis documents.

Lung Solubility and Transportability. Decisions regarding transportability of radionuclides from the lung should be documented and justified in the dose assessment. The transfer rate of a radionuclide from the lung across to other regions of the body is dependent on the physical and chemical forms of the radionuclide and its host aerosol, and on the biokinetic characteristics of the subject. These characteristics of internally deposited radionuclides can be inferred from bioassay measurements, when available. If bioassay measurements are not available or are not complete, these characteristics should be estimated from the general chemical form of the radioactive material and information given in the ICRP Publication 30 series (1979-1982). If there is no basis for specifying the chemical form, then conservative estimates based on the range of values provided for the radionuclide in ICRP publications should be used. For example, the choice of a class Y material for inhalation of uranium compounds would result in a maximum committed effective dose equivalent per unit intake. In contrast, the choice of a class D material for inhalation may result in a maximum value for the dose equivalent to bone surfaces. It would be wise to study in advance the solubility classification to be assigned to radionuclides commonly encountered in the workplace.

Transportability classes that differ from the ICRP models have been observed. At the Y-12 plant in Oak Ridge, a combination of class W and class Y uranium has been observed (Forrest and Barber 1993; Barber and Forrest 1995). This combination has been called "class Q," for "quarterly". Similarly, material that clears more slowly than class Y has been observed and termed class Super-Y (Sula et al. 1991).

Occupational exposures may involve mixtures of radionuclides with various abundances and physical and chemical compositions. These radionuclides may be contained in a host matrix with characteristics that determine the actual solubility or transportability of most or all of the radionuclides in the mixture. Prior experience or studies for specific exposure conditions are the best means for determining the presence of and behavior of individual radionuclides in such mixtures.

Radioactive Progeny. Radioactive progeny produced by the decay of retained quantities should be modeled separately from the parent if the systemic retention and biokinetics for the progeny radionuclide are well known and if the physical half-life of the progeny is long enough to make a dosimetric difference. Otherwise, the progeny should be assumed to be distributed and retained as the parent radionuclide. It is particularly important to model radioiodines and noble gases separately from parent radionuclides for internal dose assessment (ICRP 1979a).

7.3.3 Details of the Actual Dose Assessment

After the approach (direct or model) has been selected, intake or uptake can be assessed from various bioassay results. An objective best fit of the predicted to observed bioassay measurement results should be made. Documentation of the data, assumptions and methods used should be included in the dose assessment. If alternate methods result in different results, the bases for reaching the decision on the accepted result should be documented and should be reviewed by a second qualified dosimetrist (either within the organization or outside).

7.3.4 Curve Fitting (Weighting of Data)

Assessment of doses from the intake of radioactive material almost always involves "curve fitting." An operational upset or a routine bioassay result above the verification level, L_V, often lead to several follow up samples. All of these samples are considered by the dosimetrist in assessing an intake or a dose. The usual practice is to do a regression (sometimes called "fitting a curve to the data.) To do a regression one must have a weighting factor for each data point. The optimal choice of weighting factors in regressions of bioassay data requires the analyst to:

- clarify the goal desired
- choose the methods to achieve that goal
- select the parameters to be adjusted, and
- consider the overall ensemble of information that is available.

The information presented in the balance of Section 7.3.4 and subsections, is to assist the dosimetrist in choosing the best way to assign weighting factors. Often weighing factors must be determined on a case-by-case basis with considerable exercise of professional judgement. There is no appropriate, standard, "one-size-fits-all" methodology. The fuller the understanding of the weighting issues the analyst has, the more appropriate will be the choices of weighting factors for bioassay data used in the regression models. A dose assessment should identify and document the most important factors affecting the choice of weighting factors.

Choice of methods for fitting bioassay data to a model leads to different results with different assumptions (McWilliams et al. 1964; Fauth et al. 1996; Traub 1994; Strom 1992. Skrable et al. 1994a; Inkret and Miller 1995; Chang and Snipes 1991). The basics of weighted regressions are found in Draper and Smith (Draper and Smith 1981). Skrable has illustrated the pitfalls and inaccuracies that are inherent in using unweighted least squares fits (Skrable et al. 1994a), despite the fact that they are endorsed by the NRC (NRC 1993a). More than three decades ago, McWilliams, Furchner and Richmond showed that dramatically different results are obtained with uniform weighting of data compared with uniform weighting of the logarithms of the data (McWilliams et al. 1964). Uniformly-weighted or "unweighted" regressions are the result of ignoring the question of weighting altogether. Excellent explanations of the various methods are found in technical basis documentation of the Savannah River Site (Fauth et al. 1996) and the Mound Laboratory (Traub 1994). Some computer codes permit a choice of weighting factors (Kennedy and Strenge 1992; Skrable et al. 1994a). The choice of Bayesian statistical methodologies, in a sense, is a choice of weighting methodologies (Miller et al. 1993, 1995; Inkret and Miller 1995).

Strom has suggested that consideration be given to methods other than simply inverse-variance weighting, since there are other kinds of knowledge about data (Strom 1992).

The choice of weights depends on the desired goal, the choice of method to achieve the goal, the selection of adjustable parameters, and the optimal use of the information that is available. Choices of goals include the maximum likelihood estimator (MLE) of dose, the MLE of intake, the best overall determination of a biokinetic model, or some other endpoint. Two fundamental methods of achieving a given goal are intake assessment and direct dose assessment from first principles. Parameters to be adjusted should be selected from a list including value of intake, time course of intake, mixture of chemical forms, and rate constants. Finally, optimal use of available information requires considering variance in the measurement process, biological variability, unintended number weighting, and other objective or subjective weighting.

The goal of a regression may be the best assessment of intake, dose, model parameters, or something else. Computation of weighted averages of intake from ratios[3] differs in terms of weighting from direct regression of a retention function or an excretion function. Regression to predict intake differs from regression to predict dose; the best assessment of intake may not be the same intake that gives the best assessment of dose.

Because regressions differ when goals differ, weighting for the MLE of dose may differ from weighting for other choices of estimators, such as the MLE of intake or the best model for predicting later bioassay data. Furthermore, regression differs when it is done to excretion data rather than retention data. Excretion data (e.g., urine or fecal data) represent the first derivative of a retention function, while retention data (e.g., a lung count) represent the retention function itself.

Data taken from later times represent radioactive material that has been in the body a long time and that would have emitted more energy than did the activity already eliminated from the body. Therefore, the dose per unit activity is an increasing function of the time the activity has been in the body. The relative contribution of a data point to the assessment of dose (in contrast to its influence on quantifying the intake or defining the excretion function) may need to be considered. The MLE of dose is related to, but generally not directly proportional to, the following product: [activity excreted per unit time at time t] $\times$ [t]. The MLE of the intake is related to the t=0 intercept of an intake retention function. Different weighting factors may be needed for the two different MLEs. Thus, the amount of dose represented by a data point long after intake may be *relatively* greater than the amount of dose represented by data points occurring soon after intake. This kind of weighting is currently done by experienced analysts by simply ignoring or throwing out early data (i.e., these data are given a weight of zero).

The selection of the method for dose assessment affects consideration of information available to the internal dose assessor. Two methods can be identified:

1. *Assessing Intake.* The first method is to use bioassay data to assess the intake by a given route, multiply the intake by 5 rems, and divide it by the stochastic Annual Limit on Intake (*SALI*) for that route and chemical form. Essentially equivalent approaches are to use the "committed dose equivalent per unit intake" factors from Federal Guidance Report 11 (Eckerman et al. 1988) or the ICRP Publication 30 "weighted committed dose equivalent to target organs or tissues per intake of unit activity" factors. The intake assessment approach is essentially computing a weighted average intake from ratios of bioassay data to values of a fixed-parameter biokinetic model such as is done in CINDY (Kennedy and Strenge 1992) and in NUREG-4884 (Lessard et al. 1987).

2. *Assessing Dose from First Principles.* A second approach is to start from basic principles, employing bioassay data to infer the number of transitions occurring in organs or tissues of interest, employing absorbed fractions for energy emitted, using quality factors, and, finally summing committed dose equivalent values over the body. Bioassay data can be used to assess parameters of a variety of intake retention functions, including excretion functions, that may be used to infer the number of radioactive transitions that have or will occur.

Both methods share the foundation of a biokinetic model with at least one adjustable parameter.

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[3]In this context, "ratios" refers to bioassay measurements X_i observed at times t_i divided by intake retention functions IRF[t_i] for the appropriate bioassay compartment.

Another issue to consider in weighting regressions is the selection of what parameters are to be adjusted.

1. *Intake.* An estimate of intake is usually part of a regression analysis, but intake may already be known when radionuclides are administered medically or under experimental conditions.

2. *Time Course of Intake.* The regression may include *time of intake* or *time course of intake* (for multiple or chronic intakes) for optimization when these times are unknown. For single exponential intake retention functions, time of intake cannot be determined from bioassay data, but for other functional forms, it may be determinable if data are of adequate number and quality.

3. *Mixture of Chemical Forms.* The regression may choose an optimum linear combination of inhalation classes or chemical forms.

4. *Particle Size Distribution.* The regression may choose an optimum particle size distribution that best fits the data.

5. *Rate Constants.* Other parameters, such as rate constants used in the biokinetic models, may be optimized for individuals by the regression.

Optimal use of the information available dictates that once a method has been selected, at least four categories of information should be considered. Two relate to the measurement value itself; two relate to maximizing the use of other information that may be available. The discussion below applies to a general nonlinear regression of a function with more than one adjustable parameter.

There are two components of variance for a measurement result itself:

1. *Measurement-process variance* (e.g., *net* Poisson uncertainty or net fluorimeter uncertainty) depends on the amount of analyte present. In general, the relative standard deviation (coefficient of variation) becomes larger as the net activity or amount becomes smaller. Inverse variance weighting (i.e., computing the weighted sum of squares of deviations from the regression by multiplying each by $1/s_i^2$) is appropriate for this component of variance.

2. *Biological variability* is likely to be a fixed $\times\!\!\!

weighting include suspected skin contamination in the case of a chest count, difficulties in the analytical laboratory technique, suspicion of unintended or deliberate contamination of samples, suspicion of interference from other radionuclides, interference from prior intakes, interference from intakes of a different solubility class, differing types of analysis for similar samples (e.g., fluorimetry vs. mass spectrometry), etc.

Minimizing sums of squares of *ratios* of data to prediction is essentially minimizing sums of squares of fractional deviations (i.e., constant geometric standard deviation [GSD]). This is the method advocated in NUREG/CR-4884 (Lessard et al. 1987) and used in CINDY (Kennedy and Strenge 1992), discussed on the previous page.

Currently, internal dosimetrists may use an all-or-nothing subjective weighting (i.e., they ignore the data point) based on knowledge or a hunch that a point is an outlier. In particular, current methods provide maximum likelihood estimators of intake, rather than maximum likelihood estimators of dose. Under this proposal,

$$w_i \propto \left(\frac{d\hat{H}_{E,50}}{dy_i} \right)^2 , \tag{8}$$

that is, among other factors, weighting should be proportional to the square of the derivative of the estimated 50-year committed effective dose equivalent $\hat{H}_{E,50}$ with respect to the data point y_i in question.

If the $\hat{H}_{E,50}$ is simply calculated from an intake, then this leads nowhere. If $\hat{H}_{E,50}$ is calculated from a time-weighted intake, I_i, then the derivative in Eq. (8) becomes proportional to the integral of the *IRF* from the midpoint of the time interval $t_{i-1} - t_i$ to the midpoint of the time interval $t_i - t_{i+1}$. If data points are sparse in time, then small values have a large impact on $\hat{H}_{E,50}$.

7.4 CALCULATION OF INTERNAL DOSE FROM BIOASSAY DATA

A good practical summary of issues in calculation of internal dose from bioassay data is given by Carbaugh (1994).

A comprehensive method for calculating dose equivalent from intakes of radionuclides is presented in ICRP Publication 30 (1979). These concepts should be considered an acceptable standardized approach for use with this performance standard even though they were developed for deriving annual limits on intake (*ALI*), which are prospective limits used for the design and operation of facilities. The ICRP concepts may be used to calculate effective dose equivalent over any time of interest to an individual after an intake of radioactive material.

In specific cases, it may be more appropriate to apply dose assessment methods other than those provided in ICRP Publication 30. This should be acceptable provided the dose assessment methods are documented and justified.

DOE's 10 CFR 835 specifies the weighting factors and quality factors to be used in dose assessments, and also discusses the remainder organs to be used in a dose assessment.

7.4.1 Time or Time Course of Intake

Inference of dose from bioassay data requires a known or assumed time of intake or time course of intake. In most cases, the time at which an acute intake occurred is either known from observation or workplace information, can be determined from a variety of factors, or at least can be limited to a small enough period of time so that bioassay data can be unambiguously interpreted. For chronic or repeated intakes that occur between bioassay measurements, the pattern in time becomes more problematic.

7.4.1.1 Time Course of Intake to Be Assumed When There Is No Workplace Evidence

If bioassay results indicate that an intake has occurred, but there is no workplace or other evidence of an intake, then there are several possibilities:

- a non-occupational intake
- a deliberate intake
- an undetected acute occupational intake
- an undetected chronic occupational intake
- more than one undetected occupational intake
- accidental or deliberate contamination of bioassay samples
- error in or sabotage of radiobioassay analytical results
- bioassay results have been erroneously associated with the wrong individual.

Each of these possibilities has occurred in the human experience with intakes of radioactive materials.

In the rare case when there is no evidence of when an intake occurred, it is permissible to assume that the intake occurred at the time when the expectation value of all intakes consistent with a given bioassay result would have occurred. This assumption is correct on the average and, if always made, will lead to an unbiased estimate of collective dose in a population. It is also permissible to make the "midpoint assumption" (See the Time of Intake paragraph in Section 7.3.2.2).

7.4.1.2 A Method for Deducing Time of Intake from Bioassay Data

Assume Q's are retained quantities (in some compartment that can be measured) and a single, acute intake has occurred. Bioassay measurements show Q_1 at one time and Q_2 at a time Δt later (?). Let t_1 denote time between intake and Q_1, and Q_0 denote amount of initial retained quantity. It is desired to find the time of intake t_1, that is, how long before Q_1 intake occurred, and the value of Q_0 at the time of intake.

In general, retention functions giving unique relations for a given Δt have unique solutions. However, a single exponential, $e^{-\lambda t}$, has no unique solution.

Numerical solutions are possible for any retention or excretion functions with other than a single, linear first-order clearance. Below are analytical solutions for a two-exponential radionuclide retention function.

The retention function, $R(t)$, is

$$R(t) = k_1 e^{-\lambda_1 t} + k_2 e^{-\lambda_2 t} \tag{9}$$

The time of intake, t_0, is

$$t_0 = \frac{1}{\lambda_2 - \lambda_1} \ln\left[\frac{k_2\left(1 - (Q_1/Q_2)e^{-\lambda_2 \Delta t}\right)}{k_1\left((Q_1/Q_2)e^{-\lambda_1 \Delta t} - 1\right)}\right], \tag{10}$$

and the retained quantity at t_0 is

$$Q_0 = \frac{Q_1}{k_1 e^{-\lambda_1 t_1} + k_2 e^{-\lambda_2 t_1}} . \tag{11}$$

For more complex models, analytical solutions are probably not available, but such problems can be worked out by taking ratios of bioassay results or retained quantities and comparing them to the ratios of the intake retention functions for the appropriate compartment evaluated at various values of t_1 and $t_1 + \Delta t$ values until the correct answer is found. In some cases, there may be two answers; in such cases, other information, such as three or more bioassay measurements, may be needed to uniquely deduce the time of intake. The approaches outlined above are meaningful for a single, acute intake. They do not deal with uncertainties in measured values of Q, which may significantly affect results in some cases. In particular, these methods are not useful when Q_1 and Q_2 have similar magnitude and large uncertainties.

7.4.1.2 The Time of Intake to Be Assumed for Calculating *DIL*s and *DRL*s

*DIL*s and derived reference levels (*DRL*s) should be calculated at a time of intake that corresponds to a dose that is the expectation value of dose based on uniform intake probability between bioassay measurements.

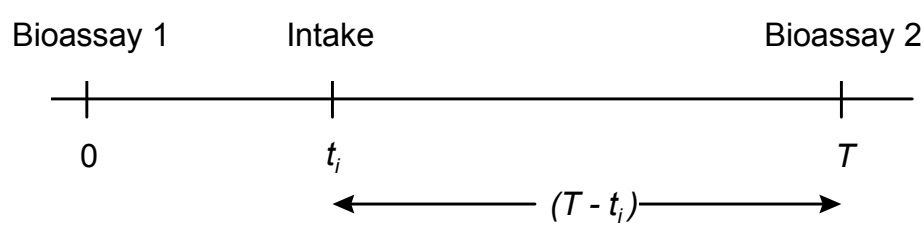


Figure 4. Time Line for Intake Between Two Bioassay Measurements

Let T be the time interval between bioassay samples as shown in Figure 4. Assume that intakes are equally likely at any time during the interval between bioassay samples, that is, the probability of intake per unit time is $p_i(t) = 1/T$. Then, the expectation value of time of intake, $\langle t_i \rangle$, is

$$<t_i> \;=\; \frac{\int_0^T p_i(t)t\cdot dt}{\int_0^T p_i(t)\cdot dt} \;=\; \frac{\int_0^T \frac{t\,dt}{T}}{\int_0^T \frac{dt}{T}} \;=\; \frac{T^2/2T}{1} \;=\; \frac{T}{2}\;. \tag{12}$$

This value, *T/2*, is used by the ICRP (1982, 1988) and has been used by DOE contractors (Johnson 1991).

The expectation value of dose, however, is not the dose that would occur from an intake at time *T*/2. Given a certain bioassay result (activity retained or activity excreted), *X* (known to arbitrary precision) and a fractional retention or excretion estimate, *R*, the intake *I* (to which dose is linearly proportional) is

$$I(t_i) \;=\; \frac{X}{R(T-t_i)}\;, \tag{13}$$

where T-t_i is the interval between intake and bioassay measurement. The expectation value of the intake, $<I>$, is

$$<I> \;=\; \frac{\int_0^T \frac{X\cdot p_i(t)}{R(t)}\cdot dt}{\int_0^T p_i(t)\cdot dt}\;. \tag{14}$$

For the simple case when *X* is the activity retained, and the retention function is a simple exponential,

$$R(t) \;=\; 1\cdot e^{-\lambda_e\cdot t}\;, \tag{15}$$

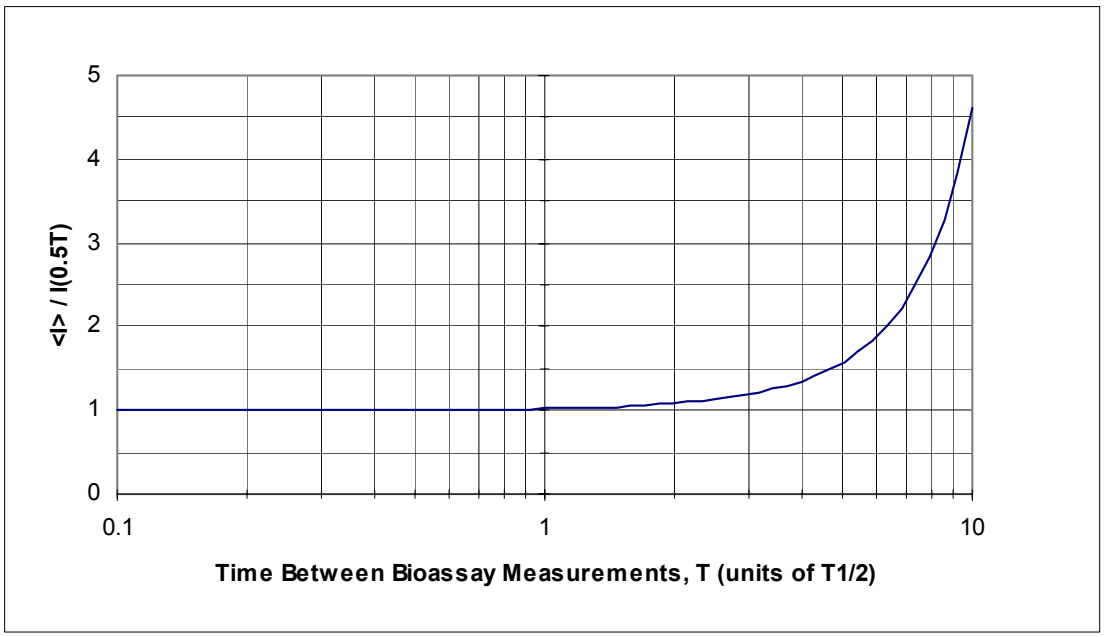
Eq. (13) then becomes

$$<I> \;=\; \frac{X}{T}\cdot\int_0^T e^{+\lambda_e(T-t)}dt \;=\; \frac{X\left(e^{+\lambda_e T}-1\right)}{\lambda_e T}\;. \tag{16}$$

The intake calculated from the assumption that it occurred at $<t_i> = T/2$ is

$$I(<t_i> = T/2) \;=\; \frac{X}{R(T/2)} \;=\; \frac{X}{e^{-\lambda_e T/2}} \;=\; Xe^{+\lambda_e T/2}\;. \tag{17}$$

The ratio of Equations (15) and (16) is



(18)

Figure 5. Expectation Value of Intake Divided by Intake at $T/2$ for a Single Exponential Retention Function

The ratio in Eq. (17) is plotted in Figure 5. Clearly, the expectation value of the intake, $<I>$, given bioassay result X and a uniform probability of intake throughout the interval between bioassay measurements, is greater than the value of the intake calculated at the expectation value of the intake time, $<t_i>$.

There exists a time, t_x, ($0 \le t_x \le T$), such that an intake of $<I>$ at t_x would yield bioassay result X at time T. This time is

$$t_x = T - \frac{1}{\lambda_e} \cdot \left(\ln\left[e^{\lambda_e T} - 1 \right] - \ln[\lambda_e T] \right) . \tag{19}$$

A plot of $t_x \div (T/2)$ as given in Eq 19 is shown in Figure 7. Clearly, t_x occurs earlier in time than $T/2$, and an intake calculated at $T/2$ underestimates the expectation value of the intake over the range of 0 to T.

What value, then, $<t_i>$ or t_x, should be used for computing *DIL*s and *DRL*s? The time of the expectation value of intake is what we're concerned with, not the expectation value of intake time. Therefore, t_x is the correct value. Use of T/2 rather than t_x for calculating the *DIL* will result in a *DIL* that is too high, depending on the nature of the intake retention function and the length of time between bioassay measurements.

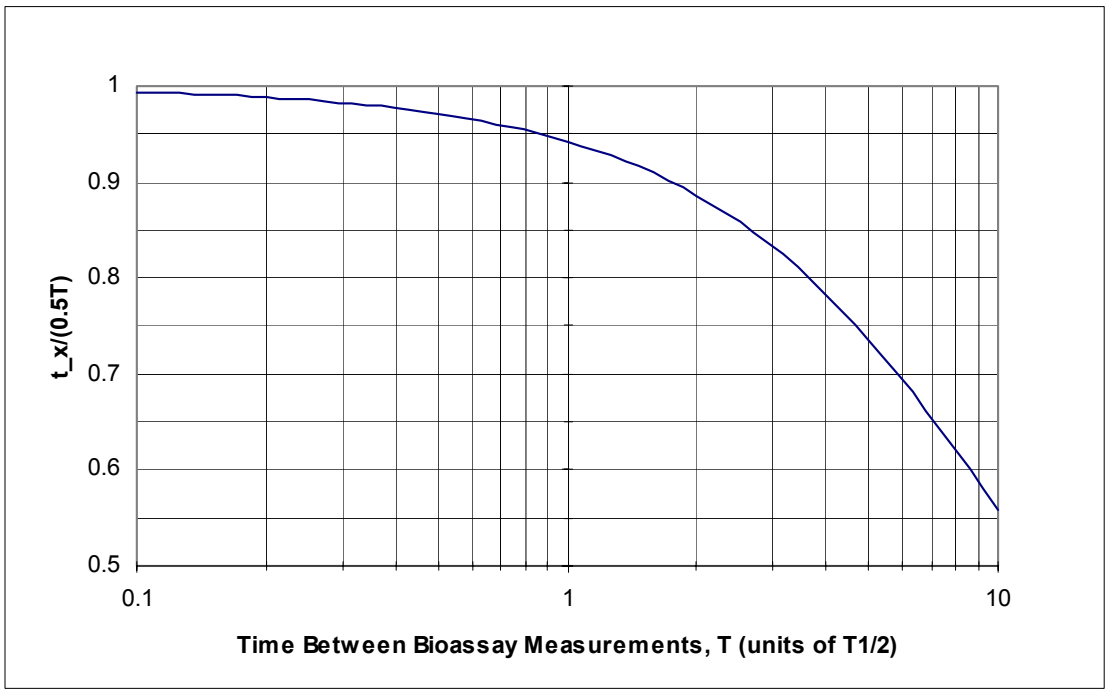


Figure 6. Time at Which Expectation Value of Intake Occurs as a Fraction of Interval Midpoint for a Single Exponential Retention Function

7.4.2 Intake and Dose Assessment for Mixtures of Radionuclides

Mixtures of radionuclides can pose difficulties in assessment due to bioassay methods for different nuclides having different sensitivities. When the isotopic composition of a mixture can be reasonably known or assumed, an effective approach to bioassay and simplified intake and dose assessment can be to select an indicator nuclide for the mixture and then base intake and dose assessments on the isotopic activity ratios of each nuclide in the mixture relative to the indicator nuclide.

Example 7.1. Radionuclide Mixture: Sludge from Tanks Containing High Level Waste

High-level waste tank sludge mixtures may consist of predominantly mixed fission product radioactivity (mainly ^{137}Cs and ^{90}Sr) with trace amounts of transuranics. From the bioassay perspective, the only readily detectable nuclides may be the fission products. However, the contribution to total dose may be far more significant from the minute quantities of transuranics. A logical nuclide for bioassay would be ^{137}Cs because of its ease of measurement by whole body counting and its relatively well-established biokinetic behavior. Detection of ^{137}Cs would result in estimating the intake of that nuclide using the standard biokinetic model for ^{137}Cs in the total body. Once the ^{137}Cs intake was obtained, that result would be multiplied by the isotope ratio of each nuclide in the mixture relative to ^{137}Cs to give the intake of that nuclide. Doses can then be evaluated by calculating the contribution from each nuclide intake.

When using indicator radionuclides and isotope ratios for mixtures, it is important to remember that the activity ratio at the time of bioassay is not necessarily the same as the activity ratio at the time of intake. If activity ratios in bioassay measurements at times following intake are being compared to those in a smear sample or other source term sample, it is necessary to consider the differing biokinetic behavior of the nuclides that are involved in the intake.

7.4.3 Special Considerations

Consideration must be given to dose assessment during treatment such as chelation or other enhanced decorporation treatment. Chelation has been treated by a number of authors (Bhattacharyya et al. 1992; Carbaugh et al. 1989; Goans 1996a; Goans 1996b; La Bone 1994a; La Bone 1994b). Another special consideration is the evaluation of intakes that include natural materials such as thorium, uranium, and radium. Thus, there are two distinct decisions to be made: whether a result differs from an analytical blank, and if so, whether the amount detected is greater than what would be expected in a population that is not occupationally exposed (Long et al. 1994; MacLellan et al. 1996). For example, the internal dosimetry program at Hanford distinguishes between the environmental decision level L_C and the analytical decision level *DL* (Carbaugh et al. 1995).

7.5 CALCULATION OF INTERNAL DOSE FROM WORKPLACE DATA

The derived air concentration (*DAC*) is the quotient of the annual limit on intake (*ALI*; not tabulated in 10 CFR 835) by the volume of air that Reference Man breathes in 1 working year (40 hr wk^{-1} × 50 wk yr^{-1} × 1.2 m^3 hr^{-1} = 2400 m^3 yr^{-1} or 2.4 × 10^9 mL yr^{-1}). The *DAC*s are expressed in µCi mL^{-1} or Bq·m^{-3}, or, for radon and thoron progeny, in working levels (WL). For a stochastic *ALI* (denoted *SALI*), breathing air at one DAC_s (stochastic *DAC*) for 2000 hours results in a committed effective dose equivalent ($H_{E,50}$) of 5 rems to Reference Man. For a nonstochastic or deterministic *ALI* (*NALI*), breathing air at one DAC_n (nonstochastic *DAC*) for 2000 hours results in a 50-year tissue committed dose equivalent to tissue *T* ($H_{T,50}$) of 50 rems. Note that the *DAC*s listed in Appendix A to 10 CFR 835 may be either stochastic (denoted as "St" in the right-hand column) or nonstochastic (denoted by "BS," "K," "L," "SW," and "T" [bone surfaces, kidneys, liver, stomach wall, and thyroid, respectively] in the right-hand column), so that reference to other documents may be needed for dose assessment, such as Federal Guidance Report 11 (Eckerman et al. 1988) or the ICRP Publication 30 series.

7.5.1 Intake

For record-keeping purposes for radioactive materials other than the short-lived progeny of radon and thoron, it is necessary to record intake, *I* (in µCi),

$$I = \frac{\overline{C}_a \cdot t \cdot \textit{Breathing Rate}\ \left(1.2 \times 10^6\ \text{cm}^3/\text{hour}\right)}{APF}, \tag{20}$$

based on a worker's exposure time, *t* (in hours); the average air activity concentration, $\overline{C_a}$ (in µCi/cm^3); the breathing rate of Reference Man, 1.2×10^6 cm^3/hour; and the assigned respiratory protection factor, *APF* (dimensionless; see below for details). In Eq. 21, it is acceptable to substitute the individual worker's actual breathing rate if it has been measured and documented doing identical or similar work.

One acceptable method for determining $H_{E,50}$ is

$$H_{E,50} = \frac{I}{SALI} \cdot 5\ \text{rems.} \tag{21}$$

Stochastic *ALI*s, *SALI*, for inhalation, can be computed from *DAC*s in 10 CFR 835 Appendix A provided the notation "St" appears in the right-hand column of Appendix A of 10 CFR 835 for the *DAC* in question.

If not, *SALI* values can be found in Federal Guidance Report 11 (Eckerman et al. 1988) or the ICRP Publication 30 series. These *SALI* values are valid for 1 µm activity median aerodynamic diameter *(AMAD)* aerosols. ICRP 30 gives a methodology for adjusting the SALIs for other particle sizes (ICRP 1979a). If the more recent respiratory tract model and dosimetric methods are used (ICRP 1994a), consideration should be given to adjustments for activity median thermodynamic diameter *(AMTD)* for aerosol size distributions below 0.1 µm, in the size region where the diffusive behavior of particles predominates.

7.5.2 Exposure in *DAC*-h

Often, before an intake is computed, an *exposure* in terms of *DAC*-h is evaluated. The exposure, *E*, is

$$E \text{ (in "stochastic" DAC-hours)} = \frac{C_a \cdot t}{DAC_s \cdot APF}. \tag{22}$$

For an aerosol whose activity median aerodynamic diameter *(AMAD)* is 1 µm, the committed effective dose equivalent $H_{E,50}$ is

$$\begin{aligned} H_{E,50} &= 5 \text{ rems} \cdot \frac{\bar{C}_a}{DAC_s \cdot APF} \cdot \frac{t}{2000 \text{ hours}} \\ &= 5 \text{ rems} \cdot \frac{E \text{ (DAC-hours)}}{2000 \text{ hours}} \\ &= E \text{ (DAC-hours)} \cdot 0.0025 \text{ (rem/DAC-hour)}. \end{aligned} \tag{23}$$

The committed dose equivalent to limiting tissue *T* (as listed in the right-hand column of Appendix A of 10 CFR 835) is

$$H_{T,50} = 50 \text{ rems} \cdot \frac{C_a}{DAC_s \cdot APF} \cdot \frac{t}{2000 \text{ hr}} \tag{24}$$

If the only *DAC* available is a nonstochastic *DAC*, then $H_{E,50}$ cannot be assessed using that *DAC* and air monitoring data. All that can be stated with certainty regarding the committed effective dose equivalent from an intake of 1 *NALI* (2000 DAC_n-h) is

$$50 \text{ rems} \cdot w_T \le H_{E,50} \le 5 \text{ rems}. \tag{25}$$

For thyroid and bone surfaces, $w_T = 0.03$, an intake of 1 *NALI* (2000 DAC_n-h) leads to $H_{E,50}$ of

$$1.5 \text{ rems} \le H_{E,50} \le 5 \text{ rems}, \tag{26}$$

and for "other" tissues (stomach wall, liver, kidneys), $w_T = 0.06$, an intake of 1 *NALI* (2000 DAC_n-hours) leads to $H_{E,50}$ of

$$3 \text{ rems} \leq H_{E,50} \leq 5 \text{ rems.} \tag{27}$$

In cases where only a nonstochastic *DAC* is listed in 10 CFR 835, it is acceptable to use the corresponding stochastic *DAC* for the radionuclide, particle size, and chemical form, as listed in Federal Guidance Report 11 (Eckerman et al. 1988) or in the ICRP Publication 30 series.

7.5.3 Assigned Respiratory Protection Factors for Use in Dose Evaluations

The American National Standards Institute has addressed the use of assigned respiratory protection factors (ANSI 1992) for planning purposes. Older information can be found in the U.S. Nuclear Regulatory Commission's regulatory guide and a NUREG report on respiratory protection (NRC 1973, 1976). In addition, "protection factors for respirators" are specified in Appendix A to §§20.1001-20.2401 of 10 CFR 20 (NRC 1993b). If a DOE site chooses to use assigned protection factors that differ from those in the ANSI Standard or 10 CFR 20 Appendix A, then the technical basis for this choice must be documented. Assigned protection factors for respirators used for radon and thoron and their short-lived progeny are treated in Section 7.5.7.

7.5.4 Assessment of Intake, Exposure, and Dose from Radon, Thoron, and Their Progeny

The basis for protection from airborne short-lived decay products of radon and thoron is explained in ICRP Publication 32 (ICRP 1981b). *Exposure* to airborne short-lived decay products of radon and thoron is given the special name *potential alpha energy exposure (PAEE)* for two reasons:

- The relevant ionizing energy is delivered to the bronchial epithelium by alpha particles from ^{218}Po and ^{214}Po in the case of ^{222}Rn and from ^{212}Bi and ^{212}Po in the case of ^{220}Rn (thoron).
- The decay-product aerosol often contains an unknown mixture of the various radon and/or thoron progeny.

For radon and thoron progeny, *PAEE* can be expressed as the product of average *potential alpha energy concentration (PAEC)* and worker stay time and divided by the assigned respiratory protection factor, if any. The traditional unit of *PAEC* is the working level (WL), and traditionally, stay times have been measured in occupational "Months" of 170 hours. Thus, the traditional unit of *PAEE* is the working level month, or WLM.

For routine monitoring of workers who are chronically exposed, weekly average air concentrations can be used for workers whose stay times are less than 40 hours in a given week.

PAEC can be computed from concentration measurements of the short-lived radon progeny in air (NCRP 1990):

$$\begin{aligned} PAEC_{Rn}\ (\text{WL}) &= \frac{(0.105\, C_{Po-218} + 0.516\, C_{Pb-214} + 0.379\, C_{Bi-214})}{100\,\text{pCi/L of radon per WL at equilibrium}} \\ &= (0.00105\, C_{Po-218} + 0.00516\, C_{Pb-214} + 0.00379\, C_{Bi-214}), \end{aligned} \tag{28}$$

where

$C_{Po\text{-}218}$ denotes the concentration of ^{218}Po in pCi/L;
$C_{Pb\text{-}214}$ denotes the concentration of ^{214}Pb in pCi/L; and
$C_{Bi\text{-}214}$ denotes the concentration of ^{214}Bi in pCi/L.

PAEC can be computed from concentration measurements of the short-lived thoron progeny in air (UNSCEAR 1993):

$$PAEC_{Tn}\ (\mathrm{WL}) = \frac{(0.913\ C_{Pb\text{-}212} + 0.087\ C_{Bi\text{-}212})}{7.43\,\mathrm{pCi/L\ of\ thoron\ per\ WL\ at\ equliibrium}} = (0.12288\ C_{Pb\text{-}212} + 0.011709\ C_{Bi\text{-}212}), \tag{29}$$

where

$C_{Pb\text{-}212}$ denotes the concentration of ^{212}Po in pCi/L; and
$C_{Bi\text{-}212}$ denotes the concentration of ^{212}Bi in pCi/L.

Another acceptable method for workplace monitoring of exposure to radon progeny is to measure the ^{222}Rn itself, and convert it to *PAEC* using known equilibrium factors. DOE's 10 CFR 835 permits calculating *equilibrium equivalent concentration, EEC,* from radon *concentration* measurements, *C*, based on knowledge or assumption of an *equilibrium factor, F*:

$$EEC\ (\mathrm{pCi/L}) = C\ (\mathrm{pCi/L}) \cdot F \tag{30}$$

If *F* has not been measured, it is acceptable under some circumstances to assume a default indoor value of $F_{\mathrm{Rn}} = 0.4$ (UNSCEAR 1988, 1993; ICRP 1993a). If *C* is in units of μCi/mL, then *EEC* will also be in units of μCi/mL (note: 1 pCi/L = 10^{-9} μCi/mL).

For ^{222}Rn, F_{Rn} is defined as

$$F_{\mathrm{Rn}} = \frac{0.105\ C_{\mathrm{Po\text{-}218}} + 0.516\ C_{\mathrm{Pb\text{-}214}} + 0.379\ C_{\mathrm{Bi\text{-}214}}}{C_{\mathrm{Rn\text{-}222}}} \tag{31}$$

where

$C_{\mathrm{Po\text{-}218}}$ = the concentration of ^{218}Po;
$C_{\mathrm{Pb\text{-}214}}$ = the concentration of ^{214}Pb;
$C_{\mathrm{Bi\text{-}214}}$ = the concentration of ^{214}Bi; and
$C_{\mathrm{Rn\text{-}222}}$ = the concentration of ^{222}Rn.

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

$$F_{\mathrm{Tn}} = \frac{0.913\ C_{\mathrm{Pb\text{-}212}} + 0.087\ C_{\mathrm{Bi\text{-}212}}}{C_{\mathrm{Rn\text{-}220}}} \tag{32}$$

where

$C_{\text{Pb-212}}$ = the concentration of ^{212}Pb;
$C_{\text{Bi-212}}$ = the concentration of ^{212}Bi; and
$C_{\text{Rn-220}}$ = 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

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

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

$$\begin{aligned} \bar{C}\ (\text{pCi/L}) &= \frac{N_V\ (\text{pCi}\cdot\text{d/L})}{t_E\ (\text{d})}\ \text{and} \\ EE\bar{C}\ (\text{pCi/L}) &= F\cdot\bar{C} = \frac{F\cdot N_V\ (\text{pCi}\cdot\text{d/L})}{t_E\ (\text{d})}. \end{aligned} \tag{34}$$

number of transitions per unit volume, N_V, and the equilibrium factor using

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

$$\begin{aligned} E\ (\text{WLM}) &= F\cdot N_V\ (\text{pCi}\cdot\text{d/L})\cdot\left(\frac{1\ \text{WL}}{100\ \text{pCi/L}}\right)\left(\frac{24\ \text{h}}{\text{d}}\right)\left(\frac{1\ \text{Mo}}{170}\right. \\ &= 1.4118\times10^{-3}\,F\cdot N_V\ (\text{pCi}\cdot\text{d/L});\ \text{and} \\ E\ (\text{WLM}) &= 5.6471\times10^{-4}\,N_V\ (\text{pCi}\cdot\text{d/L})\ \text{if}\ F = 0.4. \end{aligned} \tag{35}$$

Committed effective dose equivalent is assessed directly from *PAEE* using

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

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

From 10 CFR 835 one can infer a dose conversion factor of 1.25 rems per WLM, using the following equation:

$$
\left(\frac{5\ \text{rems}}{2000\ \text{DAC-hours}}\right)\left(\frac{1\ \text{DAC}}{1/3\ \text{WL}}\right)\left(\frac{2000\ \text{hours}}{12\ \text{Months}}\right) = 1.25\ \text{(rems/WLM)},
\tag{37}
$$

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 DACs are based on a 1 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 is larger than that recommended in the international guidance, meaning that for the same exposure, the DOE rule would impute a larger dose. Further, 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 $H_{E,50}$ of

$$
\begin{aligned}
H_{E,50}\ \text{(rems)} &= (7.0588 \times 10^{-4})\ (30\ \text{pCi}\cdot\text{d/L}) \\
&= 0.021\ \text{rems if } F = 0.4.
\end{aligned}
$$

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

7.5.5 Calculating Committed Dose Equivalent 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 $H_{E,50}$, one must calculate $H_{50,lung}$ from that portion of the committed effective dose equivalent due to radon or thoron progeny using

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

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 equivalent from the sum of committed dose equivalent to tissues multiplied by the weighting factor for those tissues, it is necessary because air concentration measurements lead to $H_{E,50}$, not to $H_{50,lung}$.

The 1988 Federal Guidance Report 11 lists the "Annual Limit on Intake" for ^{222}Rn as 4 WLM and for ^{220}Rn as 12 WLM, and identifies these values as the "Primary guide" (Eckerman et al. 1988). However, 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

$$\begin{aligned} I\,(\text{J}) &= PAEE\,(\text{WLM}) \cdot (1.30 \times 10^5\ \text{MeV L}^{-1}\ \text{WLM}^{-1}) \cdot (1.6022 \times 10^{-13}\ \text{J MeV}^{-3}) \\ &\quad \cdot (170\ \text{h Month}^{-1}) \cdot (1.2\ \text{m}^3\ \text{h}^{-1}) \\ &= PAEE\,(\text{WLM}) \cdot (3.5408 \times 10^{-3}\ \text{J h m}^{-3}\ \text{WLM}^{-1}) \cdot (1.2\ \text{m}^3\ \text{h}^{-1}) \\ &= PAEE\,(\text{WLM}) \cdot (4.2490 \times 10^{-3}\ \text{J WLM}^{-1}). \end{aligned} \tag{39}$$

In Equation 40, 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 (µCi) should be converted to equilibrium equivalent intake, *EEI*, using

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

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

---

[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.

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	0	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	5/4 = 1.25	$H_{E,50}$	rem
^{222}Rn *PAEE*	WLM	2000/4 = 500	*E* (exposure)	*DAC*·h
^{220}Rn *PAEE*	WLM	$5/12 = 0.433\overline{3}$	$H_{E,50}$	rem
^{220}Rn *PAEE*	WLM	$2000/12 = 166.6\overline{6}$	*E* (exposure)	*DAC*·h
$H_{E,50}$ 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 *ALI*s 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 should 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 3,333 pCi/L for pure ^{222}Rn and 3,730 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 mrem per hour (i.e., 25 μSv·h^{-1} or 25,000 nSv·h^{-1}), so a "5-rem 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 (effective dose equivalent and effective dose).

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/m3)	*DAC* (pCi/L)	*DAC* (μCi/cm3)
		Gas	EEC	Gas	Gas	Gas
Radon	Outdoors	0.17	9	147059	3975	0
	Indoors	0.17	9	147059	3975	0
Thoron	Outdoors	0.11	10	227273	6143	0.00001
	Indoors	0.11	32	227273	6143	0.00001

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

$$APF \leq 100 \cdot F_{Rn} \text{ for } ^{222}\text{Rn, and}$$
$$APF \leq 500 \cdot F_{Tn} \text{ for } ^{220}\text{Rn,} \quad (41)$$

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	$1 \leq APF \leq 100 \cdot F_{Rn}$		$1 \leq APF \leq 500 \cdot F_{Tn}$	
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 should 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 should 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 $H_{E,50}$ < 100 mrem), it is permissible to assign $H_{E,50}$ values using Eq. (20), 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 VII.

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	-

Assessing committed effective dose equivalent *($H_{E,50}$)* from bioassay measurements is generally *mores accurate* than assessing $H_{E,50}$ 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 $H_{E,50}$. 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.

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 $H_{E,50}$ that can be distinguished from background. Technology shortfall, as defined in the DOE *Internal Dosimetry Program Guide* (DOE 1999b), occurs when the sensitivity of a dose assessment method is not adequate to meet the dose assessment requirements of 10 CFR 835.

The best accuracy and precision for $H_{E,50}$ 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 μCi of ^{3}H per liter of urine (10,000 pCi/L) can detect 0.04 mrem immediately after a tritium intake, and 22 mrem 90 days after a tritium intake. With an average tritium sampling frequency of every 14 days, one can detect a committed effective dose equivalent 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 Equivalent Detection Limits for Tritium Bioassay When 0.01 μCi/L of ^{3}H Is Observed, as a Function of Time since Intake

	$H_{E,50}$ 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 $H_{E,50}$ 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 $H_{E,50}$ 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.

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8.0 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 dose equivalent due to external irradiation to committed effective dose equivalent due to irradiation by internal sources. Optimization principles should be applied to maintain internal and external doses ALARA (ICRP 1978b, 1989a; DOE 1990c). This necessitates a close working relationship and cooperation between staff, management, medical, and radiation protection personnel. Each site should 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 should be prevented from having additional intakes until their cumulative TEDE, 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 should 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 women. This is necessary because of uncertainties in:

- distribution and retention of radioactive materials in the embryo/fetus
- dosimetry to embryo/fetus
- associated risk (Sikov et al. 1996).

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

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 equivalent 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 1999e). 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 TEDE, 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 EPA, the ICRP, and 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 and DOE Technical Positions (DOE 1999e; Office of Worker Protection Programs and Hazards Management 1995b). 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).

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 $H_{E,50}$ 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 $H_{E,50}$ 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 (Office of Worker Protection Programs and Hazards Management 1995b).

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 $H_{E,50}$ 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 ACCIDENTAL DOSE CONTROL

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 should 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 should 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 should 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, should 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 should 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) should 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 should 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 equivalent 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.

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9.0 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 1999e), and in the *Internal Dosimetry Program Guide* (DOE 1999b). Prior dose assessments not compatible with committed dose equivalents should be converted to provide committed organ/tissue and effective dose equivalents. However, constraints discussed in Section 8.2.3 may limit some reassessments.

Requirements to 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 1999c) provides guidance for the systematic generation and retention of records relating to occupational radiation exposure.

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 judgement 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 judgements 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 should 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 $H_{E,50}$ (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 developed an internal contamination incident response plan, contained in the *Hanford Internal Dosimetry Project Manual* (Carbaugh et al. 1994), which specifically identifies the capability of response as a function of time following intake and measurements made. For example, the plan identifies the capability for various combinations of measurements following an aged weapons-grade plutonium mixture inhalation to be as shown in Table XI. This table was derived for standard Hanford dosimetry assumptions. Similar tables have been developed for other radionuclides and scenarios.

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 rem CEDE derived from initial bioassay data for a plutonium intake to ultimately be lowered to 1 rem CEDE 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 should 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 $H_{E,50}$ 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 intake is class W < 5 rems, or class Y < 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 $H_{E,50}$ class W < 500 mrem, class Y < 5 rems	From bioassay data, still won't know inhalation class of material
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 $H_{E,50}$ < 500 mrem If nothing in IPA, then $H_{E,50}$ < 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 equivalent 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. There is no requirement, and indeed no actual mechanism in place, for reporting revised intake and dose assessments to the DOE Radiation Records Repository after the annual calendar year reporting.

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 CEDE 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-mrem tritium dose with a 3.1-rem 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 TEDE

10 CFR 835.702(c)(5)(iii) requires maintaining records of cumulative total effective dose equivalent 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 TEDE contributions from intakes prior to this time. This lifetime TEDE is consistent with the guidance concept of lifetime effective dose equivalent contained in NCRP Report 91 (1987). It provides a more complete estimate of lifetime cumulative dose for comparing with the RadCon Standard lifetime control level. While determination of TEDE 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 should 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 1999c).

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

Since radon quantities and units differ from the traditional activity concentration (expressed in μCi/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)
- $H_{E,50}$ 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.

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10.0 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 actinide intake but before any confirming bioassay measurements. Treatment in response to identified intakes includes diuretics following tritium exposure, and use of adsorption agents to prevent gastro-intestinal 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 3.5.

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 should address may include 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. Guidance has been offered in the volume edited by Gerber and Thomas (Bhattacharyya et al. 1992). This guidance, summarized in Table XII, is expressed in terms of *ALI*s. However, these *ALI*s are based on the 20-mSv (2 rems) per *ALI* concept of ICRP-60, rather than the 5-rem limit of 10 CFR 835).

While Table XII can provide philosophical guidance on when therapy is needed, it does not fulfill the 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 have been calculated at Hanford and are summarized in Table XIII and Table XIV. 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

Table XII. Early Bioassay Measurement Results Corresponding to the Therapeutic Intervention Action Levels Used at the Hanford Site (Carbaugh et al. 1995) (Part 1)

Isotope and Dose ($H_{E,50}$)	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 or in vivo detection	>200,000 dpm in spot urine, or >*MDA* in vivo	Consider therapy	Alginate, Ca gluconate, Sr lactate, others
20 rems	Same	>2,000,000 dpm in spot urine, or >50 μCi in vivo	Treatment strongly recommended	Same

arguable. Knowledge of facility operations, material forms, and past experience will likely play a key role in development of such criteria.

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

Table XIII. Early Bioassay Measurement Results Corresponding to the Therapeutic Intervention Action Levels Used at the Hanford Site (Carbaugh et al. 1995) (Part 2)

Isotope and Dose ($H_{E,50}$)	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 Early urine sample	>*MDA* for Pu or ^{241}Am >4 dpm when extrapolated to first day excretion	Consider therapy	DTPA

[(a)] If soluble component is present, then urine sampling is appropriate.
Use same action levels as above for soluble uranium.

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. 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).

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 should 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 of 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 should have access to a supply of DTPA and a physician registered as a co-investigator. DTPA is approved by the U.S. Food and Drug Administration as an Investigational New Drug (IND) and is available to physicians who are registered as co-investigators (Goans 1996a, 1996b). As of September 1996, physicians can register as IND co-investigators by contacting the REAC/TS DTPA program, Patrick Lowry, M.D., Head of Medical Section, at (865) 576-4049.

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 judgement 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.

11.0 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 1996a & 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, independent review is indicated. 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.

11.3.1 Configuration Management

Dosimetry codes should 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 should undergo a two-step verification and validation (V&V) process as acceptance testing before their routine use for dosimetry (ANSI 1987). 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 should 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 should 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 should 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 should 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. 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) should to 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.

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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 XIV. 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	0.1		0	1	Reif and Andrews (1992)	at the source	0.1
USA	0.45		0.1	1	Reif and Andrews (1992)	upwind	0.4
USA	0.2		0	0.91	Reif and Andrews (1992)	downwind	0.2
USA	0.1		0	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	0.1	0.45	Medora (1996)	stabil. class A	
USA	0.22				Medora (1996)	stabil. class B	
USA	0.39	0.2	0.22	0.63	Medora (1996)	stabil. class D	
USA	0.22	0	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	1	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**						
Max of Maxima	**1**						
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 XV are 16 results of studies of F_{Rn} 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

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		0.10	0.80	Swedjemark (1983)		
Sweden	0.51				Jonassen and McLaughlin (1989)	smoker	
Sweden	0.46				Jonassen and McLaughlin (1989)	nonsmokers	
USA	0.63				George and Breslin (1980)	living areas	
USA	0.33				Israeli (1985)	living areas	
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 $F = 0.28 \pm 0.09$, 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 XVI**) are lower than many of the home equilibrium factors given in **Table XV**.

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 XVII**. 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)		
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 XVIII**. 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 XIX** (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*		*0.1*

CONCLUDING MATERIAL

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DOE-EH-52

Project Number:
SAFT-0057