TS

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DOE STANDARD

GUIDE OF GOOD PRACTICES FOR OCCUPATIONAL RADIOLOGICAL PROTECTION IN URANIUM FACILITIES

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

AREA SAFT

DOE-STD-1136-2004

Guide of Good Practices for Occupational Radiological Protection in Uranium Facilities

Foreword

This Technical Standard (TS) discusses, but does not establish any, requirements for DOE uranium facilities. Its purpose is to provide information that will assist DOE and DOE-contractor health and safety professionals in developing programs that will provide an appropriate level of protection to both affected workers and members of the public affected by DOE uranium-handling activities. This TS provides guides to good practice, update existing reference material, and discuss practical lessons learned relevant to the safe handling, processing, and storage of uranium. The technical rationale for the guidance provided herein is explained to allow affected individuals to adapt the recommendations to similar situations throughout the DOE complex. This TS provides information to assist uranium facilities in complying with Title 10 of the Code of Federal Regulations, Part 835 (10 CFR 835), Occupational Radiation Protection and various DOE Orders, and supplements DOE's 441.1 series of Guides and DOE-STD-1098-99, Radiological Control.

This TS does not include every requirement applicable to DOE uranium facilities. Individuals responsible for developing and implementing radiation protection programs at uranium facilities should be knowledgeable of the requirements that apply to their facilities.

Copies of electronic files of this TS may be obtained from the DOE Office of Worker Protection Policy and Programs Home Page Internet site (http://www.eh.doe.gov/radiation/ts.html). Copies of this TS are also available from the DOE Technical Standards Program Internet site (http://www.eh.doe.gov/techstds/).

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1.0 INTRODUCTION

1.1 PURPOSE AND APPLICABILITY

This Technical Standard (TS) provides operational guidance, practical lessons learned and experience gained, guides to good practice, and reference information on the safe handling of uranium. The TS provides information to assist uranium facilities in complying with Title 10 of the Code of Federal Regulations (CFR), Part 835, Occupational Radiation Protection (10 CFR 835) (DOE 1998a). This TS supplements the DOE G 441.1 series of Guides, DOE Orders, and DOE-STD-1098-99, Radiological Control (RCS) (DOE 1999a) and has as its sole purpose the protection of workers and the public from the hazards that are inherent in uranium storage, processing, and handling.

This TS replaces EGG-2530, Health Physics Manual of Good Practices for Uranium Facilities (EGG 1988), providing more complete and current information and emphasizing situations that are typical of DOE's current operations, including weapons assembly and disassembly, safe storage, decontamination, and decommissioning (environmental restoration). This TS may be useful to health physicists and other safety professionals. The information presented herein represents the best technical information available from within the DOE complex. Except to the extent that the guidance presented here is an exact quote from applicable regulations or contract requirements, it is not binding or mandatory. However, judicious use of this TS, in concert with applicable regulatory documents, will help in building a comprehensive and technically-defensible radiological control program.

Regulatory guidance and references are current as of October 2004.

1.2 DEFINITIONS

A glossary is provided in Appendix A. In all cases, the definitions provided in this TS are consistent with those provided in 10 CFR 835, its Guides, and the RCS.

1.3 DISCUSSION

Chapters 2 through 10 provide technical information to assist in safely managing radiological hazards associated with uranium operations. The topics covered are those considered by representatives of many of DOE's uranium facilities to be most beneficial: Properties and Relative Hazards (Chapter 2), Radiation Protection (Chapter 3), Contamination Control (Chapter 4), Internal Dosimetry (Chapter 5), External Dose Control (Chapter 6), Nuclear Criticality Safety (Chapter 7), Waste Management (Chapter 8), Emergency Management (Chapter 9), and Decontamination and Decommissioning (Chapter 10).

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2.0 PROPERTIES AND RELATIVE HAZARDS

This chapter presents basic radiological and chemical properties of uranium and discusses the basis for current control limits. A variety of materials are inherent to uranium handling processes and hazards characteristic of these materials and processes. The data and discussions are intended to provide a basis for understanding the changes in hazards as a function of such parameters as enrichment, physical form, and chemical form.

2.1 NUCLEAR PROPERTIES OF URANIUM

Naturally occurring uranium consists of a mixture of ^{234}U, ^{235}U and ^{238}U isotopes, along with their decay products. Uranium is relatively abundant in nature. The primary isotopes of uranium are long-lived alpha-emitters with energies between 4.15 and 4.8 MeV. Their progeny include numerous other radionuclides, some of which are radiologically significant at uranium facilities, the degree of significance depending upon the history of the uranium materials and the processing.

Through proper processing, uranium can be used as a fuel in nuclear reactors to generate electricity on a commercially-viable scale. The ^{235}U isotope is readily fissioned by slow, "thermal" neutrons with the release of a large amount of energy. The percentage of ^{235}U present (referred to as "enrichment") determines the fuel reactivity and the criticality hazard of the material. By concentrating the amount of the ^{235}U isotope in the uranium, the quantity of fuel and the size of the reactor needed for production decreases. This concentration of natural uranium to enriched uranium is carried out by special processes such as gaseous diffusion, centrifuging, or laser separation. The uranium by-product of the enrichment process is reduced in ^{235}U content and is called "depleted" uranium. Uranium is commonly classified by its ^{235}U enrichment as natural uranium, enriched uranium, or depleted uranium.

Uranium-235 fissions after capturing a thermal (very low energy) neutron. Its fission thermal cross-section (probability of interaction) is 577 barns (Stehn et al. 1965). Its neutron capture cross section is 101 barns. After capturing a fast neutron, ^{238}U undergoes two successive beta decays to ^{239}Pu which will also undergo thermal fission (thermal cross-section = 741 barns). Pressurized heavy-water reactors function with natural uranium isotopic composition. Other types of reactors require some ^{235}U enrichment.

2.1.1 Isotopic Characterization

Natural uranium consists of three isotopes: ^{238}U, ^{235}U, and ^{234}U. All three radionuclides undergo radioactive decay by alpha particle emission. The ^{235}U isotope (and ^{234}U to a much lesser degree and at lower energy) emits gamma radiation as well. The natural abundances of these uranium isotopes, as well as the weight percentages of the isotopes in enriched (typical commercial nuclear power reactor enrichment) and depleted uranium, are listed in Table 2-1.

Table 2-1. Typical Isotopic Abundances

Isotope	Natural	Typical Commercial Feed Enrichment	Depleted	Specific Activity Ci/g	Neutron Capture Cross Section (barns)
^{238}U	99.28	97.01	99.80	3.3 E-7	2.7
^{235}U	0.72*	2.96	0.20	2.1 E-6	101
^{234}U	0.0055	0.03	0.0007	6.2 E-3	95

The amount of uranium present determines the grade of the ore. Most of the ores found in the U.S. contain from 0.1 to 1% uranium and are considered medium grade. Lower-grade ores are mined commercially if they are a byproduct of mining for another material, such as gold or phosphate.

Uranium that has been processed to raise the concentration of ^{235}U is referred to as enriched uranium. The extent of enrichment depends on the intended end use of the uranium. Commercial light water reactors are designed for use with the ^{235}U enriched to around 3%. Higher enrichment is required for; high-temperature gas-cooled reactors, naval nuclear propulsion reactors, most research reactors and weapons. The ^{235}U enrichment process also increases the concentration of ^{234}U. The higher activity of enriched uranium is due more to increased ^{234}U than to increased ^{235}U.

Depleted uranium is a by-product of the enrichment process and is depleted in both the ^{235}U and ^{234}U isotopes. Depleted uranium, with its reduced activity and very high density, has many uses; among them are radiation shielding, counterweights, projectiles, and target elements in DOE plutonium production reactors.

In addition to the uranium isotopes discussed above, the daughter products of uranium decay and byproducts of uranium processing can have significant radiological impacts in uranium-handling facilities. Table 2-2 presents the properties of these radionuclides.

Table 2-2. Properties of Radionuclides That May Be Found at Uranium Facilities[(a)]

Nuclide	Half-Life	Energies (MeV) and Abundances of Major Radiations (All low yield radiations are not included)		
		Alpha	Beta	Gamma
Primary Uranium Isotopes				
^{238}U	4.51 x 10^9 y	4.15 (21%) 4.20 (79%)		
^{235}U	7.1 x 10^8 y	4.21 (6%) 4.37 (17%) 4.40 (55%) 4.60 (5%)		0.144 (11%) 0.163 (5%) 0.186 (57%) 0.205 (5%)
^{234}U	2.47 x 10^5 y	4.72 (28%) 4.77 (72%)		0.053 (0.12%)
Decay Products				
^{234}Th	24.1 d		0.103 (21%) 0.193 (79%)	0.013 (9.8%) 0.063 (3.5%) 0.092 (3%) 0.093 (4%)
^{234m}Pa	1.17 m		2.29 (98%)	0.765 (0.30%) 1.001 (0.60%)
^{231}Th	25.5 h		0.206 (13%) 0.287 (12%) 0.288 (37%) 0.305 (35%)	0.026 (2%) 0.084 (10%)
Impurities (e.g. irradiation and reprocessing artifacts)				
^{99}Tc	2.12x10^5 y		0.292	
^{237}Np	2.14x10^6 y	4.78 (75%) 4.65 (12%)		
^{238}Pu	86.4 y	5.50 (72%) 5.46 (28%)		
^{239}Pu	2.44x10^4 y	5.16 (88%) 5.11 (11%)		
^{240}Pu	6.6x10^3 y	5.17 (76%) 5.12 (24%)		
^{241}Pu	13.2 y		0.021	
^{232}U	72 y	5.26 (31%) 5.32 (69%)		
^{236}U	2.34x10^7 y	4.47 (24%) 4.52 (76%)		

(a) From EGG-2530 (1988).

2.1.2 Decay Chains

The natural uranium isotopes decay by alpha emission. The decay products are also radioactive and form "decay chains" that ultimately lead to a stable isotope of lead. Tables 2-3 and 2-4 present the decay chains of ^{238}U and ^{235}U (^{234}U is a member of the ^{238}U decay chain), along with the half-lives and characteristic radiations of each nuclide.

Uranium-processing steps (milling or refining) separate the decay products and other impurities in the ore from the uranium. It takes months after processing before the first few decay products build up and come to equilibrium with the parents. In depleted uranium, the beta radiation from the decay of ^{234}Th and ^{234m}Pa amounts to nearly twice the alpha radiation from ^{238}U and ^{234}U. In commercially enriched uranium, the beta radiation from ^{231}Th, ^{234}Th, and ^{234m}PA nearly equals the alpha radiation from ^{238}U, ^{234}U, and ^{235}U. In natural ore, the later decay products (especially ^{230}Th and ^{226}Ra) are present and add significant gamma radiation to the emitted radiation. In processed uranium (natural, enriched, or depleted) all decay products below ^{234}U and ^{235}U are removed. Because of the long half-lives of ^{234}U and ^{231}Pa the radionuclides that follow these two nuclides are generally ignored.

The mining and milling stages are usually conducted by commercial enterprises. DOE facilities do not routinely process uranium ore concentrates and, as a result, the decay products formed during DOE processing operations of virgin feed are limited. However, radium and its progeny may be present in waste water streams of certain facilities, so it is prudent to consider those nuclides in effluent and environmental monitoring programs.

For workplace radiological controls, ^{234}Th, ^{234m}Pa, ^{231}Th and the uranium isotopes are those requiring primary consideration; however, if there are large quantities of aged highly enriched uranium, there may be a need to also consider ^{231}Pa in establishing radiological controls. In addition, elevated radon concentrations can occur in poorly ventilated uranium storage areas from the small amounts of ^{226}Ra that grow in and carry over as contaminants in the chemical separation processes.

Table 2-3. Uranium Series (4n + 2)

Nuclide	Historical Name	Half-life	Major Radiation Energies (MeV) and intensities† α		β		γ	
$^{238}_{92}U$	Uranium I	4.47×10^9y	4.15 4.20	(23%) (77%)	---		0.0496	(0.07%)
$^{234}_{90}Th$	Uranium X_1	24.1d	---		0.095 0.096 0.188	(0.2%) (19%) (72.5%)	0.063 0.0924 0.0928	(3.8%) (2.7%) (2.7%)
$^{234m}_{91}Pa$	Uranium X_2	1.17m	---		2.28	(98.69%)	0.766 1.001	(0.21%) (0.60%)
99.87% 0.13%								
$^{234}_{91}Pa$	Uranium Z	6.75h	---		0.53 1.13	(66%) (13%)	0.132 0.570 0.883 0.926 0.936	(19.7%) (10.7%) (11.8%) (10.9%) (12%)
$^{234}_{92}U$	Uranium II	2.44×10^5y	4.72 4.77	(27%) (72%)		---	0.053	(0.12%)
$^{230}_{90}Th$	Ionium	7.7×10^4y	4.62 4.68	(23%) (76%)		---	0.067 0.142	(0.37%) (0.07%)
$^{226}_{88}Ra$	Radium	1602y	4.60 4.78	(6%) (94%)		---	0.186	(3.3%)
$^{222}_{86}Rn$	Emanation Radon (Rn)	3.823d	5.49	(100%)		---	0.510	(0.07%)
$^{218}_{84}Po$	Radium A	3.05m	6.00	(~100%)	0.33	(~0.019%)	0.8337	(0.001%)
99.98% 0.02%								
$^{214}_{82}Pb$	Radium B	26.8m	---		0.67 0.73 1.03	(48%) (42%) (6%)	0.242 0.295 0.352	(7.5%) (19%) (37%)
$^{218}_{85}At$	Astatine	~2s	6.65 6.70 6.757	(6%) (90%) (4%)			0.736 0.053	(1.1%) (6.6%)
$^{214}_{83}Bi$	Radium C	19.9m	5.45 5.51	(0.012%) (0.008%)	1.42 1.505 1.54 3.27	(8.3%) (17.6%) (17.9%) (17.7%)	0.609 1.120 1.764 2.204	(46%) (15%) (16%) (5%)
99.98% 0.02%								
$^{214}_{84}Po$	Radium C'	164 μs	7.69	(100%)		---	0.799	(0.01%)
$^{210}_{81}Tl$	Radium C"	1.3m	---		1.3 1.9 2.3	(25%) (56%) (29%)	0.291 0.799 1.21 1.31	(79%) (99%) (17%) (21%)
$^{210}_{82}Pb$	Radium D	22y	3.72	(.000002%)	0.016 0.063	(80%) (20%)	0.047	(4%)
$^{210}_{83}Bi$	Radium E	5.01d	4.65 4.69	(.00007%) (.00005%)	1.161	(~100%)		---
~100% .00013%								
$^{214}_{84}Po$	Radium F	138.4d	5.305	(100%)		---	0.802	(0.0011%)
$^{210}_{81}Tl$	Radium E"	4.2m	---		1.571	(100%)		---
$^{206}_{82}Pb$	Radium G	Stable	---			---		---

Tables 2-3 and 2-4 are from Handbook of Health Physics and Radiological Health, 3rd edition, 1998 (Shleien, Slaback, and Birky)

Table 2–4. Actinium Series (4n + 3)*

Nuclide	Historical Name	Half-life	Major Radiation Energies (MeV) and intensities†					
			α		β		γ	
$^{235}_{92}U$	Actinouranium	7.04×10^{8}y	4.2-4. 4.37 4.39 4.5-4. 6	(10%) (18%) (56%) (11%)		---	0.143 0.163 0.1857 0.205	(11%) (4.7%) (54%) (4.7%)
$^{231}_{90}Th$	Uranium Y	25.5h		---	0.205 0.287 0.304	(15%) (49%) (35%)	0.026 0.084	(14.8%) (6.5%)
$^{231}_{91}Pa$	Protoactinium	3.27×10^{4}y	4.95 5.01 5.03 5.06	(23%) (26%) (20%) (11%)		---	0.027 0.284 0.30 0.303 0.33	(9.3%) (1.6%) (2.3%) (4.6%) (1.3%)
$^{227}_{89}Ac$	Actinium	21.77y	4.94 4.95	(0.53%) (0.66%)	0.019 0.034 0.044	(10%) (35%) (54%)	0.070 0.100 0.160	(0.02%) (0.032%) (0.02%)
98.87% 1.4% $^{227}_{90}Th$	Radioactinium	18.7d	5.76 5.98 6.04	(20%) (23%) (24%)		---	0.050 0.236 0.33	(8.5%) (11.2%) (2.7%)
$^{223}_{87}Fr$	Actinium K	22m	5.44	(~0.005%)	1.15	(~100%)	0.050 0.080 0.235	(34%) (9.2%) (3.4%)
$^{223}_{88}Ra$	Actinium X	11.43d	5.61 5.71 5.75	(24%) (52%) (9%)		---	0.144 0.154 0.269 0.324	(3.3%) (5.6%) (13.6%) (3.9%)
$^{219}_{86}Rn$	Emanation Actinon (An)	4.0s	6.42 6.55 6.82	(7%) (12%) (80%)		---	0.271 0.401	(9.9%) (6.6%)
$^{215}_{84}Po$	Actinium A	1.78ms	7.38	(~100%)	0.74	(~0.0023%)	0.4388	(0.04%)
~100% 0.0023% $^{211}_{82}Pb$	Actinium B	36.1m		---	0.26 0.57 1.39	(4.8%) (1.4%) (92.9%)	0.405 0.427 0.832	(3.8%) (1.38%) (2.8%)
$^{211}_{85}At$	Astatine	~0.1ms	8.03	(~100%)		---	0.404	(0.047%)
$^{211}_{83}Bi$	Actinium C	2.14m	6.28 6.62	(16%) (84%)	0.58	(0.27%)	0.351	(12.7%)
0.28% 99.7% $^{211}_{84}Po$	Actinium C'	0.52s	7.42	(99%)		---	0.570 0.898	(0.54%) (0.52%)
$^{207}_{81}Tl$	Actinium C"	4.77m		---	1.42	(99.8%)	0.897	(0.24%)
$^{207}_{82}Pb$	Actinium D	Stable		---		---		---

a This expression describes the mass number of any member in this series, where n is an integer. Example $^{207}_{82}Pb$ (4n + 3)....4(51) + 3 = 207
† Intensities refer to percentage of disintegrations of the nuclide itself, not to original parent of series (All low yield radiations are not included)

2.1.3 Enrichment

Uranium-235 enrichment processes selectively increase the ^{235}U concentration by separating it from the ^{238}U. The method used for many years in the U.S. is the gaseous diffusion process. Laser separation has also been demonstrated in this country, but a facility built to accommodate the process has not yet been brought on-line. Centrifugation is a third separation method used by foreign sources. Uranium feed for the enrichment process is derived from virgin ore or from "very clean" recycled material. Although some uranium is still mill-derived, much of the feed is recycled material from other countries, including Canada (where natural uranium is the reactor feed material). Specifications on acceptable contamination levels limit the feed that may be processed in the U.S. gaseous diffusion plants. Recycling of reprocessed (irradiated uranium) material from DOE's reactors years ago contaminated the diffusion process equipment with transuranics, a portion of which remains in the equipment.

The specific activity of essentially pure uranium depends on its degree of enrichment and normally describes only alpha activity. The beta activity from associated decay products is not included in the uranium specific activity values, but is expressed separately. Consequently, two specific activities (one for alpha and one for beta) are frequently calculated for uranium-bearing materials. Some typical alpha specific activity values are given in Table 2-5 and Figures 2-1, 2-2, 2-3, and 2-4.

Table 2-5. Uranium Specific Activities

Type	Wt. % ^{235}U	Specific Activity (Ci/g) of Mixture
Natural	0.72	7 x 10-7
Depleted	0.20	4 x 10-7
Enriched	2.0	1 x 10-6
Enriched	20	9 x 10-6

For gaseous diffusion enriched uranium, the approximate alpha specific activity of a given uranium enrichment can be calculated from the following formula:

Specific Activity of Enriched Uranium = $(0.4 + 0.38E + 0.0034E^2) \times 10^{-6}$ Ci/g where E = % ^{235}U by weight, enrichment > or = 0.72

Gaseous diffusion, the predominant existing enrichment technology, causes a greater increase in ^{234}U concentration than in ^{235}U concentration. For example, when ^{235}U content is increased from 0.72% (natural) to 2.96%, (an increase of approximately a factor of four), ^{234}U content increases from 0.006% to 0.03%, (a five-fold increase). As a result, the specific activity increases with enrichment, not just because of the replacement of some ^{238}U with ^{235}U, but more significantly because of the increase in the amount of ^{234}U present.

Laser isotopic separation (under research) selectively enriches only the ^{235}U, leaving the ^{234}U with the "tails," or depleted uranium. Therefore, the radiological characteristics of both enriched and depleted uranium will change when compared to conventional separation techniques. Figures 2-3 and 2-4 illustrate this effect.

The specific activity of recycled irradiated uranium varies from the value calculated from the equation given above because that equation is not applicable to recycled material with its added contaminants. For these reasons, specific activities that are calculated from the formula should be

considered approximations only. If exact values of specific activity are required, they should be determined analytically. See Example 1 for the calculation of blending enrichments.

Example 1

One kilogram of 20% enriched uranium is blended with 1 kilogram of 2% enriched uranium.

$SA = [0.4 + 0.38E + 0.0034E^2] \times 10^{-6}$ Ci/g

$SA_{20} = [0.4 + 0.38\ (20) + 0.0034\ (20)^2] \times 10^{-6}$ Ci/g
$= 9.36 \times 10\text{-}6$ Ci/g

$SA^2 = [0.4 + 0.38\ (2) + 0.0034\ (2)^2] \times 10^{-6}$ Ci/g
$= 1.17 \times 10^{-6}$ Ci/g

The specific activity of the resulting mixture is

$[(9.36 + 1.17)/2] \times 10^{-6}$ Ci/g $= 5 \times 10^{-6}$ Ci/g

Figure 2-1. Specific Activity for Mixtures of ^{238}U, ^{234}U, and ^{235}U

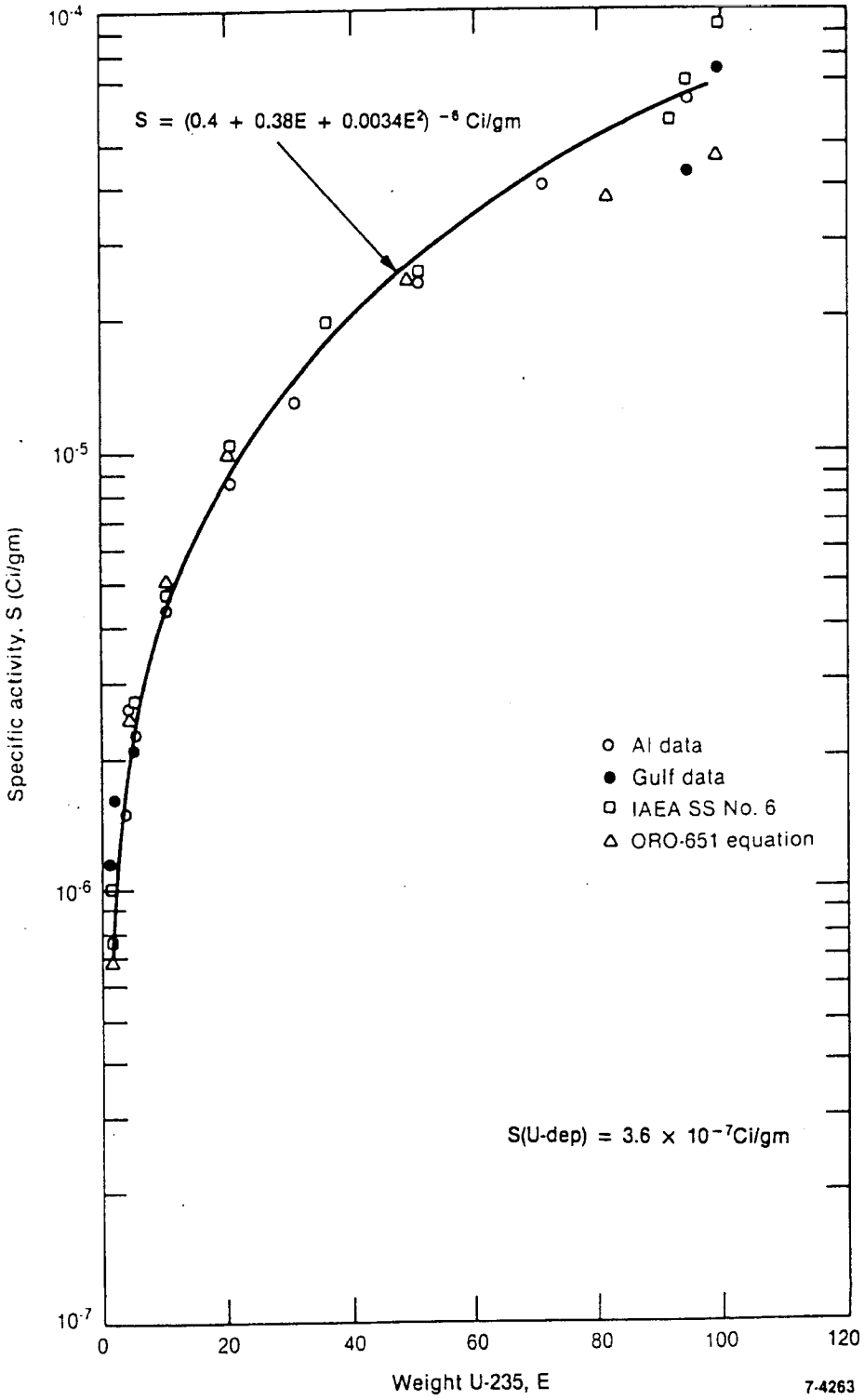


Figure 2-2. % Total Radioactivity by Isotope vs. % Weight ^{235}U Enrichment

Calculated from SA = $(0.4 + 0.38E+0.0034E^2)\ 10^{-6}$ Ci/g (gaseous diffusion process) (NRC Reg Guide 8.11)

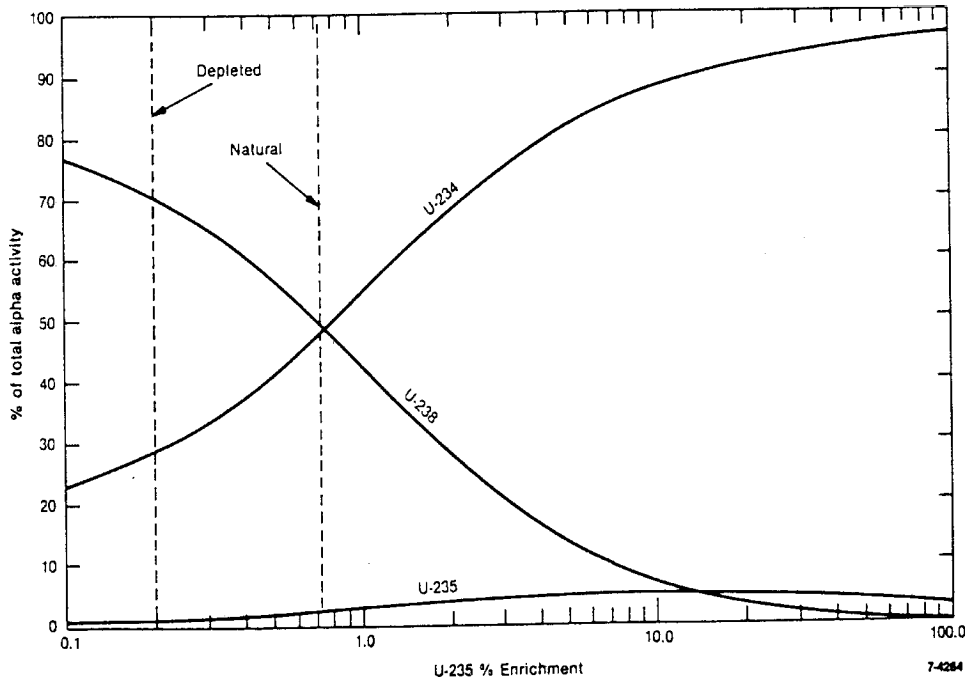


Figure 2-3. Approximate Percent Alpha Activity Contribution for Laser Enriched Uranium

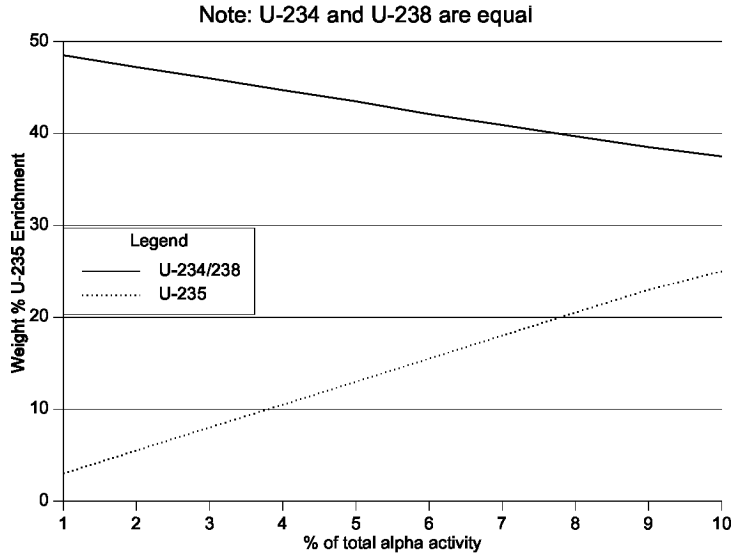
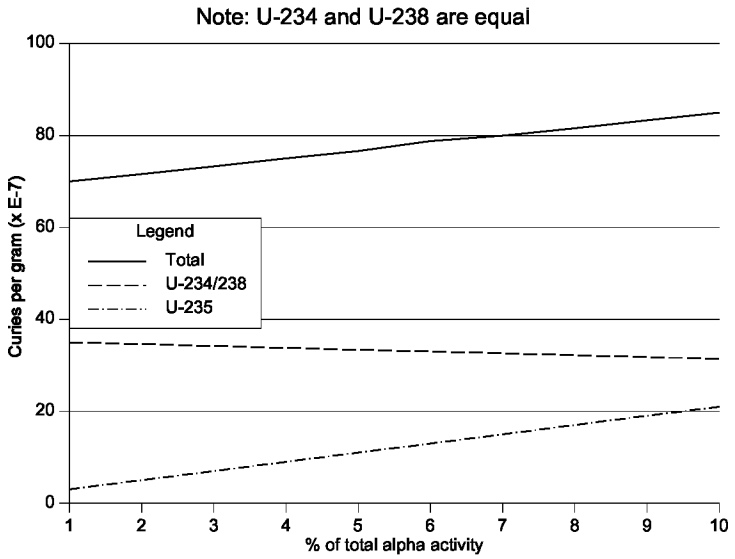


Figure 2-4. Estimated Uranium Specific Activity for Laser Enrichment (of natural uranium)



The Derived Air Concentration (DAC) values for several radionuclides are shown in Table 2-6. Because the ALIs for the three primary uranium isotopes are expressed in activity units, enrichment has little impact on inhalation and ingestion ALIs. However, as illustrated in Table 2-5, as enrichment increases from 2% to 20%, the specific activity increases nine-fold. Consequently, the mass of material that corresponds to one ALI decreases by a factor of nine. The degree of enrichment also affects the controls that are required for external penetrating radiation exposure because of the increase in the amount of gamma-emitting ^{235}U that is present.

As a historical note, some of the earlier documentation refers to the "special curie" of natural uranium, which was defined as 3.7×10^{10} d/s of ^{234}U, 3.7×10^{10} d/s of ^{235}U, and 1.7×10^{9} d/s of ^{235}U. Thus, 1 "curie" of natural uranium was actually slightly more than 2 curies of uranium alpha activity. This essentially obsolete term has caused considerable confusion. Readers are cautioned to be aware of the use of this special curie in the older literature. Use of this unit in any current application is strongly discouraged.

2.1.4 Contaminants from Recycled Uranium and Associated Hazards

Some of the uranium feed material that was handled at DOE facilities had been reclaimed or recycled from reprocessed, spent reactor fuel. The chemical processes by which recycled uranium was purified left trace amounts of transuranic elements (neptunium, americium, and plutonium) and fission products (mainly ^{99}Tc). The recycled uranium also contained trace amounts of uranium isotopes not found in nature, such as ^{236}U. At the minute concentration levels in uranium from fuel reprocessing facilities, the radiological impact of these impurities was negligible in most cases. However, there were many routine chemical processes that tended to concentrate these impurities, either in the uranium product or in reaction by-products, such that radiological controls and environmental monitoring programs must consider these impurities.

The following text discusses the environmental, safety, and health challenges presented by the introduction of recycled uranium into the DOE system for enrichment.

2.1.4.1 Transuranics

Transuranics (neptunium and plutonium isotopes) exist in small quantities in reclaimed or recycled feed materials. In most cases, a regimen of radiological controls based on uranium hazards is adequate to control the additional activity. However, because of their higher specific activities (compared to uranium isotopes), transuranics can represent a significant internal dose concern even at very low mass concentrations. As a result, the ALIs for transuranics are lower than those for uranium isotopes. For example, for a moderately soluble transportability mixture, if ^{239}Pu contamination contributes 0.1% of the total alpha activity in uranium, then it will contribute roughly 14% of the total inhalation dose equivalent (see Example 2). Example 2 illustrates that it takes only 11 parts of ^{239}Pu per billion parts of natural uranium to attain an activity fraction of 0.1%.

Radiological controls based solely on uranium content may provide insufficient protection with increases in the TRU concentration. Processes to recover uranium from by-product streams recover a portion of the impurities as well and may require additional controls to adequately protect individuals when the TRU concentration exceeds 0.1%.

Table 2-6. ALIs and DACs for Uranium and Selected Contaminants in Recycled Uranium

	Inhalation		
Nuclide	Class D *	Class W *	Class Y *
Annual Limits on Intake (Bq values converted from the rounded off ALIs), from 10 CFR 20			
	uCi (Bq)	uCi (Bq)	uCi (Bq)
^{238}U	1 (4 x 10^{4})	8 x 10^{-1} (3 x 10^{4})	4 x 10^{-2} (1 x 10^{3})
^{235}U	1 (4 x 10^{4})	8 x 10^{-1} (3 x 10^{4})	4 x 10^{-2} (1 x 10^{3})
^{234}U	1 (4 x 10^{4})	7 x 10^{-1} (3 x 10^{4})	4 x 10^{-2} (1 x 10^{3})
^{234}Th	NL[1]	2 x 10^{2} (7 x 10^{6})	2 x 10^{2} (7 x 10^{6})
^{234m}Pa	NL	NL	NL
^{231}Th	NL	6 x 10^{3} (2 x 10^{8})	6 x 10^{3} (2 x 10^{8})
^{99}Tc	5 x 10^{3} (2 x 10^{8})	7 x 10^{2} (3 x 10^{7})	NL
^{237}Np	NL	4 x 10^{-3} (1 x 10^{2})	NL
^{238}Pu	NL	7 x 10^{-3} (3 x 10^{2})	2 x 10^{-2} (7 x 10^{2})
^{239}Pu	NL	6 x 10^{-3} (2 x 10^{2})	2 x 10^{-2} (7 x 10^{2})
^{240}Pu	NL	6 x 10^{-3} (2 x 10^{2})	2 x 10^{-2} (7 x 10^{2})
^{241}Pu	NL	3 x 10^{-1} (1 x 10^{4})	8 x 10^{-1} (3 x 10^{4})
^{236}U	1 (4 x 10^{4})	8 x 10^{-1} (3 x 10^{4})	4 x 10^{-2} (1 x 10^{3})
Inhalation DAC, From 10 CFR 835, Appendix A			
	uCi/mL (Bq/m^3)	uCi/mL (Bq/m^3)	uCi/mL (Bq/m^3)
^{238}U	6 x 10^{-10} (2 x 10^{1})	3 x 10^{-10} (1 x 10^{1})	2 x 10^{-11} (6 x 10^{-1})
^{235}U	6 x 10^{-10} (2 x 10^{1})	3 x 10^{-10} (1 x 10^{1})	2 x 10^{-11} (6 x 10^{-1})
^{234}U	5 x 10^{-10} (2 x 10^{1})	3 x 10^{-10} (1 x 10^{1})	2 x 10^{-11} (6 x 10^{-1})
^{234}Th	NL	9 x 10^{-8} (3 x 10^{3})	6 x 10^{-8} (2 x 10^{3})
^{234m}Pa	NL	3 x 10^{-6} (1 x 10^{5})	3 x 10^{-6} (1 x 10^{5})
^{231}Th	NL	3 x 10^{-6} (1 x 10^{5})	3 x 10^{-6} (1 x 10^{5})
^{99}Tc	2 x 10^{-6} (8 x 10^{4})	3 x 10^{-7} (1 x 10^{4})	NL
^{237}Np	NL	2 x 10^{-12} (9 x 10^{-2})	NL
^{238}Pu	NL	3 x 10^{-12} (9 x 10^{-2})	7 x 10^{-12} (3 x 10^{-1})
^{239}Pu	NL	2 x 10^{-12} (8 x 10^{-2})	6 x 10^{-12} (2 x 10^{-1})
^{240}Pu	NL	2 x 10^{-12} (8 x 10^{-2})	6 x 10^{-12} (2 x 10^{-1})
^{241}Pu	NL	1 x 10^{-10} (4)	3 x 10^{-10} (1 x 10^{1})
^{236}U	6 x 10^{-10} (2 x 10^{1})	3 x 10^{-10} (1 x 10^{1})	2 x 10^{-11} (6 x 10^{-1})

[1] NL = Not listed.

* See last paragraph of Section 2.5 for discussion of Class D, W and Y.

Example 2

One gram of natural uranium contains ^{239}Pu contamination to the extent that the ^{239}Pu activity is 0.1% of the uranium alpha activity. The relative inhalation hazards of the two materials are determined by dividing each material's relative activity by its derived air concentration.

U-Nat relative activity = 1
^{239}Pu relative activity = 0.001
U-Nat derived air concentration (W) = 3×10^{-10} µCi/mL
^{239}Pu derived air concentration (W) = 2×10^{-12} µCi/mL

$$\frac{1}{DAC_a(U\text{-}Nat)} = \frac{1}{3x10^{-10}} = 3x10^9$$

$$\frac{0.001}{DAC_a(^{239}Pu)} = \frac{0.001}{2x10^{-12}} = 5x10^8$$

These values represent the relative hazards of the two materials in the mixture.

$$Fraction\ of\ total\ hazard = \frac{5x10^8}{(5x10^8)+(3x10^9)} = 0.14$$

Therefore, ^{239}Pu at 0.1% of the U-Nat activity represents 14% of the potential inhalation dose.
The activity of 1 gram of U-Nat = 2.5×10^4 dps
Therefore, $0.001 \times 2.5 \times 10^4 = 2.5 \times 10^1$ dps = the ^{239}Pu activity in the 1 gram of U-Nat.
The specific activity of ^{239}Pu is 2.27 dps/nanogram:

$$25\,dps/g\,U \times \frac{1\,nanogram\,Pu}{2.27\,dps} = \frac{11\,nanograms\,Pu}{1\,gram\,U}$$

Therefore, 0.1% ^{239}Pu activity fraction corresponds to 11 parts per billion on a mass basis.

Several DOE facilities have adopted specifications on recycled uranium that limit the amount of transuranic alpha activity to 0.1% of the total uranium alpha activity, thus limiting the potential inhalation dose from transuranics to a small fraction of the total. Facilities that handle recycled uranium with higher levels of transuranics should establish a regular program of analyzing feeds, products, and by-products for transuranics, and then modifying control limits and action levels as appropriate to reflect the transuranic content of those materials. This monitoring of the TRU content is essential when the analytical technique used to identify the level of radiological control needed is based on gross alpha counting (such as for air sampling), which does not distinguish the plutonium from the uranium fraction, or chemical analysis for uranium (such as photofluorometric urinalysis) which does not detect plutonium.

Raffinate from refinery operations, MgF_2 from metal production operations, and chemical traps from UF_6 operations have all been observed to have higher TRU-to-U ratios than either reactants/feeds or uranium products. Frequently, reaction by-products are not discarded as wastes but are processed further to recover the remaining uranium. When this occurs, a portion of the impurities is recovered along with the uranium and can become a perpetual radiological control problem. All facilities that process recycled uranium should periodically analyze feeds, products, and by-products for transuranics to ensure that radiological controls are adequate for the mixtures of uranium and transuranic elements that are present.

The uranium isotopes (viewed as contaminants) that will increase due to the recycled uranium feed are ^{232}U, ^{234}U, and ^{236}U. The health and safety risks of ^{236}U are similar to those of natural uranium because its specific activity and radiation emissions are similar (See Table 2-2). Its presence in uranium fuel requires slightly higher enrichments for the same reactor applications, however, because it absorbs neutrons. The increased concentration of the ^{234}U increases the specific activity of any enrichment of ^{235}U. It is expected that the specific activity for a given enrichment would be about double that obtained from enrichment of non-recycled uranium.

The isotope in recycled uranium presenting the greatest potential radiological hazard from external sources is ^{232}U. ^{232}U a daughter product of neutron activation of ^{231}Pa. The health hazards of ^{232}U are primarily due to the rapid buildup of gamma activity of its decay products, particularly from ^{228}Th. The gamma activity buildup is both time- and process-dependent. The ^{232}U decay products form nonvolatile fluorides and will concentrate in cylinders when UF_6 is vapor-fed. The gamma activity in equipment that processes gaseous UF_6 is a function of the mass fraction of ^{232}U present in the gas phase. Estimates indicate that the level of gamma activity within the enrichment cascade equipment would increase by about a factor of 3 due to the presence of ^{232}U. The exposure rates on internal surfaces would increase from 10-20 mrad/h to 30-60 mrad/h; those on external surfaces would increase to about 3-4 mrad/h. The major exposure increase from the ^{232}U occurs in the handling of UF_6 cylinders. Currently, the exposure rate at the external surface of empty UF_6 cylinders is about 50-100 mrad/h. Assuming a ^{232}U concentration of 0.5 ppm based on ^{235}U and a feed enrichment of 1%, a full 10-ton feed cylinder would have a surface exposure rate of about 80 mrad/h. The exposure rate at 30 cm from the surface of an emptied cylinder would be about 500 mrad/h without the shielding provided by material in the cylinder. These values are based on the ^{232}U being in secular equilibrium with its decay products; in reality, it is unlikely that the decay products would reach much more than 50% of equilibrium values.

Product cylinders produced from processing of recycled uranium typically have higher gamma radiation fields than the feed cylinders. At 4% ^{235}U enrichment, the contribution from ^{232}U over time could increase the radiation field at the surface from 80 mrad/h to 300 mrad/h from a full 10-ton cylinder and from 500 mrad/h at 30 cm to 2 rad/h from an empty cylinder. About half of this increase would be

apparent within 2 years of initial usage and the highest levels could occur in 20 years without mitigating actions. Frequent cylinder cleaning can prevent this significant exposure rate buildup. The presence of ^{232}U may also require other changes in processes used to handle cleaning solutions due to the higher gamma radiation present.

2.1.4.2 Technetium

In facilities with significant quantities of ^{99}Tc, radiation monitoring techniques must be able to detect the low-energy beta radiation from this isotope. Individual and area monitoring equipment and techniques selected to measure the 2.29 MeV (E_{max}) beta from ^{234m}Pa may not measure the ^{99}Tc 0.292 MeV (E_{max}) beta effectively. If a mixture of uranium and ^{99}Tc is suspected to be present, the monitoring technique selected must be based on ^{99}Tc or on the actual mixture, rather than on ^{234m}Pa. The ^{99}Tc levels have not been the controlling factor in many situations to date. However, it is important to ensure that monitoring instruments and techniques are adequate to detect ^{99}Tc.

Technetium-99 tends to deposit within enrichment equipment and will "pocket" in the higher enrichment sections of the gaseous diffusion cascade. Special precautions must be taken when evacuating and purging or performing other maintenance work on this equipment. In equipment with accumulations of ^{99}Tc, low energy beta radiation fields of a few rad per hour may be encountered. This radiation is effectively attenuated by the protective clothing required for contamination control (one pair of industrial cloth coveralls, one pair of impermeable (Tyvek) coveralls and heavy neoprene gloves). While the ^{99}Tc should be effectively removed from the Gaseous Diffusion Plant (GDP) product, it will be present in uranium used by other DOE facilities. Because the ALI for ^{99}Tc is higher than that of uranium, inhalation is the controlling concern only in situations where the technetium activity greatly exceeds that of the uranium that is present. Technetium as pertechnetate is also difficult to remove from skin and can therefore cause significant skin doses from skin contamination.

The tendency of technetium to become airborne more readily than uranium can lead to beta contamination in areas where it is not otherwise expected and environmental emissions even when the uranium is effectively confined in the work place. Residues in ventilation systems from high-temperature operations, such as uranium remelting/casting, or uranium chip burning, tend to have higher Tc-to-U ratios than either feed or product material in uranium metal processing facilities. Because of its low atomic weight and relative volatility, technetium also tends to concentrate at the top of the gaseous diffusion cascade, where it becomes an inhalation and effluent concern when the cascade is opened for maintenance. Facilities that handle recycled uranium should 1) analyze feeds, products, and by-products to determine the fate of ^{99}Tc within their processes, then 2) modify monitoring equipment, control limits, and action levels as needed to properly evaluate and control ^{99}Tc hazards.

Environment, safety and heath personnel should also evaluate the presence of and radiological consequences from other fission products impurities in recycled uranium.

2.2 PHYSICAL AND CHEMICAL PROPERTIES

Uranium fuels vary with reactor type. Some reactors use the natural isotopic composition in the fuel. Others use enrichment varying from 2% to > 90%. Because of the radiation-induced growth of uranium metal used in the early reactors, alloys were developed to stabilize dimensional changes. Many of the alloys with favorable dimensional stability characteristics had sizeable neutron absorption cross-sections, resulting in poisoning of the nuclear reaction. Zirconium-clad ceramic uranium dioxide and uranium carbide fuels were found to have acceptable characteristics and are in common use.

2.2.1 Uranium Fuel Processing

The process of reducing uranium ore to metal begins with the discovery and mining of uranium in ore bodies. Most medium grade ore consists of oxides of uranium, of which carnotite ($K_2(UO_2)_2(VO_2.3H_2O)$) is predominant. Although some ore is mined using *in situ* leach techniques, most is hard-rock mined with a small amount removed by open pit mining. Uranium ore is milled by crushing, leaching, extracting, and precipitating, usually to ammonium diuranate ($(NH_4)_2U_2O_7$) commonly called yellow cake. The radioactivity of this product is low because the decay products have been stripped away and it is in an unenriched form. The yellow cake is purified and converted to UF_4 and then further fluorinated to uranium hexafluoride (UF_6). Gaseous diffusion enrichment changes the uranium isotopic, but not the chemical, composition of the gas. The UF_6 is hydrolyzed to uranyl oxyfluoride, which is precipitated with an ammonia solution to ammonium diuranate. This precipitate is filtered or centrifuged, dried, and calcined. The uranium compound is reduced to UO_2 powder, which is pelletized, sintered, and encapsulated in tubes for reactor usage.

Laser enrichment can use feed forms including metal and UF_6.

Steel was an early cladding material that was discontinued because of its thermal-neutron poison characteristics. Fuel bundles used in commercial LWRs are now made of fuel pins that consist of pellets of UO_2. The pellets are stacked into free-standing cladding tubes of a zirconium or zirconium-tin alloy. Differences in fuel design between the two common types of nuclear reactors in use in the United States, pressurized water reactors (PWRs) and boiling water reactors (BWRs), are rod diameter and cladding thickness.

Reactor fuel for the Canadian pressurized heavy water reactors (CANDU-PHWR) is similar but the cladding need not be free-standing. Additionally, the fuel pins are smaller in diameter. Breeder reactors like the Fast Flux Test Facility (FFTF) use a mixture of PuO_2 and depleted UO_2. In the case of the FFTF, the pellets are loaded into stainless steel cladding tubes (which have a smaller effect on fast neutrons).

Uranium carbide (UC_2) microspheres were developed as an alternative to UO_2, primarily for the high-temperature gas-cooled reactor. These fuel particles, developed for high thermal and radiation stability, prevent the release of fuel and fission products over a wide range of conditions.

2.2.2 Uranium Metal

Conversion of UF_6 to uranium metal involves, first, the production of UF_4, commonly called green salt. Enriched uranium green salt is reacted with granular calcium to produce metal slag. This product is then reacted with magnesium or calcium to reduce the material to metal. Depleted uranium green salt is more commonly reacted with magnesium to produce DU metal as a derby. In both cases, most of the uranium decay products are concentrated in the calcium or magnesium slag, leaving the metal relatively pure and with a reduced level of radioactivity. Buildup of decay products to near-equilibrium levels takes about six months.

The metallic uranium is processed into desired forms using machining, melting, casting, and other treatments. This very dense metal is usually alloyed with another metal for greater stability. Uranium is a reactive metal that oxidizes easily. In the newly minted metal, a very thin surface layer tends to undergo rapid oxidation. This surface layer may protect the rest of the metal from further corrosion, and prevent the generation of removable contamination. Certain environmental conditions, particularly moist

air and saline solutions, can accelerate the corrosion of the material over time and produce greater possibility for generating airborne radioactive material. Stored in a dry environment or coated with an anti-corrosion surface treatment, the metal may show no visible signs of corrosion for many years.

Uranium metal may be dissolved using nitric acid, which is also used to passivate ("pickle") the metal to inhibit oxidation.

2.3 RADIOLOGICAL CHARACTERISTICS AND EFFECTS

Uranium isotopes decay by alpha particle emission and some also emit low-energy gamma rays. For Classes W and Y material (See last paragraph of Section 2.5 for discussion of Class D, W and Y), the inhalation hazard from alpha particle release in the respiratory tract is the predominant radiological hazard associated with the alpha-emitting uranium isotopes. The primary uranium decay products, listed in Table 2-2, decay by beta particle emission, most with a small yield of gamma emissions as well. These decay products increase the shallow dose equivalent and lens of the eye dose equivalent resulting from external radiation exposures, due mainly to the 2.29 MeV (E_{max}) beta from ^{234m}Pa. The surface exposure rates shown in Table 2-7 result primarily from beta radiation from decay products. The exposure rates decrease quickly with distance because of the attenuation of the beta radiation and the small yield of the gamma radiation.

Table 2-7. Beta Surface Exposure Rates from Equilibrium Thickness of Uranium Metal and Compounds

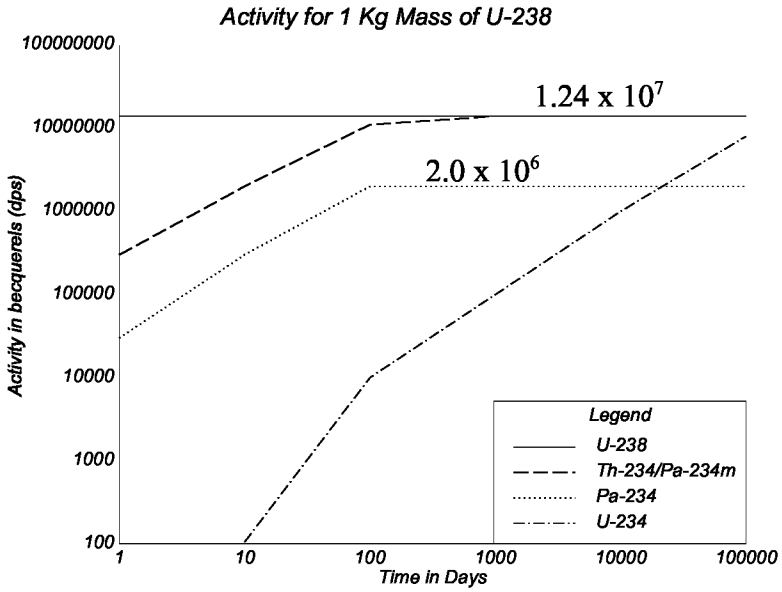
Source	Beta Surface Exposure Rate, mrad/h
U-Nat metal slab	233
UO_2	207
UF_4	179
$UO_2(NO_3)_2 6H_20$	111
UO_3	204
U_3O_8	203
UO_2F_2	176
$Na_2U_2O_7$	167

Beta surface exposure rate in air through a polystyrene filter 7 mg/cm^2 thick.

Because some uranium decay products have short half-lives (on the order of days), those decay products will usually be present with uranium during processing. Figure 2-5 illustrates the ingrowth of the ^{238}U decay products. An assumption of secular equilibrium should not be made until processing is complete because many routine chemical processing steps separate uranium from its decay products. Both the inhalation and external exposure hazards associated with the decay products are increased in areas where the decay products are concentrated. The overall inhalation hazard will typically decrease in those areas as the uranium is removed. In the case of cast uranium metal, the exposure rates from high

beta levels from decay products may be many orders of magnitude greater than the exposure rates from the uranium.

Figure 2-5. ^{238}U Decay Product Ingrowth



2.3.1 Alpha-Neutron External Hazard

The interaction of alpha particles from uranium with the nuclei of fluorine and other low-Z atoms generates neutrons of approximately 2-MeV energy. The magnitude of the neutron flux varies, based on the total activity of uranium (which is a function of enrichment) and the chemical compound in question (mixing of U and F). In the case of UF_6, the typically measured neutron dose rates for cooled storage cylinders are as follows:

Natural-5% enrichment: 0.01-0.2 mrem/h

Very high enrichment (97+%) : 2-4 mrem/h (contact)
1-2 mrem/h (3 ft)

The preceding values were measured with a 9-in. spherical BF_3 rem meter. In general, the exposure potential of personnel to neutrons generated by the (alpha, n) reaction is not high. However, if personnel are required to spend more than a few hours per week in close proximity to containers of uranium fluoride compounds or if their assignments require them to spend time near storage or processing areas for large quantities of uranium fluoride compounds, the exposure to neutrons should be evaluated. This is particularly necessary since the personnel monitoring badges may not be neutron-sensitive or may need to be calibrated to the specific spectra. Penetrating radiation exposures from

photon radiation will not be indicative of neutron exposures. This is because the higher photon penetrating radiation exposures tend to be associated with used but empty containers, where decay products have plated out on the sides, while the maximum neutron exposures are associated with full containers. There is a small additional neutron flux from spontaneous fission associated with full containers. Neutron sensitive personnel monitoring badges are recommended for operations dealing with uranium fluoride compounds.

2.3.2 Mode of Uranium Entry into the Body

Work practices are designed to control radiation exposure to levels that are as low as is reasonably achievable (ALARA). Reductions in exposure time and increases in shielding help reduce external doses. Effective contamination control techniques and ventilation/filtering systems help reduce airborne radioactive material concentrations and resulting internal doses. Where complete contamination control is not reasonable, internal exposure of uranium compounds as aerosols or deposited particulates may occur. The effects of uranium exposure on the body depend on the mode of exposure. External exposure concerns are limited to beta and gamma emissions, of which the gamma field is quite low and the beta field may be mitigated using protective clothing including safety glasses with side shields. Internal exposure and its potential effects through radiological or chemical toxicity depend on the route of entry, and its distribution depends on the solubility of the material. Solubility is complicated by the wide variety of stoichiometric and crystalline uranium compounds. Inhalation and ingestion are most commonly assessed as routes of entry. Although not covered here, entry of uranium into wounds is also a concern, and its distribution depends on its solubility (See sections 5.9 and 5.10 for further discussion). Absorption through intact skin is unlikely. The type of radiation to which the body is exposed and the length of the exposure determine the biological effect of the radiation exposure.

2.3.2.1 Inhalation

Inhalation hazards from uranium result primarily from the alpha emissions. Inhalation of uranium particles and deposition into the respiratory system are dependent on particle size. The nasal-pharynx system filters out most large particles that are still small enough to be inhaled. Larger particles can be inhaled--a common convention is to assume inhalation possible for all particles 10-µm or less aerodynamic equivalent diameter (AED)--but most particles that penetrate to the lower respiratory tract are less than 3- or 4-µm AED. Uranium in the lungs has been shown to exhibit a wide range of retention values. Clearance may occur through physical processes removing particles that are not embedded into the lung by cilia motion to the esophagus. Uranium particles that are soluble in lung fluid are chemically dissolved, and the ions are transported into the bloodstream where they are further distributed. Uranium particles remaining in the lung constitute a potential radiological hazard as they impart their alpha emission energy into the surrounding absorbing tissue, potentially causing significant damage within a small sphere around each particle. Particles removed from the lung to the bloodstream primarily represent a potential chemical hazard.

The significance of these hazards is evaluated using models of uptake and removal recommended by national and international scientific radiation protection organizations. The lung model described in ICRP Publication 66 (ICRP 1994) uses solubility Types of F (fast), M (moderate), and S (slow). In comparison to previous models, this model better describes deposition, retention, and clearance data and decouples physical and chemical clearance processes.

2.3.2.2 Ingestion

Appropriate uranium contamination controls should prevent ingestion of uranium. Nevertheless, the potential exits for accidental ingestion of uranium. Particles inhaled through the mouth and temporarily deposited there are removed from the respiratory system to the esophagus. Deposition and removal of ingested uranium are approximated using the Gastrointestinal (GI) Tract Model adapted from Eve (Eve 1966). This model calculates material transferred from the GI tract to the blood based on solubility classes (ICRP 1979 and IAEA 1994) or based on a single value for all compounds, as described in ICRP Publication 69 (ICRP 1995).

Distribution of uranium transferred into the bloodstream is calculated using a once-through metabolic model. ICRP Publication 30 also provides values for this distribution and excretion to calculate committed doses and long-term tissue retention. Recent models (Wrenn et al. 1994 and ICRP 1995) have been developed to include recycling of uranium back into the blood.

2.4. CHEMICAL TOXICITY

The chemical toxicity of uranium is a primary concern in establishing control limits. A heavy metal, uranium is chemically toxic to kidneys and exposure to soluble (transportable) compounds can result in renal injury. The factors to be considered in determining whether the chemical or radiological hazard is controlling are the enrichment, mode of entry, and the solubility/transportability of the material. Chemical toxicity is a higher risk with soluble material of 10% or less enrichment.

A concentration of 3 μg of uranium per gram (μg U/g) of kidney tissue has traditionally been used as the guideline for controlling the chemical toxicity of uranium. Reference man has a kidney mass of 310 g, so this concentration translates to a total kidney burden of 1 mg. A review of the literature by Leggett (Leggett 1989) suggests that worker exposure to 2 to 6 μg U/g kidney might be tolerated with no serious effects. However, he emphasizes that this range is not necessarily the same as the level causing no detectable damage. He concludes that a lower limit would be prudent until more of the physiological mechanisms of response to uranium in the kidney are better understood. Other studies (McGuire 1991) report that detectable effects from an intake of soluble uranium of 10 mg or less is unlikely and that an intake of 40 mg and perhaps as high as 100 mg is unlikely to cause permanent damage. Other evaluations of toxicity to the kidney concluded that a limit of 1.0 μg U/g kidney is consistent with results in the recent literature.

An airborne concentration limit of 0.2 mg/m^3 was adopted by the Nuclear Regulatory Commission (NRC) and the American Conference of Governmental Industrial Hygienists (ACGIH) for occupational exposures, based on the 3 μg/gm of tissue value. The Occupational Safety and Health Administration (OSHA) has adopted a limit of 0.05 mg/m^3 for soluble uranium and 0.25 mg/m^3 for insoluble uranium. In most DOE facilities, the most conservative of the two standards (OSHA or ACGIH) should be used unless enrichment and solubility dictate more stringent controls based on radiological concerns. Table 2-8 lists airborne concentration limits for transportable uranium that have been published by various organizations.

Table 2-8. Toxicological Limits on Airborne Concentrations of Transportable (soluble) Uranium

Agency	Chronic Exposure Occupational Limit, mg/m^3	Reference
NRC	0.2	Footnote to Appendix B, 10 CFR 20 (NRC 1992a)
ACGIH	0.2	Threshold Limit Values and Biological Exposure Indices for 1997, American Conference of Governmental Industrial Hygienists (ACGIH 1997)
OSHA[(a)]	0.05 (soluble) 0.25 (insoluble)	29 CFR 1910.1000
NIOSH	0.05	National Institute for Occupational Safety and Health

(a) Preferred/recommended limit.

Past limits for single acute inhalation intakes have been set by the International Commission on Radiological Protection in its Publication 6, (ICRP 1964) to 2.5 mg of soluble uranium inhaled in any one day. This value is based on one day's intake at the maximum permissible concentration (at the time) of 210 $\mu g/m^3$. Lawrence (Lawrence 1984) derived acute inhalation intake limits of 15 and 80 mg for Class D and Class W materials, respectively. This derivation is based on not exceeding a kidney burden of 3 µg U/g kidney after a single acute inhalation. NRC regulations at 10 CFR 20 limit the intake of soluble uranium to 10 mg in a week.

Chronic exposure to a concentration of 0.2 mg/m^3 results in a weekly intake of 9.6 mg (40 h/week x 1.2 m^3/h x 0.2 mg/m^3) and a steady-state kidney burden of roughly 900 µg, when the ICRP Publication 30 metabolic model for Class D uranium is used. This same model indicates that an acute intake of 18 mg will result in a prompt kidney burden of approximately 900 µg. However, 10 CFR 20 limits acute exposures to 40 DAC-hours, or 9.6 mg.

Recurrent concerns have arisen about the adequacy of existing limits intended to prevent chemical damage to kidneys. These concerns have focused particularly on the

- lack of data on the effects of combined exposures to UO_2F_2 and HF
- lack of detailed information on effects of short-term exposures to soluble/transportable uranium in the range from 100-1000 mg/m^3
- lack of data on thresholds for repairable injury.

DOE sponsored research to determine the exposure levels that would be expected to 1) have no effect, 2) cause non-lethal injury, and 3) be lethal to 50% of the exposed population (LD 50). Researcher consensus resulted in the uptake levels (in µg U/g) listed in Table 2-9 along with the corresponding total uranium in 70-kg standard man.

Table 2-9. Uranium Levels for Various Effects

Effect	Uranium Absorbed into Bloodstream, µg U per g body weight	Corresponding Class D Uranium Intake in Standard Man, mg	Corresponding Kidney Burden µg U/g Kidney
No effect	0.04	5.9	1.1
Maximal Nonlethal	0.08	11.6	2.2
LD_{50}	2.0	294	54.8

The values for "standard man" are based on the ICRP Publication 30 model for uranium metabolism (47.6% of inhaled Class D uranium is taken up into the bloodstream, and 12% of that goes to the kidneys). For example, the "no effect" value in Table 2.9 corresponds to a kidney burden of (5.9)(.476)(.12) = 0.337 mg. The mass of kidney tissue in standard man is 310 g, so this kidney burden represents 1.1 µg uranium per gram of kidney tissue.

An airborne contamination limit from this "no effect" kidney burden can be derived by calculating the airborne uranium concentration at which chronic exposure would result in a kidney burden that just equals the "no effect" burden. In the illustrative analyses below, the 1500-day component of ICRP Publication 30's kidney retention function is neglected, since this contribution is negligible.

For chronic exposure to a constant concentration, the maximum kidney burden will occur at the equilibrium condition--when the amount of uranium entering the kidney each day equals the amount being removed from the kidney. The daily kidney uptake rate and removal rate are calculated from the following formulas:

$$K = B_r \times C_a \times f_b \times f_k$$

where

K = kidney uptake rate (mg/day)
B_r = breathing rate (m^3/day)
C_a = air concentration (mg/m^3)
f_b = inhaled fraction entering bloodstream (0.476)
f_k = bloodstream fraction entering kidneys (0.12) and

$$R = \lambda K_b$$

where
R = kidney removal rate (mg/day)
$\lambda = 0.693/T_{1/2}$ (day^{-1})
K_b = amount in the kidney (mg)
$T_{1/2}$ = biological half-life of U in kidney = 6 days

To calculate the concentration at which chronic exposure would result in a kidney burden of 0.337 mg, the uptake rate in kidney is set equal to the removal rate for a 0.337-mg kidney burden:

$R = (0.337) \times 0.693/6 = 0.039$ mg/day

$K = B_r$ (m^3/day) x C_a (mg/m^3) x (0.476) x (0.12)

$K = R = 0.039$ mg/day

$B_r C_a$ x (0.476) x (0.12) = 0.039 mg/day B_r

x C_a = 0.68 mg/day

Standard man breathes 9.6 m^3 of air in an 8-hour day, so the resulting concentration limit is 0.68/9.6 = 0.07 mg/m^3. This is 40% higher than the OSHA standard for soluble uranium of 0.050 mg/m^3. Consequently, the OSHA limit is somewhat conservative for exposures to soluble/transportable (i.e., Class D) uranium.

2.4.1 Human Response Indicators

Most data on human response to uranium exposure comes from accidental exposures (generally UF_6 releases). Accidental exposures to UF_6 have resulted in fatalities on at least three occasions. The primary cause of injuries and fatalities has been HF that was formed by hydrolysis of UF_6, rather than exposure to UF_6 itself. Several individuals who received high, non-fatal exposures experienced pulmonary edema, nausea, vomiting, abdominal cramps, and chemical burns on the skin due to HF exposure. In addition, urinary abnormalities, such as transient albuminuria (albumin in urine) and the presence of red cells and casts, were observed, as was retention of nitrogenous products such as urea and non-protein nitrogen in the blood.

The urinary and blood abnormalities are indicators of kidney damage, and are the result of inhibited resorption in the tubules. Animal studies indicate that urinary abnormalities can be observed after exposures that are well below lethal levels. In addition, urinary abnormalities such as proteinuria (protein in urine), glucosuria (glucose in urine), and polyuria (increased urine volume) have all been observed following uranium exposure, as has the presence of certain enzymes in urine. Of all these abnormalities, glucosuria appears to be the most sensitive and most nearly proportional to uranium exposure.

Once absorbed into the blood, uranium is distributed to bone and kidneys, with a portion of the uptake being generally distributed throughout the body. For inhaled uranium, residence time in the lungs depends upon the solubility of the material. Material that is deposited in the lungs is cleared via the bloodstream, the pulmonary lymph, and the gastrointestinal (GI) tract. Approximately 1 % of the uranium is absorbed into the bloodstream from the GI tract.

In the event of an acute exposure to highly transportable (Class D) uranium compounds, urine samples should be collected 3-4 hours post-exposure and analyzed for uranium as soon as possible. If the uranium concentration is less than 2.0 mg/L, it is unlikely that any significant kidney damage has occurred or will occur. However, it is important to check the urine for biological indicators of damage at any exposure above 2.0 mg/L. While the most sensitive indicators are increased volume and glucose levels, these are useful only if data on what is "normal" for the individual involved are available. Lacking that information, it is best to check for albuminuria as an indicator of kidney damage. If kidney damage is suspected, a specialist in urinary disorders should be consulted. In general, a urine uranium level greater than 6.0 mg/L will produce some level of albuminuria. A level of 20 mg/L indicates a very serious exposure with potentially life-threatening consequences and would indicate the need for immediate hospitalization.

2.4.2 Transfer to the Fetus

Little information exists on the placental transfer or developmental toxicity of uranium isotopes (Sikov et al 1992). The data available with pregnant rats suggest that the effects produced from exposure to uranium may be due to chemical toxicity to the pregnant animals and their embryos/fetuses. Fetoplacental concentrations of uranium peak one day following intravenous injection of a pregnant rat. Although concentrations in the placenta decrease thereafter, the concentration in the fetal membranes remains relatively constant. Selective deposition in some fetal organs will occur when exposure is during the fetal developmental stages (NRC 1992b).

Data from animal experiments suggest that the distribution pattern of uranium is fairly uniform, especially at the early stage of gestation. Concentrations of uranium in the embryo/fetus are taken to be the same as those in the maternal soft tissues (excluding the kidney) during the first two months, and they progressively increase thereafter. Following transfer into the embryo-fetus, uranium activity is assumed to be distributed uniformly and to remain without excretion.

2.5 CHEMICAL VERSUS RADIOLOGICAL HAZARDS

Both the chemical and radiological hazards of uranium are moderate compared to those of other industrial materials and radionuclides. Table 2-6 provides 10 CFR 835 derived air concentration values for selected radionuclides. Table 2-10 compares Threshold Limit Values (TLV) published by ACGIH for uranium and selected other metals. The comparison of TLVs is presented to provide perspective on the need for uranium workplace controls, as compared to other hazardous materials. Since these materials affect the body in different ways, this should not be considered a comparison of relative hazards.

The predominant hazard associated with uranium exposure depends upon its degree of enrichment, its chemical form, and its physical form. The degree of enrichment determines the gamma radiation intensity and the overall specific activity. The effect that enrichment has on specific activity is illustrated in Figure 2-2. That figure (adapted from NRC Regulatory Guide 8.11) also gives 3.6×10^{-7} Ci/g as the specific activity of depleted uranium and lists the formula used in Section 2.1.1 for calculating specific activity of enriched uranium.

Table 2-10. 1999 ACGIH Threshold Limit Values (TLVs) for Selected Metals

	Soluble and Insoluble TLV	
Metal	**TLV-TWA, mg/m^3**	**TLV-STEL, mg/m^3**
Uranium	0.2	0.6
Beryllium	0.002	--
Lead	0.05	0.45
Mercury vapor, all forms except alkyl	0.05	--
Arsenic	0.01	--

TLV-TWA = Threshold Limit Value, Time-Weighted Average
TLV-STEL = Threshold Limit Value, Short-term Exposure Limit

The relative activities of the primary uranium isotopes are also significantly affected by the degree of enrichment (see Figure 2-2). The figure shows that total activity is due chiefly to ^{238}U for depleted and ^{234}U for enriched uranium, while ^{235}U accounts for little of the total activity, even at very high enrichments.

Chemical form determines solubility and consequent transportability in body fluids. ICRP Publication 30 classifies all materials into three inhalation classes–D, W, and Y (soon to be Types F for fast, M for moderate, and S for slow). Class D is most transportable (pulmonary removal half-time of days), Class Y the least transportable (removal half-time of years), and Class W an intermediate category (removal half-time of weeks). The transportability of an inhaled or ingested material determines its fate within the body and, therefore, the resulting radiation dose or chemical effect. Table 2-11 lists several common uranium compounds and their assigned transportability classes.

Table 2-11. Inhalation Classification for Some Uranium Compounds

Uranium hexafluoride	UF_6	Class "D"[(a)]
Uranyl fluoride	UO_2F_2	Class "D"[(a)]
Uranyl nitrate	$UO_2(NO_3)_2$	Class "D"
Uranyl acetate	$UO_2(C_2H_3O_2)_2$	Class "D"
Uranyl chloride	UO_2Cl_2	Class "D"
Uranyl sulfate	UO_2SO_4	Class "D"
Uranium trioxide	UO_3	Class "W"
Uranium tetrafluoride	UF_4	Class "W"[(a)]
Uranium oxide	U_3O_8	Class "Y"[(b)]
Uranium dioxide	UO_2	Class "Y"[(b)]
Uranium tetroxide	UO_4	Class "W"
Ammonium diuranate	$(NH_4)_2 + U_2O_7$	Class "W"[(b)]
Uranium aluminide	UAl_x	Class "Y"[(a)]
Uranium carbide	UC_2	Class "Y"
Uranium-zirconium alloy	UZr	Class "Y"
High-fired uranium dioxide	UO_2	Class "Y"[(b)]

[(a)] "D" and "W" and "Y" are inhalation solubility classes established by the ICRP: "D" class material is very soluble, with lung retention time in days; "W" class material is moderately soluble, with lung retention time in weeks; "Y" class material is relatively insoluble, with lung retention time in years.

[(b)] Ammonium diuranate is known to contain uranium as UO_3, and should not be assigned to a single inhalation class. The solubility of uranium oxides is very dependent on heat treatment. The rate of oxidation may also affect the solubility. Although references assign inhalation classes to various uranium compounds, it is recommended that solubility studies be performed to characterize the actual materials present.

This listing is intended to provide general guidance only, as a given material's transportability will depend upon a number of parameters including its processing history. It is recommended that each facility determine the transportability of materials it handles using one of the accepted techniques. Physical form influences potential hazards since non-dispersible forms generally do not constitute an ingestion or inhalation hazard.

Because inhalation of uranium potentially poses both radiological and toxic hazards, one must determine which hazard is most limiting and whether or not either hazard can be ignored under certain circumstances. When radiological hazards are limiting, chemical hazards can generally be neglected, except in overexposure situations. When chemical hazards are limiting, radiological hazards can be neglected only if radiation doses are below regulatory concern. Radiological monitoring is required by DOE for individuals who are likely to exceed 100 millirem CEDE in a year. Therefore, it is prudent to calculate organ doses and CEDE for all confirmed intakes, since additional exposures in the same year may result in a total dose exceeding the mandatory individual monitoring threshold. Even in low-potential exposure level situations, a comprehensive dosimetry/control program can prove invaluable in public relations concerning possible future legal litigation.

The limiting hazard (chemical or radiological) depends on the transportability (solubility in body fluids), enrichment, and duration of exposure (acute or chronic). As discussed in Section 2.4, the "no effect" value of intake corresponds to a kidney burden of 0.337 mg. The 0.337 mg kidney burden and ICRP Publication 30 metabolic models are used in the following examples to determine the relative hazards for acute exposure situations.

The 0.337 mg kidney burden corresponds to a chronic exposure of 0.07 mg/m^3. OSHA exposure limits for uranium are 0.05 mg/m^3 for soluble forms and 0.25 mg/m^3 for insoluble forms. These exposure limits are used to determine the relative hazards for chronic exposure situations. For radiological considerations, soluble forms of uranium are considered to be Class D and insoluble forms, Classes W and Y.

To determine which hazard is limiting for an acute exposure, the intake corresponding to "no effect" kidney burden is first calculated and appropriate annual limit on intake (ALI) determined. The formula for specific activity is solved in order to determine the enrichment at which the "no effect" intake is equal to one ALI. For chronic exposure scenarios, the OSHA exposure limit and appropriate derived air concentration (DAC) are used. The formula for specific activity is solved to determine the enrichment at which the DAC is equal to the OSHA limit. These enrichments form the "dividing line" between chemical and radiological effects as the limiting hazard. Exposures to higher enrichments are limited by radiological effects; exposures to lower enrichments by chemical effects.

Example 3a provides the methodology for determining the "dividing line" enrichment for the acute exposure scenario. Example 3b provides the methodology used for the chronic exposure scenario. The following variables are used in these examples:

f_b = fraction of inhaled uranium that promptly enters the bloodstream
f_k = fraction of uranium in bloodstream that enters kidneys
SA = specific activity of uranium in microCi/g obtained from ALI/intake or DAC/concentration
B_r = breathing rate for standard man = 2,400 m^3/year

Table 2-12 shows the values used for f_b and f_k, ALI, and the resulting "dividing line" enrichments for acute and chronic exposures. Several aspects of these derivations must be kept in mind when using this information. First, the derivation is based on standard metabolic models and therefore does not necessarily reflect the effects of a uranium uptake on a real person. Because individual metabolism's will not necessarily agree with the model, the enrichment at which chemical and radiological effects are equally limiting cannot be precisely determined. Uncertainty in the relationship between enrichment and specific activity introduces additional imprecision. Consequently, exposures for both chemical and radiological impact for uranium uptakes at enrichments near the calculated "dividing line" enrichment should be evaluated.

Table 2-12. Determination of "Dividing Line" Enrichments Above Which Radiological Monitoring Requirements Become Limiting

Class	f_b	f_k	Annual Limit on Intake µCi	Specific Activity of "Dividing Line" Enrichment for Radiological Dose Limit	"Dividing Line" Enrichment for Radiological Dose Limit	"Dividing Line" Enrichment for 2% Monitoring Threshold
D	0.476	0.12	1	169.5 µCi/g	(b)	7.38%
W	0.12	0.12	0.7(a)	29.9 µCi/g	52.8%	0.52%
Y	0.05	0.12	0.04	0.71 µCi/g	0.82%	(c)

(a) ICRP Publication 30 lists Class W ALIs of 0.7 µCi for ^{234}U and 0.8 µCi for ^{235}U and ^{238}U. This difference is the result of rounding to one significant figure. Non-rounded values for the three isotopes are all approximately 0.75 microCi.
(b) The resulting enrichment is greater than 100 %. Consequently, chemical toxicity is limiting for acute exposures to Class D uranium approaching the radiological dose limit.
(c) The resulting enrichment is lower than that of depleted uranium. Consequently, radiological concerns are limiting for acute exposures to Class Y uranium at the monitoring threshold.

The impact of the requirement to perform individual radiological monitoring at 2% of the regulatory dose limits can be assessed by reducing ALIs by a factor of 50, then repeating the calculations described in Examples 3a and 3b. Table 2-13 summarizes the results of these calculations.

The effects that enrichment, chemical form, and physical form have on the hazards associated with uranium are summarized in Table 2-13. The comparison of relative chemical and radiological hazards is based on a derived kidney burden resulting from an acute exposure at the "no effect" threshold. The effect of using the OSHA exposure limits of 0.05 mg/m^3 for soluble forms of uranium (Class D) and 0.25 mg/m^3 for insoluble forms (Classes W and Y) is shown for chronic exposures. The derivations used here can be applied to any limit on radiological or chemical toxicity, be it a regulatory or an internal dose control limit. It should be emphasized, however, that the radiological impact should be considered for all intakes, even for exposure situations where chemical toxicity is limiting.

Table 2-13. Impact of Requirement To Monitor at 2%

Enrichments above which radiological concerns predominate

	Acute		Chronic	
Transportability Class	**Using 100% of Radiological Limit**	**Using 2% of Radiological Limit**	**Using 100% of Radiological Limit**	**Using 2% of Radiological Limit**
D	(1)	7.38%	18%	(2)
W	52.8%	0.52%	12.8%	(2)
Y	0.82%	(2)	(2)	(2)
(1) Chemical toxicity concerns are limiting at all enrichments. (2) Radiological effects are limiting at all enrichments.				

Example 3a - General Solution, Acute Exposure

Step 1. Determine the intake that results in a kidney burden of 0.337 mg:

$$SA = \frac{0.337}{f_b \times f_k \times ALI}$$

where SA = specific activity, micoCi / g
ALI = annual limit on intake, microCi
f_b = fraction of uptake that promptly enters bloodstream
f_k = fraction of activity in bloodstream that enters the kidney

Step 2. Use the quadratic formula and equation for determining specific activity to calculate the enrichment that corresponds to the specific activity obtained in Step 1.

$$SA = (0.4 + 0.38E + 0.0034E^2)\ \mu\ \text{Ci / g}$$

$$0.0034E^2 + 0.38E + (0.4 - SA) = 0$$

$$E = \frac{-0.38 \pm \sqrt{(0.38)^2 - 4(0.0034)(0.4 - SA)}}{2(0.0034)}$$

Step 3. One solution will be less than zero. The other will be the enrichment that is the "dividing line" between chemical and radiological effects.

Example 3b - General Solution, Chronic Exposure

Step 1. Determine specific activity at which chronic exposure results in being exposed to one Derived Air Concentration (e.g., ALI divided by breathing rate) at the OSHA exposure limit. The ALI for Class D is used with the OSHA exposure limit for soluble forms of uranium. The ALIs for Classes W and Y are used for the OSHA exposure limit for insoluble forms of uranium.

$$SA = \frac{ALI}{(OSHA\ Exposure\ Limit) \times B_R} \times Unit\ conversion\ factors$$

where SA = specific activity, microCi / g

ALI_D = 1 microCi

ALI_W = 0.7 microCi

ALI_Y = 0.04 microCi

B_R = Breathing Rate (2,400 m^3 / year)

OSHA Exposure Limit (soluble) = 0.05 mg / m^3

OSHA Exposure Limit (insoluble) = 0.25 mg / m^3

Step 2. Use quadratic formula and equation for determining specific activity to calculate enrichment which corresponds to the specific activity obtained in Step 1.

$$SA = (0.4 + 0.38E + 0.0034E^2)$$

$$0.0034E^2 + 0.38E + (0.4 - SA) = 0$$

$$E = \frac{-0.38 \pm \sqrt{(0.38)^2 - 4(0.0034)(0.4 - SA)}}{2(0.0034)}$$

Step 3. One solution will be less than zero. The other will be the enrichment that is the "dividing line" between chemical and radiological effects. If both solutions are less than zero, then radiological effects are always limiting.

2.6 INDUSTRIAL HAZARDS

The principal industrial hazards associated with uranium are fires, hydrogen generation, generation of oxides of nitrogen, and associated mechanical hazards characteristic of heavy objects, i.e., back injuries from lifting, dropping heavy parts on feet, etc. Hydrogen fluoride (HF) and oxides of nitrogen (NO_x) are by-products or reactants of common chemical processes. Hydrogen (H_2) can be generated by reaction of water with uranium metal, and finely divided uranium or uranium chips with a large surface area to volume ratio can ignite spontaneously.

2.6.1 Hydrogen Fluoride

Hydrogen fluoride is an extremely corrosive acid that is relatively volatile in its anhydrous form. Anhydrous HF is a reactant for the production of UF_4 from UO_3, a by-product of the production of UF_4 from UF_6, and is generated whenever UF_6 is released to the atmosphere (H_20 in air + $UF_6 \rightarrow UO_2F_2$ and HF). External contact with HF results in chemical burns of the skin, while exposure to airborne HF causes chemical burns/irritation of the eyes, nose, and throat. Significant inhalation can result in pulmonary edema. Chronic exposure to excessive fluoride concentrations results in increased radiographic bone density and may eventually cause fluorosis (osteosclerosis). In general, individuals can smell HF at levels of 0.02-0.2 mg/m^3, much lower than the TLV of 2.5 mg/m^3. The TLV was set based primarily on the irritation of eyes and mucous passages rather than on permanent damage. Because an airborne concentration of 10 mg/m^3 is intolerable, personnel exposed to such levels will evacuate the area if they are able to do so. Exposure for as little as 15 minutes to an airborne concentration of 20-30 mg/m^3 may prove fatal (pulmonary edema). The AIHA Emergency Response Planning Guides (ERPGs) for HF are as follows: ERPG-3, 42 mg/m^3; ERPG-2, 17 mg/m^3; and ERPG-1, 4 mg/m^3. The NIOSH IDLH value is 25 mg/m^3.

2.6.2 Nitric Compounds

Nitric acid is widely used for digesting uranium metal and uranium-bearing compounds and for "pickling" metal products to inhibit oxidation. Concentrated nitric acid gives off fumes that cause irritation to eyes, mucous membranes, and skin. Significant inhalation can result in pulmonary edema. The ACGIH TLV-TWA and TLV-STEL values for nitric acid are 2 ppm and 4 ppm, respectively.

When uranium materials, especially metal, are dissolved in nitric acid, oxides of nitrogen (NO_x) are generated. The term NO_x is applied to mixtures of nitric oxide (NO) and nitrogen dioxide (NO_2). The ACGIH TLV-TWA and STEL are 25 ppm and 35 ppm, respectively. Exposure to NO_2 can cause eye irritation, coughing, mucoid frothy sputum, shortness of breath, chest pain, pulmonary edema, cyanosis, tachypnea (abnormal rapid breathing), and tachycardia (abnormal rapid heartbeat).

2.6.3 Hydrogen Gas

Hydrogen gas (H_2) is used as a reactant in the production of UF_4 from UF_6 and in the reduction of UO_3 to UO_2, an intermediate step in the production of UF_4 from UO_3. The H_2 is usually generated by dissociating ammonia, so associated ammonia rather than hydrogen is frequently identified as the reactant in those processes. Any facility where H_2 is used as a reactant should include design features (e.g., H_2 monitors, roof vents, etc.) to ensure that hydrogen accumulations do not occur. Generally, H_2 hazards and control features are identified in facility Documented Safety Analyses. Hydrogen can also be generated when moisture contacts uranium metal, especially finely divided uranium metal such as

machining chips. Care must be taken to ensure that H_2 generated in this manner does not accumulate (in closed drums or storage containers for example).

2.6.4 Fire

Finely divided uranium metal is highly reactive or pyrophoric, capable of igniting spontaneously. This type of material should be handled and stored in a manner that minimizes fire potential. Typically, machining chips are stored under water or machining oil in open storage containers so that any H_2 generated does not accumulate. Neither water spray, CO_2, nor halon extinguishers are effective in fighting uranium fires. In fact, halon may be explosive if directed at burning uranium and can produce very toxic fumes and gases. Small uranium fires can be smothered in MET-L-X powder (a mixture of sodium chloride and potassium carbonate). Larger fires, involving drums of machining turnings, for example, can be controlled by immersing the burning container in water. Even this will not immediately extinguish the fire because the hot uranium metal dissociates the water into H_2 and O_2, providing fuel and oxygen for the fire. If the quantity of water is sufficient, eventually the water will provide enough cooling to extinguish the fire, but a significant amount of water can boil away in the process. If the water level is allowed to fall low enough to uncover the uranium while the fire is still burning, it will resume burning visibly. DOE-HDBK-1081-94, Primer on Spontaneous Heating and Pyrophoricity, (DOE 1994e) contains additional guidance.

3.0 RADIATION PROTECTION

An effective radiation protection program at a uranium facility requires scrupulous attention to controlling both internal and external doses. The radiation protection program should ensure the detection and quantification of all types of radiation (i.e., alpha, beta, neutron, gamma, and x-ray) over wide energy ranges. The radiation detection instruments should be properly calibrated and routinely checked. Emphasis should be on establishing controls for internal and external radiation exposure using ALARA guidelines. Prompt and accurate assessment is important in determining each individual's dose and in establishing an accurate historical record. This section defines the basis for establishing a comprehensive radiation protection program.

3.1 REGULATIONS AND STANDARDS

DOE has established occupational radiation protection regulations in 10 CFR 835, Occupational Radiation Protection. DOE has provided supporting and clarifying guidance in the DOE G 441.1 series of Guides, DOE-STD-1098-99, Radiological Control, and DOE Radiological Control Technical Positions. Other related source documents include publications of the EPA, ANSI, ICRP, NCRP, and UNSCEAR. Individual states may also have their own radiological control regulations, with equivalent or more restrictive requirements than the Federal regulations.

3.2 RADIATION PROTECTION PROGRAMS

An effective radiation protection program consists of a group of related and integrated functional elements. The documentation that describes the DOE activity's program to control occupational radiation protection is referred to as the documented radiation protection program (RPP). Although the actual titles and contents of the functional elements are left to the discretion of DOE's operating entities, DOE G 441.1-1A, Management and Administration of Radiation Protection Programs (DOE 2003a), suggests the following, based on the content of 10 CFR 835:

- Organization and Administration
- ALARA Program
- External Dosimetry Program
- Internal Dosimetry Program
- Area Monitoring and Control
- Radiological Controls
- Emergency Exposure Situations
- Nuclear Accident Dosimetry
- Records
- Reports to Individuals
- Radiation Safety Training

Each of these functional elements is discussed in more detail below.

3.2.1 Organization and Administration

This functional element addresses the overall administration of the program, including the documented RPP itself, various organizational and institutional issues, and program assessment. DOE G 441.1-1A and the RCS provide detailed guidance on implementing these requirements.

Although 10 CFR 835.101 requires that DOE activities be conducted in compliance with a documented RPP, the rule does not establish specific requirements for RPP format and content. Due to the wide range of activities undertaken by and for DOE, there is significant flexibility in these provisions. Cognizant DOE line management determines the acceptable format and content of the documented RPP. However, the documented RPP shall address each requirement of 10 CFR 835 and shall be approved by DOE (10 CFR835.101). Any changes that decrease the effectiveness of the RPP shall be approved by DOE before implementation (10 CFR835.101).

Internal audits of the RPP, including examination of program content and implementation, shall be conducted through a process that ensures all functional elements are reviewed no less frequently that every 36 months (10 CFR 835.102). An effective quality assurance program for radiation protection should include establishment of appropriate standards of performance for essential activities and equipment, with an effective system of documentation and traceability of those activities and of the use of the equipment. Proper maintenance of those records will be necessary for reference purposes. Additional requirements and guidance are provided in 10 CFR 830, Nuclear Safety Requirements (DOE 2001d), DOE O 414.1B (DOE 2004), Quality Assurance, and their associated guides. Specific guidance applicable to RPPs is provided in DOE G 441.1-1A.

3.2.1.1 Administrative Controls

In any facility that handles radioactive materials, the major controls protecting workers, the public, and the environment are physical design features, such as structures and installed equipment, that shield, contain, and confine the radioactive materials. However, to allow useful work to be performed in the facility and to ensure that its protective features remain effective, a number of administrative controls are ordinarily required. These controls are usually described in and implemented through a series of policy statements and procedures related to the operations and maintenance activities to be carried out in the facility. All personnel who work in controlled areas should be familiar with the administrative controls that apply to their work. Changes or additions to administrative controls should be effectively communicated to all persons who may be affected.

Radiation Protection Procedures

A uranium facility should have a written policy on radiation protection, including a policy on keeping exposures ALARA.

To ensure facility activities are executed safely and in a manner that consistently meets management expectations, documented procedures should provide detailed instructions for implementing various functional elements of the RPP. Written procedures shall be developed and implemented as necessary to ensure compliance with 10 CFR 835, commensurate with the radiological hazards created by the activity and consistent with the education, training, and skills of the individuals exposed to those hazards (10 CFR 835.104). Responsibilities and actions required of management and workers should be clearly and unambiguously stated. It is not necessary for written procedures to be developed and implemented for all of the requirements of 10 CFR 835. Written procedures should be developed and employed under the following circumstances:

- Worker health and safety are directly affected;
- the expected outcome for the process or operation requires that a specific method be followed;

- the process or operation is infrequently used and competence training cannot assure adequate implementation; or

- to document the approved method to implement specific processes or operations.

In evaluating the need for written procedures, consideration should be given to the level and extent of the radiological hazards, the complexity of the measures required to achieve compliance, and the education, training and skills of the individuals who must implement those measures. Under such a regimen, a low hazard activity employing a stable staff of highly educated and skilled workers having demonstrated an advanced knowledge of radiation protection principles and practices could have fewer and less detailed procedures than a higher hazard activity employing a transient workforce with less knowledge of radiation protection practices and principles. The DOE G 441.1 series of Guides provide additional guidance regarding specific procedural aspects of the RPP.

All radiation protection procedures and controls should have formal, recognizable technical bases for limits, methods, and personnel protection standards. Procedures should be adequately documented, updated periodically, and maintained in a centralized historical file. A control system should be established to account for all copies and ensure all new procedures are included in the historical files. A designated period of time for maintaining historical files should be established. ANSI/HPS N13.6, Practice for Occupational Radiation Exposure Records Systems (ANSI/HPS 1999) provides guidance on maintaining historical files. In addition, radiation protection procedures should have a documented approval system and established intervals for review and/or revision. A tracking system should be developed to ensure that the required reviews and revisions occur. Guidance for writing procedures can be found in DOE/NE/SP-0001T, Writer's Guide for Technical Procedures (DOE 1991a).

Management Commitment

Management commitment to safety is the most important characteristic of an effective radiological control program. If the management commitment to safety is strong, the radiological control program will be valued and respected. The radiological control program should be provided adequate authority to permit performance of necessary assignments and program implementation. Management commitment to the ALARA concept is particularly important (see Article 111 of the RCS). Adequate personnel, equipment, and funding should be available as a part of this commitment.

Radiological Control Organization

The radiological control organization should be structured so that all of the activities required to provide support to line management and workers can be accomplished.

Radiological Control Organization Independence and Reporting Level

The radiological control organization should be independent of the line organization responsible for production, operation, or research activities and should have an equivalent reporting level. Because radiological control personnel should have the authority to balance operations with safety, they should not report directly to the administrators of operations. When shift work is involved, the operations shift supervisor may make minor radiological control decisions in support of the shift's Radiological Control Technicians (RCTs); however, decisions involving basic policies and procedures should be directed to a separate radiological control organization.

If a safety organization includes the radiological control program, it must be high enough in the company to allow direct access to the company president or equivalent. If the radiological control program is administered by a separate radiological control organization, that organization must also be in a position to have direct access to the company president. This is to safeguard the program from the pressures of production that exist in the operational environment and to keep it independent of operating organizations.

A system of guides, policies, and procedures should be established to clearly identify the inter-relationships, responsibilities, and authorities of those involved with the development, operation, and maintenance of the facility and the health and safety of the employees. These guides, policies, and procedures should be documented and reviewed at least once every year.

Adequacy of Personnel and Equipment

A sufficient number of qualified and, where required, certified radiological control personnel must be available to perform necessary tasks for support of uranium facility startup and operation. Sufficient equipment, including protective clothing, respiratory protective equipment, and radiation detection instrumentation should be available to support RCTs and operating personnel in the performance of work in controlled areas.

Staffing and Staff Qualifications

A cadre of operating and maintenance personnel who have experience in the operation of a uranium facility should be established during the construction of a new facility. The remainder of the operating and maintenance staff should be hired as soon as possible and should receive formal and informal training from the experienced personnel. This step is extremely important to enable all personnel to grow with the facility and learn the details of the operations. Once operations start, potential problems already should have been identified, and engineering or administrative changes should have been made to resolve them.

Staffing in the radiological control organization requires technicians and professionals in many support areas. A successful radiological control program is highly dependent upon the availability of adequate staff support in disciplines such as environmental monitoring, instrument maintenance and calibration, internal and external dosimetry, meteorology, safety analysis, and risk management.

Radiological Control Technician Training

A thorough RCT training program should be established at uranium facilities. Before uranium operations begin, a trained and qualified staff of RCTs should be present. All RCT training should be accomplished in accordance with the RCS and DOE-HDBK-1122-99, Radiological Control Technician Training Program (DOE 1999d).

Professional Staffing and Qualifications

The senior staff of the radiological control organization should include health physicists and other professionals with four-year degrees in science or engineering. A continuing training program should be established for facility personnel. Pursuit of certification by the American Board of Health Physics for senior and professional staff members should be encouraged. At least one professional staff

member at the uranium facility should have a minimum of three years of radiological control experience in the operation of uranium facilities.

Technician Staffing and Qualifications

Recommendations for minimum entry-level requirements for RCTs are given in the RCS and the Radiological Control Technician Training Program. They include a high school education or equivalency and knowledge of certain scientific fundamentals. If a two-year degree in nuclear technology or an equivalent discipline is locally available, completion of such a program should be encouraged.

Where possible, the RCTs and other members of the radiological control staff should have a minimum of one year's experience working at a uranium facility. Such experience is an important prerequisite to allow them to work unsupervised. Personnel hired without such experience should work an internship of six months under the leadership of a qualified RCT or supervisor with experience in that facility. RCTs should be encouraged to pursue registration by the National Registry of Radiation Protection Technologists.

Training Staff Qualifications

All training instructors and materials should meet the requirements of DOE Order 5480.20A Ch.1. The RCS provides additional guidance. Each uranium facility should develop performance-based training that reflects radiological conditions present at the facility. This training should be monitored to ensure that site-specific, worker-performance-based measures, and practical factors are included in the uranium training.

Health Physicist Training Involvement

Facility health physicists should have comprehensive knowledge of all of the material on uranium radiation safety included in the training programs for radiation workers and RCTs. In addition to the previously discussed RCT training material, DOE has developed several other radiation safety training courses and qualification standards which may provide useful information. These documents include:

DOE-HDBK-1113-98, Radiological Safety Training for Uranium Facilities (DOE 1998c)
DOE-HDBK-1130-98, Radiological Worker Training (DOE 1998d)
DOE-HDBK-1131-98, General Employee Radiological Training (DOE 1998e)
DOE-STD-1107-97, Knowledge Skills and Abilities for Key Radiation Protection Positions at DOE Facilities (DOE 1997)

Staffing Levels

At least one professional health physicist is recommended to be on the staff of each major uranium facility as a full-time employee.

There is no rule of thumb for determining the number of RCTs needed for a given uranium facility.

The number of RCTs should be based on an analysis that provides for sufficient coverage on each shift, given the number of samples, surveys, and other work to be performed, the time of training, donning and doffing of protective clothing, shift turnover procedures, and other similar considerations. The dose rate and individual dose limits in the facility may also lead to the need for additional personnel. Consideration should be given to having sufficient numbers of personnel to respond to off-normal conditions and emergencies as well as routine work. Major maintenance, modifications, or decommissioning activities may require additional personnel.

3.2.2 ALARA Program

The policy for maintaining radiation exposures ALARA has existed in principle since the early 1940s. The evolution of ALARA into a formal program began in the early 1960s.

Although there is, and has been since the 1940s, a series of official established dose limits, they do not represent ALARA. ALARA is a continuous process of controlling and managing radiation exposure to workers, the general public, and the environment. Although ALARA is based upon protection of people and the environment, the philosophy is also grounded on sound economic and operating principles. The responsibility for maintaining radiation exposures ALARA is not a unique responsibility of management or radiological control personnel. It is a responsibility of everyone involved in managing, supervising, or performing radiation work. It is imperative to teach administrative personnel to support the principles and practice of ALARA, and to train all workers to consider ALARA as they prepare for and perform their work.

3.2.2.1 Assignment of ALARA Responsibility and Authority

Limiting radiation exposures to the lowest levels commensurate with economics and the work to be accomplished has long been a part of radiological control and radiological protection programs of DOE and its contractors. 10 CFR 835 establishes the policy of maintaining ALARA doses for workers and the public resulting from radiation from DOE operations. Plans and programs are required to be prepared and implemented, and records must be maintained to demonstrate the implementation of ALARA. DOE G 441.1-2, Occupational ALARA Program Guide (DOE 1999e), the RCS, and PNL-6577, Health Physics Manual of Good Practices for Reducing Radiation Exposure to Levels That are as Low as Reasonably Achievable (PNL 1988a), provide additional guidance.

An ALARA committee should be established at the uranium facility. The membership should include managers and workers from the line, the technical support organization, and the radiological control organization. A line manager, such as a Director of Operations, Research, or Maintenance, should serve as the committee chair. The ALARA committee should make recommendations to management to improve progress toward minimizing radiation exposure and radiological releases.

3.2.2.2 Current Status of ALARA Programs

Currently, it is common practice in DOE facilities to have a well-structured ALARA plan for the entire facility, with more detailed plans in the various buildings or functional subunits of the facility. There is ordinarily a facility coordinator who administers the overall ALARA plan and reports to top-level management of the facility. Coordinators for the various buildings or subunits of the facility receive guidance from the overall facility coordinator and report the results of their ALARA programs to that individual.

3.2.2.3 Achievement of Goals

To ensure improving radiological performance, at the beginning of each year, each facility should prepare radiological performance goals. At intervals commensurate with the radiological risk, the contractor should provide DOE with an interim status report of the goals. At the end of the calendar year, the contractor should provide DOE an Annual Goal Status Report.

Identifying specific ALARA goals in uranium facilities requires close coordination between the facility ALARA team members (operations, maintenance, and radiological control personnel) made up from a cross-section of personnel representing the various work elements of the facility. ALARA goals may be formulated as qualitative or quantitative types of goals, but must be measurable and achievable, with clearly defined endpoints.

3.2.2.4 Quality Assurance

Important aspects of any ALARA program are the measurement of beneficial effects and the determination that important factors, such as economic impacts, the time involved in accomplishing tasks, and the utilization of personnel, are being optimized. To accomplish these objectives, it is necessary to have a written plan for the ALARA program and high quality records of activities involving exposures to workers, the public, and the environment. These permit comparisons with past experiences and analysis of the recorded activities. In many cases, such studies of the recorded activities not only confirm satisfactory execution of the work, but reveal opportunities for future improvements.

One approach that works well is the inclusion of an ALARA worksheet with the RWP. Such a worksheet should be prepared by an individual with responsibilities for the work to be performed, a relatively detailed knowledge of the radiological conditions, and knowledge of what is required to accomplish the task. The worksheet should contain estimates of the time to complete the task and the expected radiation doses to be received. If any specially-engineered devices are used to control personnel exposure, they should be noted on the ALARA worksheet, with any special instructions they require. These worksheets provide valuable information for analysis of the effectiveness of the ALARA program for each job.

3.2.2.5 Technical Aspects

The technical aspects of ALARA programs include not only the standard equipment regularly used in controlling dose to workers, the public, and the environment, such as facility shielding, ventilation filters, installed and portable radiation measuring instruments, but also many special devices that may be used temporarily. Special devices can be used to provide exposure control and/or containment when it may not be practical without them. These include temporary shields, tents or greenhouses, portable fans, ductwork and filters, and special fixtures to hold highly radioactive materials requiring detailed inspection, repair, modification, or fabrication. Such devices can permit doing difficult work at low radiation doses, which might not be possible otherwise.

Some of these special devices may have general application and be kept on hand for use as needed. In some cases, devices would have to be specially fabricated for a specific task. Because this would ordinarily have a significant effect on the cost of doing that job, the economic aspects of doing or not doing the job should be carefully evaluated.

3.2.2.6 Attributes of Effective Review and Audit

Evaluation of the effectiveness of an ALARA program requires both reviews and auditing. The reviews will include detailed examination of the written ALARA program plan and the records of ALARA activities. The objectives in such reviews are to find if the written plan is being followed, and what is working or not working well. Such reviews can be performed adequately by either a knowledgeable member of the facility staff or an equally knowledgeable outsider. The written report of a review should be directed to a member of management who is responsible for implementation of the ALARA program.

Audits are best performed by an outside expert who is knowledgeable about work with uranium and its radiological characteristics so that the auditor can look for problems and make appropriate evaluations and recommendations. The auditor should not only examine the ALARA program plan and records, but should also visit the working areas and laboratories in the facility with a knowledgeable escort who can answer questions about activities and conditions in the facility.

Reviews and/or audits provide the means to evaluate the effectiveness of the ALARA program through a detailed analysis of the data. Through these analyses, specific opportunities for improvement may be identified. For example, the exposure experience of a specific group can be tracked to evaluate trends and their probable causes. An increasing exposure trend can signal degradation in the radiological control program, a need for specialized training, changes in the work force, or a change in equipment or operational procedure in the areas in which higher exposures are being experienced. Similarly, a decreasing exposure trend could mean either that the ALARA program is accomplishing its objective or that a major change in radiological work has occurred. Such trends should be examined at least quarterly to permit initiation of timely corrective actions.

When exposure trends and probable causes are clearly understood, the information should be provided to both management and staff. If an increasing exposure trend is identified, it can call attention to the problem allowing corrective action to be taken or to signal special procedures or precautions that may be needed. When the ALARA program is successful in reducing exposures, immediate feedback can verify program effectiveness and encourage further support of the program.

Reviews and/or audits and communication of the results provide the base for program upgrade. Audits and/or reviews are also an effective means to evaluate the effectiveness of a policy or procedure change and assist in determining what changes are most effective for a given set of conditions, provide a basis for future decisions as to effective means for reducing exposure, provide a basis for comparing costs with results, and provide a measure of the program's effectiveness for controlling individual and person-rem exposures as well as dose ranges and percentage of total person-rem represented by the ranges.

3.2.2.7 ALARA at Uranium Processing Facilities

The ALARA concept has wide application and serves as a basis for sound radiological control programs. The fundamental ALARA objective is to reduce radiation doses to the lowest practical levels commensurate with sound economics and operating practices. Realistic numerical goals can be set and achieved; however, compliance with numerical standards does not provide evidence that the ALARA concept is fully incorporated in the radiological control program. Rather, the success of a mature ALARA program is measured by many factors including intangibles, such as dedication to the concept of

dose control. A set of ALARA recommendations will therefore include both numerical goals and some relatively general philosophical guidance that, by itself, may not appear to assist in achieving ALARA goals.

Development and implementation of an ALARA program in many uranium facilities may be a challenging task, due primarily to the fact that penetrating radiation doses are typically low and few individuals are exposed near the regulatory limits for occupational exposures. As a result, convincing management to spend valuable funds to further reduce radiation exposures can be a problem. The ALARA program must have the support and active participation of all levels of management. It must be understood by the worker in the field and receive his or her continued support and attention.

Detailed guidance on developing and implementing an effective ALARA Program is provided in DOE G 441.1-2.

3.2.3 External Dosimetry Program

The details of the external dosimetry program are discussed in Chapter 6 of this Technical Standard and in DOE G 441.1-4, External Dosimetry Program Guide (DOE 1999f).

3.2.4 Internal Dosimetry Program

The details of the internal dosimetry program are discussed in Chapter 5 of this Technical Standard and in DOE G 441.1-3, Internal Dosimetry Program Guide (DOE 1999g).

3.2.5 Area Monitoring and Control

The details of the area monitoring program are discussed in Chapters 4 and 5 of this Technical Standard and in DOE G 441.1-3, DOE G 441.1-4, and DOE G 441.1-8, Air Monitoring Guide (DOE 1999h).

3.2.5.1 Radiological Surveys and Data Trending

Sections 835.401 - 835.403 of 10 CFR 835 establish requirements for radiological monitoring of areas and individuals. A program of routine, scheduled surveys should be established and followed, including surveys in areas that are not ordinarily expected to be affected by radiological hazards. The program should define minimum requirements, survey type, and frequency.

Surveys should be performed at frequencies adequate to identify changes in posting required or an activity buildup and to ensure current radiological controls are appropriate. The surveys specified by this section should be considered minimum requirements; additional surveys should be conducted, recorded, and reviewed as necessary to ensure adequate personnel protection.

Surveys should be performed to identify radiological area boundaries and the conditions within those boundaries, the appropriate posting of sources or areas, and the location and extent of localized radiological hazards. They should be performed and documented prior to the start of radiological work, during general work activities at times when changes in radiological conditions may occur, and following work to determine that final radiological conditions are acceptable and documented. A sufficient number of points should be surveyed to adequately assess the radiological status of the area being surveyed.

Routine radiological surveys should be regularly conducted, recorded, and reviewed for all areas where personnel could be exposed to radiation or radioactive material throughout the site. Surveys should be performed at frequencies adequate to ensure protection of personnel. The following surveys should be considered the minimum. Additional surveys should be conducted, recorded, and reviewed as necessary to ensure personnel exposures are maintained ALARA. General radiation surveys should be performed to:

a. identify and verify the boundaries of areas which must be radiologically controlled,

b. verify that radiation levels in uncontrolled areas remain less than specified limits,

c. determine the appropriate posting of localized higher radiation levels, beams, or hot spots,

d. ensure radiological conditions are acceptable and documented prior to, during, and at the completion of work that may cause changes in radiation levels to occur, and

e. satisfy required predetermined procedure hold-points in work areas and adjacent areas, whenever operations are performed that may cause significant increases in radiation levels.

The survey may be required as part of a radiological inspection step required by the work procedure. This includes areas above and below the work area as appropriate during special processing operations or cell decontamination, movement of permanent or temporary shielding, radioactive waste processing, and relocation of highly radioactive materials.

Routine external radiation level surveys should be performed in the workplace at a frequency commensurate with the radiation hazard, to detect trends related to equipment, systems, environment, and work habits. Non-routine surveys of external radiation levels in the workplace should be performed:

a. before initial use of a new installation, system, or equipment, or as soon as possible after a radiation source is brought into the area,

b. whenever changes in procedures, equipment, or sources have occurred that may cause changes in the external radiation levels,

c. after modification to a shield or changes in shield materials,

d. as the basis for trend evaluation of external radiation level conditions,

e. when a radiological accident has occurred or is suspected, or

f. when requested by the personnel performing the activity.

A sufficient number of points should be surveyed to adequately assess the radiological status of the area. Regular predetermined points may be used, but additional spot monitoring should be done to ensure all changes in dose rates are identified, recorded, and reviewed. All records of surveys should clearly identify, as a minimum:

a. name, signature, and employee number of the surveyor,

b. survey instrument(s) model number, serial number, and calibration date,

c. type(s) of radiation being monitored (e.g., neutron, gamma, etc.),

d. dose rates,

e. estimated doses to surveyors (from direct-reading dosimeters, if applicable),

f. date and time the survey was performed, and

g. locations where radioactive material is located temporarily (or is being temporarily stored) or where equipment that generates ionizing radiation is being operated.

Records of the results of radiation surveys should be retained in accordance with facility policy.

Survey data should be reviewed by the facility radiological control supervisor. Significant findings should be presented to the facility manager in a timely manner. Radiological control personnel should summarize survey data in each building or area at least once a month. Significant changes or trends in area dose rates and/or radiological contamination should be noted and corrective actions assigned. The survey summary should be presented to the facility management monthly.

Survey results and data summaries should be made available to the ALARA committee periodically and should be used to:

a. provide a basis for evaluating potential worker exposure on a job and in ALARA preplanning,

b. provide a baseline for trend analysis, investigation, and correction of unusual conditions,

c. track the status of jobs (including identification of good practices) and detect departures from good operating procedures and/or the failure of radiation controls, and

d. identify the origin of radiation exposures in the plant by location, system, or component.

Radiological control personnel should post survey maps at the entrance to all radiological areas so personnel can be aware of radiological conditions within the area.

A survey data trending program should be conducted to indicate the continuing effectiveness of existing control, to warn of deterioration of control equipment or effectiveness of operating procedures, to show long-term variations in radiation levels, and to identify and correct improper radiation work practices. See NUREG-0761, Radiation Protection Plans for Nuclear Power Reactor Licensees (NRC 1981), sections 07.B(I)(C), 09.B(4), and 09.C(I)(C).

Radiological control personnel should perform trend analyses on all permanent radiological areas. At a minimum, one complete survey record should be evaluated and included in the trend analysis program for each survey required to be performed by the facility routine control program. See NUREG-0761 (NRC 1981), 07.B(I)(C), 09.B(4), and 09.C(I)(C).

Radiological control personnel should use the facility reporting system to identify discrepancies and abnormal trends and should summarize the data review results in their monthly reports to the radiological control manager. Survey data trends should be investigated when either an upward trend occurs, causing a significant increase (10% or more), or an abrupt change in conditions occurs that cannot be directly correlated to normal activities.

3.2.5.2 Instrumentation Considerations

Instrumentation performance criteria are necessary for portable, fixed, and emergency monitoring instrumentation. There are also requirements for instrument calibration and testing.

General Performance Criteria for Instruments

Programs for in-plant monitoring of uranium consist mainly of airborne and surface contamination surveys and dose rate surveys. The general and specific performance criteria for the instrumentation needed to conduct these programs are described in ANSI N317-1991, Performance Criteria for Instrumentation Used for In-Plant Plutonium Monitoring (ANSI, 1980). Performance specifications are also given in ANSI N323-1993, Radiation Protection Instrumentation Test and Calibration (ANSI 1993), ANSI N42.17A, Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions (ANSI 1988a), and ANSI N42.17C-1989, Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Extreme Environmental Conditions (ANSI 1987a) for portable radiological control instrumentation and IEC Publication 325, Alpha, Beta, and Alpha-Beta Contamination Meters and Monitors (IEC 1981) for alpha and beta contamination meters and monitors. Criteria for air monitoring instrumentation are provided in ANSI/HPS N13.1-1999, Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities (ANSI/HPS 1999b), IEC Publication 761-2, Equipment for Continuously Monitoring Radioactivity in Gaseous Effluents (IEC 1983), and ANSI N42.17B-1989, Performance Specifications for Health Physics Instrumentation - Occupational Airborne Radioactivity Monitoring Instrumentation (ANSI 1987b). Criticality alarm systems are discussed in ANSI/ANS 8.3-1986, Criticality Accident Alarm System (ANSI 1986a). The criteria discussed in the following sections are specified in these standards as referenced.

Portable Monitoring Instruments

ANSI N317 discusses several criteria related to the performance of portable monitoring instruments:

a. The overall accuracy should be within ±20%, and the precision should be within ±10% at the 95% confidence level.

b. The response time (i.e., the time for the instrument reading to go from zero to 90% of full scale) should be <10 seconds on the most sensitive scale and <2 seconds at readings of 100 mrem/h, 100 mR/h, and 500 dpm or greater. (This criterion is unrealistic with current neutron instrument capabilities. Response time is typically 30 to 60 seconds.)

c. The instrument should be able to maintain accuracy and precision for a minimum of 24 hours of continuous operation.

d. The instrument should have a minimum battery lifetime of 200 hours of continuous operation. ANSI N42.17A specifications differ slightly.

e. The response of the instrument should not change by more than ±15% from a reference value taken at 20°C over the anticipated temperature range for operation.

f. The instrument system should function within specifications over all anticipated combinations of temperature and humidity (e.g., 15° to 65°C, 40% to 95% relative humidity).

ANSI N317 states the minimum detection capability for alpha monitoring instruments ideally should be 220 dpm/100 cm^2 of surface area and should not be more than 500 dpm/100 cm^2. This requirement should be met in the presence of a radiation field of 0.10 rem/h of neutrons in the energy range of thermal to 10 MeV, and/or in the presence of 0.10 rem/h of photons in the energy range of 0.010 to 1.25 MeV. The operating range should be from 0 dpm to at least 100,000 dpm/100 cm^2 of surface area. The response of the instrument to beta-interfering radiation is an important specification that should be stated by the manufacturer.

Photon monitoring instruments should meet the accuracy requirements stated in ANSI N317 over the energy range of 0.01 to 1.25 MeV. The angular response of this type of instrument should be within ±15% over a 2 pi steradian frontal direction using at least two photon sources with energies ranging from 0.06 to 1.25 MeV. Experience has shown this response specification is not met by most instruments at lower energies due to attenuation of the photon. The energy dependence should be within ±15% over the range of 0.01 to 1.25 MeV and the operating range should be from 0.5 mR/h to at least 5000 mR/h. Experience has shown that ±20% over 0.01 to 1.25 MeV is more realistic. This specification applies to a specific window selection (e.g., below 0.05 MeV, the electron equilibrium cap or beta shield must be removed).

ANSI N42.17A has a broader scope than ANSI N317, but the criteria in it apply to portable survey instruments. Additional criteria include geotropism (maximum change of 6% from reference reading for all orientations), temperature shock, mechanical shock, vibration, and ambient pressure (maximum change of 15% from reference reading for the latter four criteria). Some differences exist between ANSI N42.17A and ANSI N317. In most cases, the criteria for ANSI N42.17A are more applicable because these criteria are based on substantial testing, which was sponsored by DOE. In ANSI N42.17A, precision is tied into a measurement level; for example, it quotes a precision of 15% at <500 cpm and 10% at >500 cpm. Also, with the advent of liquid crystal displays and other digital readouts, "response time" is defined as the time it takes for the reading to move from 10% to 90% of the equilibrium or steady-state reading. Another significant difference in the standard is the battery lifetime specification is 100 hours instead of the 200 hours mentioned in ANSI N317.

For direct alpha contamination surveys, the use of audible signals (headphones or speaker) greatly facilitates the detection of "hot spots." IEC Publication 325 provides additional guidance on the uniformity of probe response for alpha and beta contamination meters. Surface sensitivity measurements are also discussed in this standard.

Performance Criteria for Fixed Monitoring Instruments

Airborne contamination monitors, surface contamination monitors, and photon area monitors, and emergency instrumentation are fixed monitoring instruments subject to the following standard performance criteria.

Airborne Contamination Monitors. Airborne contamination monitors, normally CAMS should meet the following criteria according to ANSI N317. The primary purpose of any CAM is to detect the presence of airborne radioactivity and activate an alarm to warn personnel in the area so actions can be taken to minimize personnel exposures. The goal for any CAM should be to perform this function as quickly as possible and at the lowest detectable level of radioactive airborne concentration. The quantity of airborne radioactivity that will result in an alarm within a given time interval is defined in units of DAC-h for a particular radionuclide and is a function of the nuclide's airborne concentration in DACs, the sampling rate, the lower limit of detection of the instrument, and the time needed for the alarm to occur. Mishima et al. provides guidance on each of these functions.

ANSI N42.17B provides additional performance criteria for air monitors used to detect uranium. This standard provides specifications for general criteria (sampler design, units of readout, alarm threshold, etc.), electronic criteria (alarms, stability, response time, coefficient of variation, and line noise susceptibility), radiation response, interfering responses (radiofrequency, microwave, electrostatic, and magnetic fields), environmental criteria (temperature, humidity, and pressure), and air-circuit criteria. More detailed specifications are provided in ANSI N42.17B than in ANSI N317; however, the environmental criteria and the limits of variation are not as restrictive as those in ANSI N317. With respect to accuracy, ANSI N317 requires less than ±20%, and ANSI N42.17B requires 40% at the 95% confidence level. For the environmental criteria, ANSI N317 requires that the readings change less than 5% under ambient conditions, while ANSI N42.17B gives a 15% limit of variation. As discussed previously, criteria from ANSI N42.17B are more applicable because they are supported by instrument testing.

ANSI N13.1 provides detailed guidance on sampling methods from stacks and ducts. One criterion that relates to CAMs is that air sample lines between air inlet and filter media should be eliminated where possible; where not possible, they should be designed to meet the sampling criteria contained in the standard (e.g., short lines, proper sampling rate, smooth bends). The use of Tygon tubing as sample lines should be minimized or eliminated. Air in-leakage from surrounding areas can be a problem when using sampling lines. Testing for air in-leakage should be performed at least annually or when seals or "O" rings are replaced.

Surface Contamination Monitors. Surface contamination monitors include hand and/or shoe counters and instruments (or probes) with sufficient flexibility to survey pieces of equipment, including exterior clothing. ANSI N317 states these instruments should have an audible alarm, a frequency that is proportional to the count rate, or a pre-selectable trip setting, and upon reaching that level, should activate an audible or visible alarm or both. These instruments should be calibrated according to the requirements in ANSI N323 and be equipped with a check source. Fixed instruments should be powered by alternating current (AC) and provided with an emergency power source.

Performance Criteria for Emergency Instrumentation

Meeting the criteria for criticality accident alarm systems, fixed nuclear accident dosimeters, and other emergency instrumentation is essential.

Criticality Accident Alarm Systems (CAAS). See section 7.0 for discussion of nuclear criticality safety, including CAAS.

Fixed Nuclear Accident Dosimeters. All DOE facilities that have sufficient quantities and kinds of fissile material to potentially constitute a critical mass should provide nuclear accident dosimetry. Requirements for fixed nuclear accident dosimeters are found in 10 CFR 835.1304 and DOE Order 420.1A, Facility Safety (DOE 2002).

Effluent Monitors. Facilities should evaluate potential emissions in accordance with ANSI/HPS N13.1 to determine the need for stack sampling and/or monitoring.

Other Emergency Instrumentation. Other emergency instrumentation should provide ranges for all radiation dose rates and contamination levels potentially encountered at the time of an accident. Normally, dose rate capabilities from a few millirem per hour to a few hundred rem per hour should be required while capability requirements for the contamination level may range upward from 200 dpm/100 cm^2 for alpha contaminants and 100 dpm/100 cm^2 for beta-gamma emitters. Performance specifications for emergency radiological monitoring instrumentation can be found in ANSI N320-1979, Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation (ANSI 1975) and BNWL-1742, Technological Consideration in Emergency Instrumentation Preparedness. Phase II-B - Emergency Radiological and Meteorological Instrumentation for Mixed Oxide Fuel Fabrication Facilities (Andersen et al. 1974).

Instrument Calibrations and Testing

Radiation doses and energies in the work areas should be well characterized. Calibration of instruments should be conducted where possible under conditions and with radiation energies similar to those encountered at the work stations. Knowledge of the work area radiation spectra and instrument energy response should permit the application of correction factors when it is not possible to calibrate with a source that has the same energy spectrum. All calibration sources should be traceable to recognized national standards. When the work areas have been well characterized, the calibration facility used by the uranium facility should be set up to represent as closely as possible the work area's radiation fields.

DOE G 441.1-7, Portable Monitoring Instrument Calibration Guide (DOE 1999i) and ANSI N323 provide guidance on radiation monitoring instrument calibration. The reproducibility of the instrument readings should be known prior to making calibration adjustments. This is particularly important if the instrument has failed to pass a periodic performance test (i.e., the instrument response varies by more than ±20% from a set of reference readings using a check source) or if the instrument has been repaired. The effect of energy dependence, temperature, humidity, ambient pressure, and source-to detector geometry should be known when performing the primary calibration. Primary calibration should be performed at least annually.

Standards referenced in Section 3.5.2 discuss specific performance testing of radiation detection instruments. Testing procedures in these standards should be used for periodic requalification of instruments or detailed testing of instruments.

The calibration of photon monitoring instruments over the energy range from a few keV to 300 keV is best accomplished with an x-ray machine and appropriate filters that provide known x-ray spectra from a few kiloelectron volts to approximately 300 keV. Radionuclide sources should be used for higher

energies. Most ion chambers used to measure photon radiations have a relatively flat energy response above 80 to 100 keV; ^{137}Cs or ^{60}Co are typically used to calibrate these instruments. These sources also should be used to calibrate Geiger-Mueller (GM) type detectors. It should be noted that some GM detectors (e.g., those with no energy compensation) can show a large energy dependence, especially below approximately 200 keV.

Whenever possible, beta detectors should be calibrated to the beta energies of interest in the workplace. A natural or depleted uranium slab source can be used for calibration of beta detectors when beta radiations in the workplace have energies similar to the uranium. International Organization for Standardization beta sources should be used for all other purposes: the energy dependence of beta detectors can be tested using the calibration sources listed in the ISO Publication 1980 (ISO, 1984); these include ^{90}Sr, ^{90}Y, ^{204}Tl, and ^{147}Pm.

The calibration and testing of crucial monitoring systems are extremely important to the overall radiation protection program, but have often been neglected. Effluent monitoring and sampling systems (when present) and remote area monitoring systems should be given several tests. The radiological, environmental, and mechanical characteristics of the instrumentation portion of the system should be fully evaluated prior to its first use to ensure its compatibility with performance requirements and facility operating conditions. The effluent sampling losses from the sample probe to the collector/detector should be determined. This test should be repeated at least annually and when a significant change in the sampling equipment is made. The sample probe should be examined at least once a year to verify its design or performance has not been changed by corrosion. The recorder of the sample flow rate should be calibrated when it is installed and annually thereafter. The operability of the overall system should be completely tested once, with repeat tests only after modification, repair, or maintenance. Operability checks should be scheduled at least monthly and calibration performed at least annually.

The operation of criticality or other radiation alarm signal systems should be checked periodically to ensure the alarms are audible at all potentially occupied locations (ANSI 1986a). To prevent any desensitizing of staff, the staff should be aware the tests will be performed, and where possible, tests should be scheduled during off-shift hours. Building systems should be tested semiannually and the area-wide system should be tested at least annually. Any portion of the detector/alarm system affected by the test should be reconfirmed for operability after the test is completed (e.g., if a detector is disconnected and a signal is injected at that point, the detector should be tested immediately after it has been reconnected).

3.2.6 Radiological Controls

3.2.6.1 Work Authorizations

Written authorizations shall be required to control entry into and work within radiological areas and shall specify radiation protection measures commensurate with the existing and potential hazards (10 CFR 835.501(d)). ALARA considerations need to be included in the work authorization. One approach that works well is the inclusion of an ALARA worksheet with the radiological work permit (RWP). Although the written work authorizations may take any appropriate form (e.g., written procedures, policy statements, technical work documents, etc.), RWPs are most often used. RWPs should be used for entry into high and very high radiation areas, high contamination areas, and airborne radioactivity areas. RWPs should also be used to control entry into radiation and contamination areas and for handling materials with removable contamination. The RWPs should be initiated by the work group responsible

for the activity. All RWPs should be reviewed and approved by the radiological control staff and cognizant line management. The RCS provides detailed guidance for RWPs.

Radiological workers should read and understand the applicable RWP before entering the affected area. Copies of the RWP should be located at the access point to the applicable area. Workers should acknowledge by signature or through electronic means that they have read, understood, and will comply with the RWP before they initially enter the area and after changes. Out-of-date RWPs should be removed.

3.2.6.2 Facility Posting and Labeling

Radiological areas, controlled areas, and radioactive material areas shall be posted, unless the conditions constituting the authorized exceptions specified in 10 CFR 835 exist (10 CFR 835.601-835.606). DOE Guide G-441.1-10, Posting and Labeling for Radiological Control Guide (DOE 1999j) and the RCS provide appropriate guidance. The technical criteria for defining the required areas should be established, documented, and consistently applied. The radiological control staff should establish and document the conditions that require areas to be barricaded and marked to prevent personnel from inadvertently entering them and to be physically locked to preclude unauthorized personnel from entering them.

Entrance to areas where radioactive materials are used or stored should be restricted, based upon established criteria.

The radiological control staff should post current surveys at the access control point for use in pre-job planning. Additional precautions, such as protective clothing, dosimetry, and respiratory protection requirements should also be posted.

3.2.6.3 Unposted Areas

Certain areas of facilities that handle radioactive materials should be maintained free of detectable radioactive contamination. These areas should also be maintained at ambient radiation levels equivalent to the environmental background of the facility. Parts of the facility that should meet these requirements include lunchrooms, offices, restrooms, janitor rooms, corridors outside operational areas, foyers, and outside areas surrounding the facility, including building roofs.

To determine that these areas meet the requirements of non-radioactive cleanliness, they should be surveyed with count-rate instruments sensitive to the radioactive isotopes of interest. These clean areas should be maintained below the detection levels cited in 10 CFR 835.

3.2.6.4 Visits by Regulatory Personnel

Periodically, personnel from DOE and other Federal and state agencies visit radiological facilities for audit purposes or to discuss regulatory changes. In most cases, they will look at records of the radiation protection program and, in some cases, will also enter posted areas of the facility. These regulatory personnel should have ready access to the facility; provided that applicable training, dosimetry, and other requirements are met. They should have complete access to facility personnel knowledgeable in the subjects they wish to discuss. New commitments requested should be referred to the appropriate facility and DOE management.

3.2.7 Emergency Exposure Situations

Requirements and guidance for emergency exposure situations are discussed in detail in Chapter 9 of this TS.

3.2.8 Nuclear Accident Dosimetry

Nuclear accident dosimetry is discussed in detail in Chapter 6 of this TS.

3.2.9 Records

The systematic generation and retention of records relating to the occupational radiological control program are essential to describe the occupational radiation dose received by individuals and the conditions under which the exposures occurred. Such records have potential value for medical, epidemiological, and legal purposes.

10 CFR 835 establishes radiation protection program records requirements. Detailed guidance is provided in DOE G 441.1-11, Occupational Radiation Protection Record-keeping and Reporting Guide (DOE 1999k) and the RCS provide guidance for radiation protection program records. The following types of records should be maintained:

a. Individual radiological exposure records

   1. internal doses,

   2. external doses (whole body, skin of the whole body, extremities, and lens of the eye),

   3. total effective dose equivalent (summation of internal and external doses),

   4. lifetime and cumulative total effective dose equivalent,

   5. non-uniform exposure to the skin,

   6. supportive data for determining individual doses, and

   8. individual medical records.

b. Radiological status of work area records

   1. radiation safety analysis and evaluation reports,

   2. radiation work procedures and permits (RWPs),

   3. radiation and contamination surveys,

   4. records of releases of potentially contaminated materials and equipment from radiological areas,

5. airborne radioactivity monitoring records, and

6. area monitoring instrumentation records.

c. Records of monitoring methods

1. radiation protection policies and procedures,

2. evaluation of exposure data,

3. functional capabilities of dosimeters and instruments,

4. calibration and maintenance records,

5. audits and programmatic reviews,

6. changes in procedures, techniques, and equipment, and

7. individual radiation safety training.

Most of the required radiological records have established retention periods. The retention periods are discussed in DOE Guide 1324.5B, Records Management (DOE 1996b). Individual records may be covered by the Privacy Act; the DOE has codified the Privacy Act in 10 CFR 1008, Records Maintained on Individuals (Privacy Act) (DOE 1994b).

3.2.10 Radiation Safety Training

A thorough radiation safety training program should be established at uranium facilities. Training programs should ensure that personnel have the training to work safely in and around radiological areas and to maintain their individual radiation exposure and the radiation exposures of others ALARA. Separate training programs should be established for general employees and radiological workers. DOE's core training materials should form the basis for the training programs, and should be augmented with site-specific information. The training of all staff members should be carefully documented. DOE G 441.1-1A, DOE G 441.1-12, Radiation Safety Training Guide (DOE 1999l) and the RCS provide guidance on information to be presented during the training programs.

DOE requires biennial radiation safety training for general employees and radiological workers. In the alternate year when retraining is not performed, refresher training should be provided. Individuals who work with uranium should have special uranium facilities training in addition to or as part of the appropriate level of Radiological Worker Training.

3.2.10.1 Radiological Worker Training

Before working in uranium operations, all radiological workers should be trained and qualified. A thorough radiation protection training program should be established at uranium facilities. Before beginning uranium training, each uranium worker should receive General Employee Radiological Training (DOE 1998e) or either Rad Worker I or Rad Worker II Training (DOE 1998d). In addition, DOE-HDBK-1113-98 Radiological Safety Training for Uranium Facilities (DOE 1998c) provides guidance on providing radiation safety training to workers at uranium facilities.

The level of radiation worker training should be determined in accordance with the Table 3-1 of the RCS. All training should be consistent with the guidance provided in DOE G 441.1-12. All training dispositions and records should be documented in accordance with 10 CFR 835.704.

3.2.10.2 Training for Other Facility Personnel

Non-radiological workers in a uranium facility should be given a general orientation on the radiation safety concerns for working with uranium, the general protective measures used for work with uranium, and the engineered safety features of the facility.

3.2.10.3 Members of the Public

Members of the public with a demonstrated need to enter the following areas may be allowed access if such access is controlled with a combination of training and the use of escorts trained for the specific area:

a. Radiological Buffer Areas,

b. Radiation and High Radiation Areas,

c. Contamination Areas, and

d. Radioactive Material Areas.

Guidance for training of members of the public is provided in DOE G 441.1-12 and the RCS. Individuals under 18 years of age should not be permitted to enter radiation areas or contamination areas without the approval of the radiological control manager. Area entry requirements and access restrictions for members of the public should be established in facility procedures. Members of the public should be prevented from entering very high radiation, high contamination, and airborne radioactivity areas.

All facility personnel serving as a qualified escort should ensure that each visitor under his/her cognizance completes a facility radiological visitor form. The qualified escort should also sign the visitor form and complete it as appropriate.

Facility-sponsored visitors should provide the following before entering radiological areas, unless these records have already been entered into the facility entry control system:

1. evidence of completing required training, as applicable

2. visitor radiation exposure disclosure

3. a medical disclosure form or the results of a medical evaluation.

The host facility manager should forward the visitor radiation exposure and medical disclosure forms to Dosimetry.

The use of offsite mask fit certification is authorized under the following conditions:

1. A mask fit has been completed within the previous year.

2. The individual presenting the mask fit certification card has not changed physical appearance in a way that would affect the seal of the mask to the face. For example, this could be determined by a combination of: review of photograph on mask fit certification card (if available), examination of facial hair on areas which could affect mask seal, and discussion with wearer of any physical changes which could affect mask seal.

3. The facility has the masks available that the individual is certified to wear.

If there are members of the public who live or work near a uranium facility, a plan for orientation of members of the public should be developed to inform them of facility activities. Such a plan should include information on the concerns that require protection of people from potential injuries by uranium, the general protective measures used at the facility to confine it and keep it out of the public domain, and solicitation of information on the concerns of members of the local public about uranium. To the extent possible, efforts should be made to allay those concerns. The information in the public education plan should also be provided to local news media.

3.3 RELATED PROGRAMS

3.3.1 Onsite Packaging and Transportation

The hazardous materials organization conducts onsite radioactive shipments with the assistance of radiological control. This program requires the hazardous materials organization representatives to review onsite radioactive shipping records, document the errors or omissions observed, and evaluate trends and revise training as needed. Serious deficiencies are to be documented and the reports should be submitted in accordance with DOE O 460.1B, Packaging and Transportation Safety (DOE 2003b).

The packaging organization is responsible for coordinating onsite package design and preparation of safety analysis documentation. The following sections describe typical process, review, and approval requirements for onsite safety analysis documentation.

3.3.1.1 Initiation

New safety analysis documentation or reviews/changes to existing documentation can be requested by a user organization based on programmatic or operational requirements. The request is submitted in writing to the packaging organization and includes proper justification and support documentation. The packaging organization makes routine revisions as necessary to reflect policy and regulation changes.

3.3.1.2 Preparation

The packaging organization coordinates the analysis, prepares safety analysis documentation, and guides the documentation through the review and approval process, including the resolution of review comments and the obtaining of required approval.

3.3.1.3 Control

Safety analysis documentation is prepared and maintained according to facility policy. The document control system provides an accessible, auditable, and retrievable method for maintaining and changing safety analytic documentation.

3.3.1.4 Review and Approval Cycle

Safety analysis documentation is reviewed, approved, and changed according to facility policy. Additional reviews and approvals include the following people and organizations:

a. user,

b. cognizant engineer,

c. packaging organization,

d. quality assurance,

e. responsible environmental assurance organization, onsite only,

f. packaging, shipping, and waste safety assurance organization,

g. criticality engineering analysis, if criticality analysis is required,

h. packaging and shipping approval authority, and

i. DOE field office, if the package is to be used for HRCQ inter-area shipments.

3.3.1.5 Approval for Editorial Changes

Inconsequential editorial changes to a safety analysis document may be approved at the operating level.

3.3.1.6 Utilization

Once a safety analysis document is approved, copies are sent to the affected organizations, including operations and applicable facility engineering, to incorporate the administrative controls from the safety analysis document into the affected operating documents. User organizations must obtain the packaging organization review of all operating procedures that incorporate instructions or administrative controls found in COCS, SARPS, SEPS, DAPS, DOT exemptions, and Federal and state packaging requirements to ensure that they are properly incorporated.

Onsite packages currently approved for onsite use should be cataloged and described in a hazardous materials packaging directory maintained by the packaging organization. New packages are added to the directory as they are developed and approved.

3.3.2 Conduct of Operations

The organization and administration of operations should ensure a high level of performance in DOE facility operations is achieved through effective implementation and control activities. Administration of operations activities should recognize that protection of the environment, maintaining a high-quality safety program and productivity are compatible goals. DOE policies and standards describe the standards of excellence under which the facility is expected to operate. Clear lines of responsibility for normal and emergency conditions must be established. Effective implementation and control of operating activities are achieved primarily by having readily accessible written standards for operations, periodical monitoring and assessment of performance, and personnel accountability for performance. For a more detailed discussion, see DOE Order 5480.19, Ch.2, Conduct of Operations Requirements for DOE Facilities (DOE 2001a).

A high level of performance in DOE operations is accomplished by management establishing high operating standards and then by communicating the operating standards to workers by providing sufficient resources to the operations department, ensuring personnel are well trained by closely monitoring performance in operations, and holding workers and their supervisors accountable for their performance in conducting activities.

Senior management establishes operating standards, considering input from workers when appropriate. Working-level personnel will more strongly support the standards when they have had appropriate input into their development. Standards should define operating objectives, establish expected performance levels, and clearly define responsibilities in plant operations. Standards for operating activities should be integrated into operations department procedures and programs. Operating standards should also be communicated to workers by training them in operating practices and by having supervisors monitor and guide work involving facility operations. Sufficient staff, facilities, equipment, and funding should be allocated to permit the operations department to effectively perform its functions. Performance in operations should be closely monitored by facility management, preferably using operating reports and goals, so the performance of the operations department can be effectively measured. Operations personnel should be held accountable for their performance through supervisor counseling, performance appraisals, and, when necessary, disciplinary measures. Remedial training should be provided when appropriate.

The radiological control organization, as a support element, must ensure that all aspects of radiation safety are considered in the establishment of operations standards and policy. A well-instituted cooperative relationship between operations and radiological control is paramount to the health and safety of workers and the public and to protection of the environment.

A uranium facility should have a written policy on radiation protection, including an ALARA policy. All radiation protection procedures and controls should have recognizable or formal technical bases for limits, methods, and personnel protection standards. Procedures should be adequately documented, updated periodically, and maintained in a centralized historical file. A control system should be established to assure all copies are accounted for and all new procedures are included in the historical files. A designated period of time for holding the historical files should be established. ANSI N13.6-1989 (ANSI, 1966) provides guidance on historical files. In addition, radiological control procedures should have a documented approval system and established intervals for review and/or revision. A tracking system should be developed to ensure the required reviews and revisions occur.

3.3.2.1 Radiological Work Procedures

Radiological work procedures, including RWPs, survey procedures, ALARA reviews, sample counting, and other task procedures, fall within the requirements for conduct of operations. All sections of DOE Order 5480.19 apply. The guidance and requirements of Section XVI, "Operations Procedures," is especially pertinent to radiological work procedures. Procedures are a key factor affecting radiation protection performance. Appropriate attention should be given to writing, reviewing, approving, and monitoring implementation of radiation protection procedures. There should be documented qualification and training requirements for those who prepare and approve procedures. A formal approval process should be established. Procedure changes and revisions should be subject to the same review and approval process as the initial procedure.

Personnel should be trained in the use of the procedures they will be expected to perform. For RWPs, workers are required to read the RWP and verify by signature they have read it, understand its contents, and will comply with its requirements in the conduct of the work. Procedures should be available for personnel use. The RWPs should be posted at the entrance to the work location. There should be a system in place to assure posted copies of all work procedures, including RWPs, are current.

3.3.2.2 Posting and Labeling

The requirements for area posting and radioactive material labeling are established in 10 CFR 835, Subpart G. Guidance on implementing the regulatory requirements can be found in DOE G 441.1-10 and the RCS. Conformance to conduct of operations requirements should assure a reasonable degree of uniformity in the posting and the signs used, as well as verifying that operator aids and other posted information do not interfere with necessary radiological posting. Radiological postings should be reviewed in the same manner as the posting of operating aids, in conformance with DOE Order 5480.19.

3.3.2.3 Instrument Calibration

The status of installed and portable radiological instruments should be well known and appropriate to the use.

"Ownership" of installed monitoring instruments should be well known and the responsibility and authority for calibration, repair, and notification clearly established. Because such information is often used by more than one group, formal notification procedures should be established to cover those times when the instrument is out of service or beyond the required calibration schedule. Configuration control and quality assurance requirements for installed systems should be established commensurate with their safety significance.

For portable instruments, conduct of operations requirements are normally built into the routine calibration and survey program. Functional checks are routinely made to verify calibration, instruments are checked to assure they are within the calibration period, and survey procedures require identification of the instruments used so if a problem is later found, measurements can be repeated.

3.3.2.4 Audits

Conduct of operations does not, in itself, address requirements for auditing. The guidance does state that inspections, audits, reviews, investigations, and self-assessments are part of the checks and balances needed in an operating program. Auditing is one of the many tools line management has at its

disposal to identify problems. Each one of the 18 topics addressed in DOE Order 5480.19 should be subject to both internal self-assessment and external auditing to assure effective implementation of requirements. Any deficiencies identified should be documented and corrective actions aggressively pursued and tracked to completion. The self-assessment and audit process should include conducting trend analyses and root cause evaluations of deficiencies and communication of results throughout the organization.

3.3.2.5 Decommissioning of Weapons and Weapon Facilities

Decommissioning of nuclear weapons and nuclear facilities is subject to the same conduct of operations requirements as operating facilities. In general, some components, once they are separated, can be downgraded in safety significance. Also, facilities undergoing decommissioning will have fewer safety systems.

During decommissioning, status control and shift turnover are extremely important considerations. Posting of radiological areas and labeling of radioactive materials are also an increasing challenge because of the rapidly changing radiological status. In extreme cases, it may be desirable to have workers review or sign the RWP each day to ensure they are aware of the status.

3.3.3 Integrated Safety Management

The radiological control program should be developed and implemented in a manner that is consistent with the DOE approved Radiation Protection Program required by 10 CFR 835.101 and the requirements of DOE Policy P 450.4, Safety Management System Policy (DOE 1996a), and its associated guidance documents. The RPP should describe a system of radiological controls that can be implemented on a site-wide basis and tailored to meet facility-and hazard-specific needs. The program should provide for increasing worker involvement in identification and implementation of appropriate controls. Like the ALARA process, an effective integrated safety management system emphasizes the development and implementation of controls that are commensurate with the hazards associated with any specified activity. Under ISM, both DOE and DOE-contractor line managers are charged with responsibility for integrating safety measures into all facets of work planning and execution. Line managers at uranium facilities should use the RCS and this TS as a guide to integrating radiological control measures into work planning and execution.

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4.0 CONTAMINATION CONTROL

Contamination control is an important part of the overall radiological control program. There are three main aspects to this: 1) control of the release of contamination into the work-place environment; 2) control of personnel exposure to the contamination that does get into the work place; and 3) protection of personnel from intake of contaminants. Effective control of personnel exposure to uranium and its decay products is accomplished mainly by controlling the potential for inhalation and ingestion of radioactive materials. Monitoring provides an indication of the effectiveness of physical design features and administrative controls in controlling exposure to radioactive material.

This chapter addresses the basic features of an effective contamination control program and the technical considerations of implementing the program. A release of radioactive material from containment typically results in surface contamination and airborne dispersion. Airborne contaminants are continuously cleared from the work place by ventilation. Strategic air sampling detects the release of an airborne contaminant and provides the means for control, minimization of personnel exposure, and evaluation of inhalation exposure. Considerations for design of an air monitoring program are followed in this chapter by a section on surface contamination control. Finally, protection of personnel from contaminant intake is accomplished with protective clothing and respiratory protection.

4.1 AIR MONITORING

The most common route of uranium intake for workers is by inhalation. Airborne particles deposit throughout the respiratory tract. Some of the deposited particles are swallowed, contributing to ingestion, requiring that both inhalation and ingestion be considered with an exposure to airborne material. The particle size distribution that determines deposition in the respiratory tract is affected by the mechanism of dispersion and the nature of the source material. Characterization of inhalation exposure should make use of all available information about the chemical and physical form of airborne material. This information, along with spatial and temporal distribution, provides the basis to minimize personnel exposure for air contamination control.

4.1.1 Internal Versus External Dose Philosophy

The widespread application of methods to contain uranium in DOE facilities has resulted in a history of relatively minor internal exposures. The methods used to control internal dose have been developed for a variety of reasons:

a. The assessment of internal dose requiring bioassay is difficult, imprecise, time-consuming, and offensive to personnel as compared to external dosimetry. For example, an accidental internal uptake may require the subject to submit dozens of biological samples over the span of many months, as well as requiring extensive analytical support for measurement of sample content, considerable time of trained professionals to analyze data and calculate the internal dose, and long lapses before dose estimates are available, thus handicapping the assessment of the occupational exposure status and treatment of the worker.

b. Prevention of internal exposure is often more feasible and successful than prevention of external exposure. Contained radioactive material may continue to produce external penetrating fields of radiation, but no internal exposure potential. Portable protective devices (respiratory protection equipment) can minimize internal exposure when containment is not practical.

c. Recommendations of the ICRP in formulating a dose equivalent limit system have resulted in combining internal and external dose. Again, the difficulty and time delay of internal dosimetry make elimination of significant internal exposure an economic incentive.

In facilities that process large quantities of uranium, however, there may be situations in which exposure to work-place airborne activity at low levels occurs daily. The fact that tons of material are handled, rather than gram quantities, and that the material is less toxic (on a mass basis because of low specific activity), make total containment impractical.

4.1.2 Purpose of Air Monitoring

The goal of the air monitoring program is to identify, evaluate, and control internal dose received by workers from routine occupational exposure to airborne radioactive materials, to confirm that source controls are functioning properly, and to assess the exposure resulting from an unusual event. There are two general aspects of air sampling that must receive equal consideration in a properly executed monitoring program. The first involves the methods and equipment by which a sample is collected and analyzed to yield an accurate measurement of the specific radionuclides. The second is the protocol of sampling location, duration, and frequency that focuses on determination of the radionuclide exposure in the work area.

Air monitoring should include both active and passive air samplers. A continuous air monitor (CAM) provides for immediate alarm, warning workers of an unusual release of high levels of airborne radioactive material. This active monitoring is needed for high hazard and high potential areas to provide immediate and timely protective response, while passive sampling provides high-sensitivity activity records, trends, continuous documentation, etc. Three types of air samplers are used to accomplish the air monitoring: general area sampling (GAS), breathing zone sampling (BZS), and personal air sampling (PAS).

The CAM continuously draws air through a sampler that has an active radiation detector. The sampled air is automatically monitored for an increase above normal or background levels of contamination. When airborne activity exceeds the alarm level, workers are warned of the potential problem and prompted to follow alarm procedures. This type of monitor is usually practical only for stationary samplers (GAS or BZS). It is important that a CAM be placed to sample air that accurately represents the most likely area of material release. This will protect most workers from a worst-case exposure and minimize total work-force exposure.

General Air Samplers (GAS)

Air sampling is performed at a single point in the general area of a site where work with radioactive material is being performed. The sampler is placed in a position to give the best overall representation of the area, often in the main airflow exiting the area. Airflow patterns can be determined by tests with tracer smoke or balloons. This method is typically used to measure airborne radioactivity for the following purposes:

a. to determine if the work-place environments are free of significant contamination and are inherently safe for routine occupational activities,

b. to detect measurable air activity which would signal the need for use of respiratory protection equipment,

c. to detect unexpected loss of containment or malfunction of systems (which may not be detected by a CAM), and provide the basis to initiate corrective actions,

d. to detect low-level trends in activity which can signal a gradual loss of containment in early stages, and

e. to estimate personnel exposure retrospectively and evaluate compliance with applicable requirements.

Breathing Zone Samplers (BZS)

Breathing zone sampling is performed by placing air samplers in the immediate area in which workers will spend the majority of their time. The intent is to measure the air activity concentrations to which the workers are actually exposed. The purposes of breathing zone sampling are the same as those listed for general air sampling, but involve a greater number of samples, which gives more realistic information. Breathing zone samplers give earlier, more sensitive detection of release from containment.

Samples should be collected on a schedule corresponding to individual worker activities to best represent inhalation exposure. GAS is generally not a good measurement with which to estimate internal dose. A well-placed network of BZS gives a better representation of inhalation exposure.

Personal Air Samplers (PAS)

Personal air sampling should give the most realistic measurement of individual worker exposure. This involves greater expense, however, to equip personnel with samplers and to process all of the individual samples. Personal air sampling is performed with a small, battery-powered, low-volume (approximately 2-L/min) sampler worn by the worker, with the filter located near the worker's face. This type of sampler is potentially subject to many inaccuracies caused by improper handling, which requires trained personnel to handle the equipment operation. Personal air sampling is often used to validate breathing zone sampling strategy and to conduct special investigations.

4.1.3 Regulations and Limits

The regulations, standards, and limits pertaining to exposure of radiation workers to airborne activity in the work place are based on the probability of injury to internal organs and the total body by

radioactive materials taken into the body. To facilitate control of intake in the work place, standard-setting authorities have calculated derived air concentration (DAC) and annual limit on intake (ALI) which are designed to limit resultant dose to internal organs. Operational hazards are directly controlled by the observance of DAC and ALI values.

The ICRP and the NCRP are independent, non-governmental organizations which set standards and guidance for control of radiation hazards. Governmental agencies implement these recommendations by establishing Federal policy for the protection of workers.

Formal rules for air monitoring for DOE facilities are provided in 10 CFR Part 835. Efforts have been made to keep these rules consistent with ICRP Publication 60 (ICRP 1991a), NCRP Report 91 (NCRP 1987a), and Federal Guidance Report 11 (EPA 1988a). The RCS detailed guidance on the best practices currently available in the area of radiological control. More specific guidance is given in DOE G 441.1-8, Air Monitoring Guide, and technical standards such as this one.

Limits of chemical exposure also need to be observed, especially for materials of low specific activity, such as depleted uranium or non-radioactive materials. The threshold limit value time-weighted average (TWA) for natural uranium is 0.2 mg m^{-3} (ACGIH 1993). TWA is the chemical analog of DAC. In the case of reactor fuel uranium, enriched to about 3%, this corresponds to 4 x 10^{-10} microCi mL^{-1}, which is comparable to the DAC for soluble forms of uranium. However, the OSHA Permissible Exposure Limit for soluble uranium is 0.05 mg m^{-3}, which is more restrictive than the DAC. Soluble forms of such materials can be monitored directly by routine urinalysis, or indirectly by BZS and PAS. Internal deposits of insoluble forms may only be estimated by BZS and PAS, as with asbestos, for example.

4.1.4 Theoretical Considerations and Uncertainties

A discussion of the theoretical aspects of air contamination monitoring, and inherent uncertainties, should be useful in placing air monitoring programs in their proper perspective. In general, air sampling should not be the primary measurement of internal dose, except when bioassay information is unavailable or unobtainable. Evaluation of worker exposure potential in terms of DAC-hours, however, may be a legitimate control measure and may demonstrate compliance with federal directives.

Airborne Concentration

An appropriate air-sampling method should provide samples which accurately represent the average airborne concentration of radioactive materials present in the work place, but should not be used as a measurement of individual exposures, except in unusual circumstances. If air activity data must be used for exposure records, these samples should be collected from the breathing zones of the workers, or by using an established conversion factor for the existing sampler configuration. In contaminated areas subject to significant temporal and spatial variations in the activity concentrations, only personal air samples or virtually continuous samples collected from within the breathing zone of workers can provide reliable breathing zone concentration measurements.

A restricted area, having good ventilation and point sources of contamination, will have substantial variations in the activity concentrations observed at different locations, particularly if the movements of the workers cause resuspension of the activity. The worker often spends time closer to the source of contaminant dispersion than is the location of the nearest BZS. Several researchers have investigated the

relationship between fixed air samplers and spot samples collected at various locations in typical working areas. Discrepancies as great as two orders of magnitude are not unusual.

This deficiency of GAS monitoring for individual exposure records is caused by the high dilution factors that tend to reduce the airborne concentrations before and after contamination reaches the filter head. Much of the air sampled by a GAS originates in another part of the area and does not pass near enough to pick up contamination from the source, effectively reducing the measured concentration by dilution of the collected sample. A release of activity from a malfunctioning containment system can produce large activity concentrations in the breathing zone of the worker. These concentrations are effectively diluted in an unpredictable manner by one or two orders of magnitude before the contamination reaches a monitor located only a meter away. It has been demonstrated that in some operations (such as welding over a short period of time) differences of as much as a factor of 5 between the right and left lapel PAS measurements can be expected.

Most of the field studies that have compared urinalysis results with air sampling in natural uranium facilities have, in general, indicated very poor correlation between the estimated exposures and the bioassay data. This suggests that individual exposure records of uranium workers based on GAS methods have limited validity.

The potential for release of gaseous UF_6, and subsequent generation of its soluble hydrolysis product UO_2F_2, requires special air-sampling considerations in uranium conversion and gaseous diffusion plants, relative to those plants handling less reactive compounds. In these plants, effective processing, as well as worker safety, requires a high degree of containment. Continuous GAS operation to detect loss of containment, coupled with spot air samples, constitute the typical sampling strategy. A study conducted at the Oak Ridge Gaseous Diffusion Plant, concluded that shift-long air samples collected in the general working areas were of little use in predicting worker's urinary uranium excretion. The slight correlation observed was not statistically significant at the 95% confidence level. Thus, gaseous contaminants behave much like particulate contaminants in that localized concentrations can be much greater than the average concentration measured by GAS. These researchers also found that smear samples of alpha activity on work surfaces in the area may provide a better indicator of uranium intake than the GAS records.

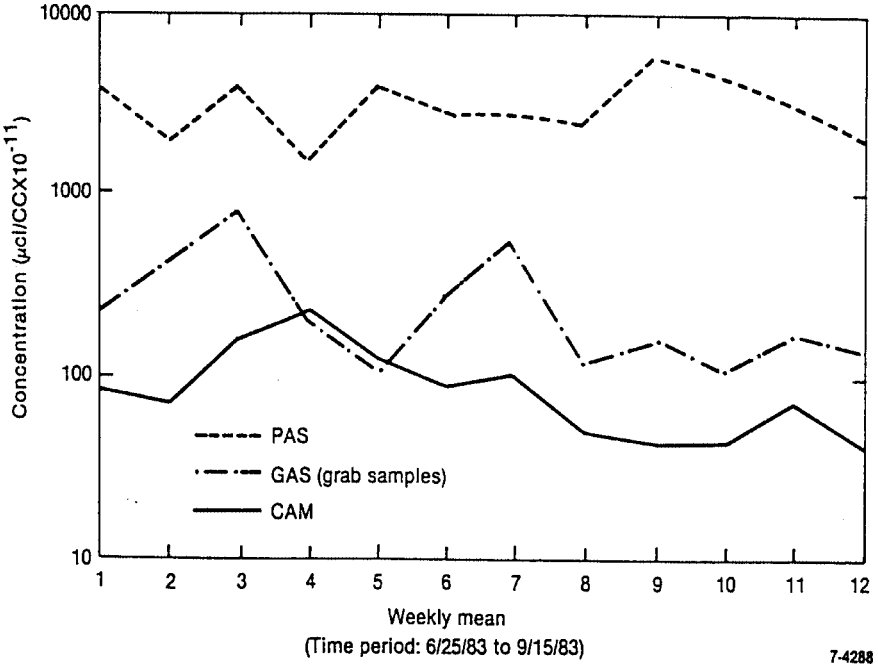
Although transuranic material is handled by DOE uranium facilities only as feed contamination, the unusual characteristics of the transuranic elements make them worthy of separate consideration. The low maximum permissible concentrations specified for these elements and their frequently low specific activities cause extreme difficulties in detection of significant airborne activity. Operations involving significant amounts of elements such as plutonium should be conducted in a ventilated glove-box environment and with monitoring systems capable of detection of small releases involving a few times one DAC. Special CAMs (GAS) and fixed BZSs are the standard air-sampling methods used in facilities of this category in the United States.

A clear example of the wide variations in observed air-activity concentrations that can occur with different sampling techniques is provided by data from the Three Mile Island Nuclear Generating Station, which is typical of operations in a large open building (EGG 1988). Between June and September 1983, over 40 multi-person entries were made into the containment building, providing 949 work-hours of PAS data. Five stationary air monitors were operated continuously at strategic locations throughout the building, and each entry was preceded by the collection and analysis of a high-volume grab sample. All samples were analyzed by a gamma spectrometer, primarily to detect cesium-137, and

by gross beta counting. A graph of the average air-activity concentrations determined by gross beta counting by each of the three sampling methods is shown in Figure 4-1.

The five continuous air samples exhibited good internal agreement when averaged over either 12- or 24-hour periods. However, the grab samples averaged a factor of 3 higher than the continuous air-sampler readings, and PAS samples were a factor of 34 higher. The major reason for this large difference was attributed to resuspension of the surface contamination by the work in progress. These data, coming from a thoroughly monitored and carefully analyzed air-sampling effort, are further evidence that GAS methods should be viewed with caution.

Figure 4-1. PAS versus GAS versus CAM Example of the Degree of Correlation between Type of Sampling TMI-1983



Even when the airborne-activity concentration in the breathing zone of a worker has been accurately measured, there are other physical and physiological parameters that can produce significant uncertainties in dose assessment. The established DACs are derived for each radionuclide assuming a standard volume of air breathed in occupational situations, specified pathways to critical organs, the "standard man" metabolic and elimination patterns, and the physical and biological properties of the isotope. Large variations are encountered, however, in breathing rates and tidal volumes (which depend on working conditions), and there are individual variations in such physiological parameters as lung clearance and metabolic rates. The particle-size distribution of the aerosol and the actual solubility of the inhaled particles can significantly affect the deposition and retention of airborne activity in the respiratory tract. The potential uncertainty in the total dose assessment should include consideration of all of these factors, as discussed in the following paragraphs.

Particle-Size Distribution

In the absence of actual measurement of particle-size distributions, an activity median aerodynamic diameter (AMAD) of 1 um and a geometric standard deviation (GSD) of 2 is often assumed as a

conservative estimate, as laid out in the ICRP-30 methodology. Particles of this size are likely to result in the greatest deposition in the pulmonary region of the lungs. The actual size distribution can be measured with instruments such as cascade impactors, but these are not practical for continuous operation in the work-place environment. Electronic instruments can give continuous information about the optical particle size, but not the AMAD. Thus, particle size can only occasionally be measured to typify the size distribution in a particular situation.

Size-selective inlets for air samplers have been developed to mimic deposition in the respiratory tract, giving more accurate estimates of deposition in the pulmonary region. Non-respirable or noninhalable particles are removed by the inlet, and the respirable or inhalable fraction is collected on a filter. These devices can be useful in minimizing the dose assessment errors resulting from uncertainties regarding the actual aerosol-size distribution; however, they require additional handling and care, and require separate samplers for total airborne activity. If the AMAD is often substantially greater than 1 μm in an area, the addition of samplers with size-selective inlets may be worthwhile. Regulations allowing the substitution of size-selective samplers are not established, however, so special arrangements may be needed with regulatory agencies.

Breathing Rates and Tidal Volumes

The actual air intake of a worker can vary from 5 L min^{-1} to 100 L min^{-1}, although typical variations from the assumed 20 L min^{-1} standard will probably be no larger than a factor of 3. Total air intake depends on the rate of breathing and on the volume of tidal air. The velocity of this air influences the regional deposition of aerosol particles. Newer, more sophisticated lung models include this breathing-rate effect in calculation of dose distribution. Information about individual breathing behavior may be useful in the application of the newer lung dosimetry models. Simpler models, such as ICRP-30, assume that regional deposition is independent of breathing rate, with total deposition determined only by the volume breathed.

Particle Solubility and Lung Clearance

When particles are deposited in the respiratory tract, they are cleared from airway surfaces by several mechanisms. Insoluble particles are cleared by the biomechanical means of macrophage and mucociliary transport, while some particles are retained in pulmonary tissues. Particles of soluble material dissolve, making the contaminant available for other means of transport such as absorption into the blood. Dosimetry of the contaminant depends on how fast the particles dissolve.

Rate of particle dissolution is divided into three categories by the ICRP-30 model. Classes D (days), W (weeks), and Y (years) refer to the retention time of the material in the pulmonary region of the lungs. A retention half-time of less than 10 days is retention class D, a half-time of 10 to 100 days is class W, and half-time greater than 100 days is class Y. Some materials have been described to have characteristic rates of dissolution and are associated with a particular retention class. Many factors can affect the dissolution rate, however, so general assignments to retention classes should be regarded with caution.

The health physicist may have some prior knowledge of the chemical compounds of the nuclides present in an area and may be able to assign them to retention classes. The ICRP-60 dosimetry model provides for a lung retention class designation of aerosols depending on the rate of dissolution; however, actual determination of the lung class for dose assessment can best be determined after an exposure utilizing appropriate chemical and bioassay data, but this can only be accomplished in retrospect. A

prospective approach uses measured dissolution rate of potential contaminants for analysis and treatment of an accidental exposure. Determination of retention class should be a valuable precaution in uranium facilities.

A realistic determination of retention class can be made by collecting a sample of airborne material by using a size-selective sampler and drawing the sample from a process that has a potential for a significant release. The material collected on the filter represents that which would be deposited in the lungs by inhalation. Methods and instruments are now available with the sensitivity needed to precisely measure the rate of dissolution of this small mass of uranium in simulated lung fluid. The same methods can be used on filter samples in operation at the time of an accidental exposure, but the time required to measure dissolution rate (at least 60 days) makes the information essentially retrospective. Prospective measurement of retention class provides for better risk assessment.

4.1.5 Samplers and Instrumentation

Air sampling equipment and monitors exist in a wide range of designs and capabilities, with characteristics specific to the application and need. Samplers range from small portable units that can be worn by an individual to high-volume units permanently mounted in the facility. Flow rates are from a few liters per minute to a few cubic meters per minute.

Key Factors in Selecting Air Samplers

Sensitivity of Detection. In general, the sensitivity required is at least DAC levels; however, in some applications, sensitivity to a small fraction of DAC is desired for early detection of loss of containment, low level trends, etc. Continuous air monitors may only need to alarm at multiple DAC levels in order to be effective in preventing or mitigating personnel exposures to an accidental airborne release.

Type of Sample. In most uranium facilities, particulates in the air are the primary concern, although gaseous forms may be most important in some areas. It may be of interest to collect samples that will allow characterization of the particle size distribution or define a "respirable fraction." In each application, the sample type will dictate the sampler design, filter media, flow rate, etc.

Convenience. Available space, noise level tolerance, portability, and weight also dictate specific designs and capabilities of air samplers and monitors.

Power Requirements. Requirements for battery-powered versus 110-120-VAC line power may dictate sampler selection.

Accuracy. Some sampling is performed to simply detect or make relative measurements of activity levels for which the accuracy requirements are not great. In other situations, accurate measurements of the air breathed by personnel may require an entirely different sampler design to achieve the needed quality assurance.

Reliability and Maintainability. Cost-effective operation and reliability need to be considered for selection of equipment design and for redundancy of components.

Filter Media

Filters should have high collection efficiencies (i.e., >99%) for particles over a wide range of sizes. Many cellulose ester (acetate, nitrate, or mixed ester) or glass-fiber filters meet these requirements and are commonly available. Other filters with reasonably high collection efficiency may be used if required for special applications or assay methods. Selection of a filter type generally involves compromises between filter efficiency, flow resistance, and requirements imposed by the desired assay method.

The specifications of a filter medium often include pore size and filter efficiency. Pore size is determined by filtration of a liquid; the particle size at which the collection efficiency is 95% in water is given as the effective pore size. Filtration efficiency for particles in air, however, is dramatically different. Aerodynamic effects make the collection efficiency dependent on the face velocity through the filter. Airborne particles of aerodynamic size equal to the pore-size rating of a filter are usually collected with high efficiency (>99%). Smaller particles may also be collected efficiently; however, some sizes may substantially penetrate the filter. Particles in the range 0.1- to 1.0 micron diameter are most likely to penetrate a filter. Many manufacturers use dioctylphthalate (DOP) to produce an aerosol of particles 0.3 micron in diameter for testing filter efficiency, following a procedure such as ASTM D 2986-71. Thus, if a filter is rated for efficiency by DOP retention, collection of other particle sizes will be more efficient. Collection efficiency is also increased by higher flow rate for particles >0.1 micron.

Cellulose ester membrane filters have interconnecting pores of uniform size. They typically produce a higher resistance to flow than glass-fiber filters and collect most particles near the surface of the filter.

Glass-fiber filters are made of a mat of randomly oriented glass fibers. They have lower flow resistance than most membrane filters, but trap an appreciable fraction of the particles within the filter mat. This interferes with detection of alpha radiation from the filter.

Cellulose filters are often used for air sampling. They have moderate flow resistance, but relatively poor collection efficiency. Their use may be justified in some situations, but only with the recognition that efficiency for certain particle sizes may be low. Generally, if analytical and sample-handling requirements allow, glass-fiber or cellulose-ester membrane filters are a better choice than cellulose filters.

Each type of filter has inherent advantages and disadvantages. The higher flow resistance of membrane filters may overtax the capabilities of older models of some PAS pumps although membrane filters can be used successfully with many of the new models of pumps. Glass-fiber filters should be substituted if a significant pressure drop occurs with the sampler being utilized.

The surface-collection properties of membrane filters can be an advantage when sampling for alpha and weak beta-emitting materials. Deposition of particles on the surface minimizes energy absorption by the filter medium. This is especially important for alpha spectrometry, where the energy spectrum is substantially degraded. Membrane filters are also advantageous if the assay procedure involves ashing or dissolution of the filters, but they are relatively fragile.

Filter Holders

Criteria for filter holders are simple, but critical. For the collection of large-volume air samples, filter holders should be open-face such that sample air is drawn directly onto the filter surface from the

atmosphere without passing through a tube, orifice, or other obstruction. This precludes loss of the radionuclide to surfaces upstream from the filter. The holder should face downward to avoid collection of large, non-inhalable particles, unless a different position is required. Closed-face cassettes are recommended for small PAS, to protect the filter from direct contamination. Research studies of commonly-available types of closed-face cassettes with 4-mm inlets indicate that these designs have good particle-collection characteristics (at a flow rate of 2 L min^{-1}) and reduce sample contamination problems. Other closed-face filter inlet diameters, geometries, and flow rates may also be acceptable, but have not been characterized.

The filter should receive adequate support so that it is not stretched or torn by the pressure drop caused by the flow of sample air. The filter holder should be free of air leakage around the filter as well as into or through the holder's component parts. Metallic filter holders are generally more reliable and durable than plastics. Finally, filter-changing and holder replacement should be convenient and positive.

Size-Selective Devices

Size-selective devices fall into two categories: respirable-fraction samplers and instruments for measuring particle-size distributions. A respirable-fraction sampler collects a range of particle sizes, with collection efficiency decreasing for larger particle sizes. Particles that penetrate the size-selector represent those that would deposit in the pulmonary region of the lungs. A particle-size distribution instrument collects all particles with classification of particle size. Size-distribution data can be used to calculate the expected deposition of particles throughout the respiratory tract.

Particle-Sizing Devices. Particle-size distribution measuring devices are typically more complex and require more sample analysis than a size-selective sampler. The major advantage in using these devices is that the size distribution of airborne contaminants is useful for estimating regional deposition of inhaled particles in the respiratory tract. This information is more accurate than that provided by a simple size-selective sampler, especially if a large part of the airborne material has particle size less than about 2 µm. Particle-size measurement should be performed only by properly trained individuals, as an investigative tool for evaluating the health hazard posed by a process or procedure suspected of generating airborne contamination.

The cascade impactor is the most commonly available particle-sizing device. Aerosol passing through a cascade impactor is forced through a series of increasingly rapid changes of velocity. The inertia of the particles causes them to deviate from the direction of the airstream at locations where the particle speed and direction are changing most rapidly. Particles of different aerodynamic size deflect to different extents so that larger particles contact the surface of the collection stage. The quantity of material deposited on each stage is measured and the size distribution calculated for the sampled aerosol.

There are some drawbacks to the use of impactors. Cascade impactors subdivide the sample so that more sensitive assay methods may be required for successful use. There is a limit to the mass of material that can be collected on each stage before overloading; inactive dust particles contribute to this mass, but not to the analyte. Each stage of the impactor is a separate fraction of the sample which must be analyzed. This multiplies sample number-capacity requirements of the activity measurement system. Careful calibration of a precisely controlled airflow rate is required for accurate particle-size measurement.

Optical particle-sizing instruments, such as a laser particle-size spectrometer, have the advantage of giving practically real time information. Most of these instruments give only an optical particle size,

however, which must be converted to an aerodynamic size to be useful for dose estimation. They are generally expensive tools used mostly for research.

Respirable-Fraction Samplers. A number of respirable-fraction samplers have been developed, but the cyclone separator is the most widely used and best characterized type. The cyclone is specified by NIOSH and MSHA for personal respirable-mass sampling in coal mines. NIOSH and MSHA currently certify entire sampling systems (PAS pump, cyclone, filter head, and filters) for personal respirable-fraction sampling. This "system" approach may be modified as the result of recent research; however, it does provide an interim standard for performance. The performance of cyclones, pumps, and filters may be characterized to allow intermixing of sampling-train components in future work; at present, however, theoretical prediction of performance of mixed systems is not reliable.

Cyclones are aerodynamic particle sizers, as are impactors, but have some different operating features. They are not affected by loading, so dusty environments are not a problem, although filter loading may limit sampling time. Cyclones are rated for performance at a particular flow rate. Performance at other flow rates cannot easily be predicted and should be determined by testing. In contrast, impactors do follow a simple, well-defined relation between flow rate and size separation.

Alternatives to mechanical methods of particle-sizing exist and other respirable-fraction separators may be available in the future. Combined total and respirable-fraction samplers would be desirable; such designs retain both the respirable and non-respirable fractions so that total airborne activity can be estimated.

4.1.6 Sample Activity Measurement

Most sample analyses at uranium facilities are performed by quantifying the radioactivity by counting the samples collected. Some fluorometric analyses are performed with equivalent sensitivity. Kinetic phosphorescence analysis is available with substantially greater sensitivity.

Alpha Counting. Alpha particles can be counted with ionization, proportional, scintillation, or other solid state detectors. The major drawback is that relatively little particle penetration, in the filter or in the dust loading, can result in a low reading caused by self-absorption of the alpha particles.

Alpha Spectrometry. Measurement of the energy spectrum of alpha-emitters on a filter paper is possible and very beneficial in some applications in identifying or verifying the identity of the isotopes present. Typically, semiconductor detectors are the choice, and membrane filters or other surface-collecting filters are used with very low dust loading.

Beta Counting. Thin-window GM, ionization, proportional, and solid state detectors are used for beta counting. Because of the wide range of beta-particle energies of even a "single energy" emitter, careful energy calibration is necessary. Beta counting results are less dependent on self-absorption effects.

Beta Spectrometry. Beta spectrometry has recently become feasible through developments in tissue-equivalent plastic detectors. For routine isotopic identification, this method is not as useful, but it may provide valuable shielding information, etc.

Gamma Spectrometry. NaI and GeLi detectors can provide essential isotopic identification of gamma-emitters.

Precautions. The intricacies and procedures of sample analysis are beyond the scope of this manual. However, a few general precautions are important to mention. The naturally occurring radionuclides, radon and thoron and their decay products, are present in all atmospheres in widely varying concentrations. These radionuclides are typically present in higher concentrations than the isotopes of interest, and tend to interfere with radiometric analysis, unless the short-lived progeny are given time to decay after sample collection. Radon progeny, which are much more abundant than thoron progeny in most areas, decay with an effective half-life of about 30 minutes and a counting delay of 3 hours may be adequate. Thoron progeny decay with an effective half-life of 10.6 hours, and where they exist in significant concentrations, a counting delay of several days is advisable. The presence of either radionuclide on a filter can be detected by recounting two or three times at intervals of a few hours.

The sensitivity of any counting method depends primarily on the background count rate of the counting instrument; estimates of low radionuclide concentrations can be seriously in error if the counting background is not accurately known. Even in stable instruments for which the background count may be quite constant, a daily check is advisable because of the possibility of contamination from sample material. Background counts should be made with a blank filter in place because some filter media contain trace amounts of radioactivity.

Counting instruments also require periodic standardization. Standard sources used for this purpose should match the samples both in size and energy.

The active (upstream) sides of filters collected in clean atmospheres can be difficult to identify. Some convention should be followed by sampling personnel to ensure that the proper sides of filters will be counted. This may consist of marking one side of the filter or placing the filter in the sample holder consistently with the exposed side toward the identifying number or label on the holder.

4.1.7 Continuous Air Monitors

The combination of an air sampler and an activity counter into a single device for automatic operation and alarm control constitutes a CAM.

4.1.8 Monitoring Strategies and Protocols

Designing an air-sampling program for the work place is a complex task because each facility has unique design and operational characteristics. It is important that the radiological control personnel who coordinate the sampling program have a thorough understanding of basic facility operations, especially with respect to the potential each operation has for generating airborne material. In addition, these personnel should be familiar with the working habits of potentially-exposed workers. The success of most sampling programs depends on the ability of radiological control personnel to accurately assess worker exposure risk and properly select workers for personal air sampling. This can only be accomplished by well-trained, observant safety personnel.

The following questions should be considered for an airborne activity hazard evaluation:

a. Where are the potential aerosol generation and release locations in the work-site, and what is the magnitude of potential exposures associated with each?

b. How effective or failure-prone are the physical and procedural barriers that protect the worker from airborne radioactive material generated at these locations?

Potential Sources of Airborne Contamination

Virtually every work site has at least one of the fundamental mechanisms for the generation and suspension of particulate material. The following descriptions of some of the basic mechanisms of aerosol generation are intended to help radiation safety personnel recognize processes which have inherently higher risk:

a. mechanical fragmentation, i.e., grinding, abrasive saws, sandblasting.

b. combustion, burning materials producing smoke, fumes, etc.

c. heating - many materials produce aerosols when heated, without actually igniting.

d. formation from bubbles, foams, or highly agitated liquids - fine solid aerosol particles can form from larger, evaporating liquid droplets.

e. condensation of liquid or solid particles from the gas phase.

f. formation of particles from the products of gas-phase reactions, e.g., $UF_6 + 2\ H_2O \rightarrow UO_2F_2 + 4$ HF.

g. formation of solid, radioactive nuclides from gaseous parent nuclides - these radionuclides usually attach to existing, nonradioactive aerosol particles.

h. adsorption of gaseous, radioactive nuclides on non-radioactive aerosols.

The program designer should be familiar with the routines and working habits of workers, especially those in situations where there is a greater potential for generating locally high concentrations of airborne contamination. This will assist in planning for exposure prevention and in selecting suitable sampling methods. Some factors to consider are:

a. Worker location and mobility - If the worker stays in a fixed location, fixed breathing-zone sampling may be useful for individual exposure estimation. This sampling may be performed using moderate flow-rate pumps (30 to 90 L min^{-1}) which can be located within a few feet of the worker. Mobile workers should be surveyed using PAS to obtain a breathing-zone sample.

b. Direct versus remote handling of radioactive material - Remote-handling facilities such as hot cells or caves usually restrict the workers to a fixed location. Well-located fixed sampling heads may be adequate for breathing-zone sampling at these work areas, provided that they have been properly located. As previously noted in this section, determining the proper sampling points for fixed breathing-zone sampling at fume hoods, glove boxes, etc., is not a straightforward exercise, and PAS may be the most expedient means for sampling a worker's true breathing zone.

   Direct-handling is commonly performed on material with relatively low intrinsic hazard, e.g., uranium metal or compounds. This sort of material may be moved around the work site and directly manipulated at a number of locations. Fixed breathing-zone samplers usually will be unsatisfactory in these situations, and PASs would be required for estimating an individual worker's exposure in DAC-hours.

c. Material with high intrinsic hazard is usually well contained, but if it is moved over wide areas in process flows, there is a potential for release at any point. The effectiveness of containment, in the process flow at locations where workers have access, is a major factor when considering use of PASs.

When evaluating risks associated with direct handling of radioactive materials, the variation in techniques employed by different workers to perform the same task must also be considered. No two workers perform the same operation in exactly the same manner. Aerosol production may depend on how each individual performs the operation (i.e., rate, accuracy, operating temperatures, etc.).

Characterization of Controls

For the purpose of evaluating work-place controls, work sites can be characterized as either "tightly controlled" or "loosely controlled." Tightly controlled work areas are preferred in all cases, but there are situations where good control is difficult or not reasonably achievable. PAS monitoring can help define those operations that pose the greatest radiological control problems and thus facilitate decisions to improve specific work situations.

Significant exposure incidents in highly controlled (i.e., tightly controlled) areas usually are the result of isolated and unforeseeable events, which are complete departures from the normal material-processing routine. These events usually include loss of containment. In tightly controlled areas, PAS can serve as a means of detecting a failure of containment because work locations may be located near potential release points, and inadequate physical controls may be apparent only during an operation performed by a worker.

4.2 SURFACE CONTAMINATION CONTROL

Uranium contamination on plant surfaces, such as floors and walls, does not present a significant risk to personnel unless the uranium becomes airborne by resuspension and is inhaled. The probability of significant airborne concentrations resulting from resuspension of uranium as a result of normal activities (such as walking) is low; however, any activity that vigorously disturbs the surface (such as floor sweeping) increases the probability of significant airborne concentrations of uranium. Resuspension is a function of both the chemical and physical forms of the uranium contamination. External exposure hazards from surface contamination can become an important concern when uranium decay products and/or fission products accumulate on surfaces. In some instances, efforts to decontaminate uranium compounds may leave behind insoluble uranium and decay product compounds which could present an external exposure hazard. Good industrial housekeeping practices and normal standards of personal hygiene will usually ensure that uranium surface contamination does not present a significant exposure hazard. However, even if the probability of resuspension is low, surface contamination on floors can result in contamination of shoes and thereby result in the potential for tracking of contamination into uncontrolled areas. Thus, contamination on surfaces must also be adequately controlled to prevent transfer of contamination above acceptable levels.

Several other contamination control objectives can be accomplished by a program of monitoring and control of surface contamination:

- The program can be designed to provide information to detect containment failures or departures from good operating practices.

- It can provide information that will assist in the design and evaluation of personnel monitoring, bioassay, and air monitoring programs.

- The contamination monitoring and control program will provide information to establish operating zones, guidelines and constraints for radiation protection, and operational procedures.

- The program will provide practical assurance that uranium contamination is confined to the operating areas of the plant and that the potential is minimized for contamination of personnel, the environment, and sensitive analytical areas.

Contamination control of work surfaces such as tools, equipment to be worked on (disassembly, machining, etc.), desks or tables in process areas, etc., is of greater concern than contamination on floors. The likelihood of personnel contamination, ingestion of material through hand contamination, or inhalation of resuspended uranium compounds through work activities represents a significant potential for exposure of personnel. Work activities that involve the destruction of surfaces such as grinding, machining, drilling, or cutting can generate significant levels of airborne uranium compounds. Operations such as welding, burning, heating, etc. can alter the physical and/or chemical state of uranium compounds that are on the surfaces of equipment. Job-specific monitoring is required to establish protection requirements as a function of surface contamination levels.

4.2.1 Reporting and Documenting Contamination Levels

Radiological control programs require the performance of contamination surveys to determine existing conditions in a given location. Maps with sufficient detail to permit identification of original survey locations should be maintained. Records shall contain sufficient detail to be meaningful even after the originator is no longer available. Contamination surveys should be recorded on appropriate standard forms and include the following common elements:

- date, time, and purpose of the survey,

- general and specific location of the survey,

- name and signature of the surveyor and analyst,

- pertinent information needed to interpret the survey results, and

- reference to a specific radiological work permit if the survey is performed to support the permit.

In addition, records of contamination surveys should include, at a minimum, the following information:

- model and serial number of counting equipment,

- contamination levels (using appropriate units) and appropriate supporting parameters, including counting efficiency, counting time, correction factors, type of radiation, and whether the contamination was fixed or removable,

- location of areas found to contain hot particles or high concentrations of localized contamination, and

- follow-up survey results for decontamination processes cross-referenced to the original survey.

Records for the release of material and equipment from radiological areas to controlled areas should describe the property, the date on which the release survey was performed, the identity of the individual who performed the survey, the type and identification number of the survey instrument used, and the results of the survey. Additional details on radiation records can be obtained from DOE G 441.1-11, Occupational Radiation Protection Record-keeping and Reporting Guide and in the RCS.

All skin and personal property contaminations should be documented and evaluated to help improve the contamination control program. Documentation should include the following:

- the person's name and work group,

- the location, amount, and type of skin or personal property contamination,

- the results of decontamination, and

- a description of circumstances involved in the occurrence, such as radiation work permit number, protective clothing required, and protective clothing actually used.

4.2.2 Monitoring

Radiological workers are often assigned tasks that could expose them to radioactive material. It is not sufficient to rely exclusively on equipment design to minimize contamination and exposure in the work place. A radiation protection program shall include both monitoring of the workers (discussed in Section 4.3) and monitoring of the conditions in the workplace (10 CFR 835.401 - 835.403, 835.1101- - 835.1102). Both functions are essential to a good radiation monitoring program.

Continuous monitoring should be provided during the periods of high or unusual risk associated with the work in the area. Periods of high or unusual risk include the potential or actual breaching of the integrity of the glove-box or associated systems, including such maintenance as replacement of panels, glove changes, bag-out operations, replacement of filters, or repair of vacuum systems. Work that involves the use of temporary enclosures (greenhouses) should be provided with continuous coverage by an RCT. For decommissioning, most activities will be new, unique, and have no historical precedent. Consequently, high and unusual risks may become the norm and the use of temporary controls and continuous coverage the routine.

Monitoring of the work place is an essential element of every routine surveillance program. It can be effectively accomplished using any or all of the techniques that are discussed in this section. The rigor with which all of the various elements of a radiation monitoring program are applied should be tailored to meet the needs of the individual work areas and depend on the kind and quantity of radioactive material present and its potential for dispersion. Each program should be designed to meet

existing needs, but also should be flexible to allow for incorporation of the possible advantages to be provided by the various available monitoring practices. Monitoring practices include, but are not limited, to the following:

- contamination surveys of the workplace,
- release surveys,
- external exposure surveys,
- airborne contamination surveys, and
- routine surveillance by an RCT.

4.2.2.1 Contamination Surveys of the Workplace

The radiation monitoring program should include documented survey procedures, a system for maintaining survey results, and contamination control limits for "fixed" and "removable" contamination. The results of contamination surveys should be reported in activity per area (e.g., dpm/100 cm^2) except for large-area swipes and swipes of very small items. This permits interpretation of the recorded data without requiring knowledge of instrument efficiency or geometry.

All workplaces should be monitored for contamination levels on a regularly scheduled basis. The frequency of such surveys will depend on the potential for dispersion of the radioactive material. As a minimum, all gloves, work surfaces, floors, equipment, etc., within the workplace should be surveyed according to the frequencies listed in DOE-STD-1098-99, Radiological Control (DOE 1999a).

The change room and other support facilities within the controlled area should be surveyed for contamination daily. Continuous air monitors, survey instruments at step-off pads, and hand and shoe counters should be functionally tested daily or once per shift in support of the weekly and monthly surveys.

These frequent surveys are also part of the routine surveillance program and permit immediate follow-up if low-level contamination is detected to minimize the potential for major incidents. Some fixtures and support areas outside the controlled area, such as door knobs and telephones of adjacent offices and the lunchroom, should also be surveyed daily. Other support areas should be surveyed monthly. If routine survey results detect any contamination in a given area, more detailed surveys should be performed to determine the extent and source of the contamination.

Two principles should be adopted to preclude the possibility that contaminated waste would be disposed of as ordinary waste: 1) all process and controlled area waste should be considered contaminated, and 2) mechanisms should be established that prevent the mixing of contaminated and non-contaminated waste.

4.2.2.2 Release Surveys

As stated in Section 2.1.4.1., transuranics exist in small quantities of recycled or reclaimed feed materials. In many instances, these isotopes may be limiting for release of materials. For transuranic and uranium radionuclides, the contamination level (fixed and removable) at which surfaces are

considered contaminated are listed in Appendix D of 10 CFR 835. That document also specifies the criteria for the release of materials and equipment from radiological areas to controlled areas.

Detailed requirements for unrestricted release of materials and equipment from controlled areas are found in DOE Order 5400.5 Ch. 2, Radiation Protection of the Public and Environment (DOE 1993b). Figure IV-1 of DOE Order 5400.5 indicates that the allowable total residual surface contamination for transuranics is reserved, i.e., no value is given. In essence, this requires release values for transuranic contamination to be developed through the project offices in the field and approved by the DOE Headquarters Program Office.

4.2.2.3 External Exposure Surveys

To delineate the levels involved, measurements of external exposure should be made at the time a program is established at all locations where personnel exposure occurs. Additional photon and neutron measurements should be made at the same frequency as the contamination surveys. The buildup of contamination in glove boxes and on gloves and equipment may contribute substantially to the external dose rates.

4.2.2.4 Measurement and Survey Techniques

This section discusses four types of contamination surveys that are typically used in DOE facilities. Surveys for removable contamination include a large-area wipe survey and a swipe or smear survey. Surveys for total/fixed contamination include a scan survey and a statistically-based survey. These surveys, or a combination of them, are used to survey material for release from radiological control. The appropriate use of each type of survey is discussed.

Surveys for Removable Contamination

Two types of surveys are used for removable contamination: a large-area wipe survey and a swipe or smear survey.

A large-area wipe survey is used to qualitatively detect gross removable contamination. A large-area wipe survey is typically performed using a large floor cloth and a dust mop type handle to wipe large areas. This technique tends to concentrate any low levels of removable contamination that may be present. The surface to be wiped and the wiping material should be industrially clean (i.e., free of debris, grease, etc.) to reduce self-absorption of alpha contamination. The survey is performed by wiping the surface of the area being surveyed and conducting frequent checks of the cloth using a portable instrument. For detection of alpha-emitting isotopes, a nonabsorbent material should be used. Removable contamination will be accumulated and concentrated on the wipe, increasing the probability of its detection. Checking for contamination is conducted by placing an alpha-measurement instrument approximately 0.25 in. (0.6 cm) from the surface of the wipe for 5 seconds, and the count rate observed. If no radioactivity above background is measured, then the material is not contaminated with removable contamination. If radioactivity above background is measured, the material is contaminated. Technical smears (i.e. 100 cm^2) need to be taken to quantify removable contamination levels. Depending upon the specific circumstances, a series of smears may be required to locate and quantify the contamination within the area covered by the large-area wipe. In most instances, if contamination is detected on the large-area wipe, decontamination should be considered.

For transuranic radionuclides, the guideline values for removable contamination are lower than the MDA of portable instruments. During a wipe survey, the surface area of the material must be large enough that the quantity of radioactivity collected on the wipe will be greater than the MDA of the instrument. Wipe surveys of areas smaller than this minimum surface area require more sophisticated measuring instruments, such as a scaler measurement, and the entire surface of the material should be wiped. The minimum area for using a large-area wipe survey is given by where GV is the guideline value of the potential contaminant, given in Table 4-1.

$$A_{minimum} = \frac{MDA}{GV_{removable}} \times 100\ cm^2$$

The purpose of a smear survey is to locate and quantify removable contamination that is known or suspected to exist. For small items, a smear may be used at any time to verify the item's contamination status. A smear or swipe survey is performed by wiping a cloth, paper, plastic foam, or fiberglass disk over a 100-cm^2 area of the surface. The swipe should be taken with a dry medium using moderate pressure. A common field practice is to use two fingers to press the swipe medium against the surface to be swiped. The swipe is then moved along an "S" shaped path that has a nominal length of 8 in. (20 cm) to 10 in. (25 cm).

When the potential contaminant emits alpha radiation, paper or fiberglass filter papers should be used so that alpha activity is not attenuated by becoming imbedded in the wipe. To improve the detection limit, smears may be taken over areas larger than 100 cm^2. However, the size of the area smeared should be limited to prevent buildup of material (radioactive or otherwise) that would attenuate alpha radiation. The current practice at DOE facilities is to use the 100 cm^2 area as the minimum size of objects being smeared. Appropriate corrections should be made for objects smaller than 100 cm^2.

If contamination is detected during a scan survey for fixed contamination, a swipe survey for removable contamination should be performed to determine if the contamination is fixed and to quantify any removable contamination. If no contamination above the guideline values for removable contamination in Table 4-1 is detected during the smear survey, the contamination is fixed, and the area should be posted appropriately.

A smear survey may be used routinely to detect removable contamination, especially for contamination surveys of radiological areas.

Table 4-1. Surface Contamination Values,[a] dpm/100 cm^2

Nuclide	Removable[b, d]	Total (Fixed+Removable)[b, c]
U-nat, ^{235}U, ^{238}U, and associated decay products	[g]1000	[g]5000
Transuranics, ^{226}Ra, ^{228}Ra, ^{230}Th, ^{228}Th, ^{231}Pa, ^{227}Ac, ^{125}I, ^{129}I	20	500
Th-nat, ^{232}Th, ^{90}Sr, ^{223}Ra, ^{224}Ra, ^{232}U, ^{126}I, ^{131}I, ^{133}I	200	1000
Beta-gamma-emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except ^{90}Sr and others noted above[e]	1000	5000
Tritium and tritiated compounds[f]	10,000	N/A

(a) The values in this table, with the exception noted in footnote (e), apply to radioactive contamination deposited on, but not incorporated into the interior or matrix of, the contaminated item. Where surface contamination by both alpha- and beta-gamma emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides apply independently.

(b) As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

(c) The levels may be averaged over one square meter provided the maximum surface activity in any area of 100 cm^2 is less than three times the value specified. For purposes of averaging, any square meter of surface shall be considered to be above the surface contamination value if: 1) From measurements of a representative number of sections it is determined that the average contamination level exceeds the applicable value; or 2) it is determined that the sum of the activity of all isolated spots or particles in any 100 cm^2 area exceeds three times the applicable value.

(d) The amount of removable radioactive material per 100 cm^2 of surface area should be determined by swiping the area with a dry filter or soft absorbent paper, applying moderate pressure, and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. (Note–The use of dry material may not be appropriate for tritium.) When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area shall be based on the actual area and the entire surface should be wiped. It is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.

(e) This category of radionuclides includes mixed fission products, including the ^{90}Sr which is present in them. It does not apply to ^{90}Sr which has been separated from the other fission products or mixtures where the ^{90}Sr has been enriched.

(f) Tritium contamination may diffuse into the volume or matrix of materials. Evaluation of surface contamination shall consider the extent to which such contamination may migrate to the surface in order to ensure the surface contamination value provided in this appendix is not exceeded. Once this contamination migrates to the surface, it may be removable, not fixed; therefore, a "Total" value does not apply.

(g) (alpha)

Scan Survey for Fixed Contamination

A scan survey for fixed contamination requires passing a detector attached to a portable instrument over the surface of the area being surveyed at a fixed, known scan speed and at a specified distance from the surface. Typically, the scan speed is 2 in./s (5 cm/s) and the maximum distance is 0.25 in. (0.6 cm) for alpha-contamination instruments. A scan survey should be used to survey material that resides in an area controlled for contamination purposes, an area where unsealed radioactive sources are used, or an area surrounding an area controlled for contamination purposes. A scan survey in conjunction with a swipe survey should be used to release from radiological control material with a total surface area less than 5 ft^2 (0.46 m^2). A statistically-based survey, which will be discussed later, should be used to release from radiological control material with a surface area greater than 5 ft^2(0.46 m^2).

During the performance of scan surveys, the audible response of the instrument is faster than the needle deflection. Therefore, audible response should be used in conjunction with meter readings. For alpha surveys, the surveyor should pause for 3 to 5 seconds each time an individual pulse is detected in

order to allow a longer count time at the location of the detected pulse, until it is determined whether the response indicates random background noise or detected contamination.

The most critical factor affecting a scan survey measurement is the speed at which scan surveys are performed. Counting time is inversely proportional to scan speed. For instruments with larger detector faces, the scan speed is faster for a given rate of meter movement because a point on the surveyed surface remains beneath the window longer. To ensure that low levels of contamination can be detected, it is necessary that a maximum scan speed be mandated and that this speed be implemented during field measurements. Empirical information is available indicating that, for most instruments in current use, a maximum scan speed of 2 in./s (5 cm/s) can detect contamination at or above the total contamination values specified in Table 4.1 for nearly all radionuclides with 67% confidence.

4.2.3 Release Criteria

Material in contamination, high contamination, or airborne radioactivity areas, shall be treated as radioactive material and shall not be released to controlled areas if either of the following conditions exist:

- Measurements of accessible surfaces show that either the total or removable contamination levels exceed the values specified in Table 4-1.

- Prior use suggests that the contamination levels on the inaccessible surfaces are likely to exceed the values specified in Table 4-1 (10 CFR 835.1101).

Material that has never been in a contaminated or airborne radioactivity area may be removed to controlled areas without survey. If the history of the item is unknown, it is appropriate to assume that it may have been in a contaminated or airborne radioactivity area.

To release material from radiological control, a methodology has been developed to reduce the time required to perform a survey while meeting DOE requirements. A logic diagram of the protocol is shown in Figure 4-2. The methodology ensures, with 67% confidence, that the guideline values of DOE Order 5400.5 and 10 CFR 835 are met. Note that Figure IV-1 of DOE Order 5400.5 does not specify values for the allowable residual surface contamination levels for transuranics. For this discussion, the residual contamination levels for transuranics given in 10 CFR 835, Appendix D are used.

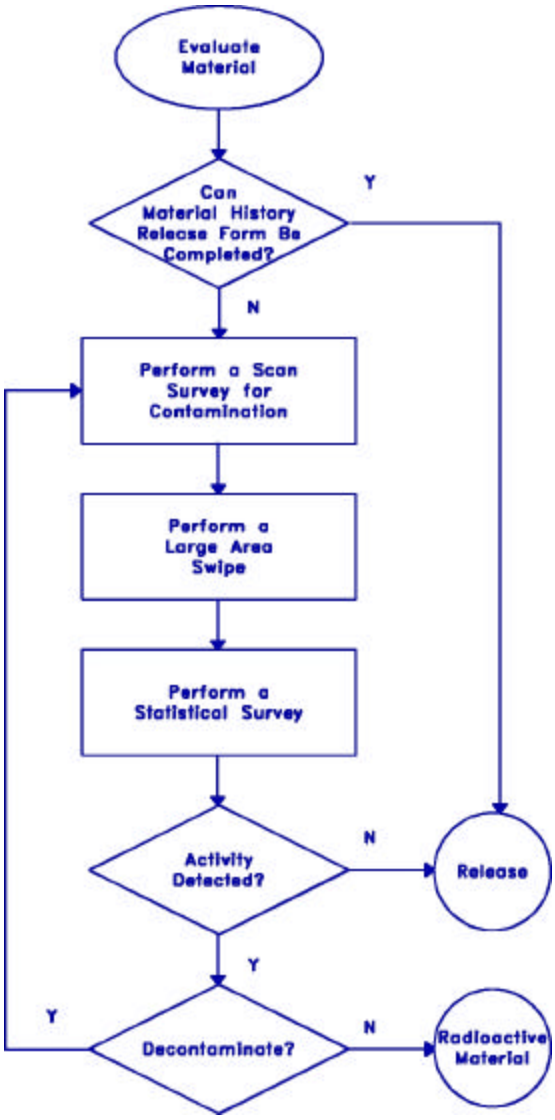


Figure 4-2. Protocol for Release of Materials

The material release methodology has four main components: material evaluation, scan survey for fixed contamination, large-area wipe survey for removable contamination (described above) followed by technical smears as necessary, and statistical survey for fixed contamination. The material evaluation process involves consideration of the previous known uses of the material, as well as typical uses and the environment in which the material was used. Material evaluation places the material into one of two categories: not potentially contaminated or potentially contaminated.

Non-radioactive material can be released without an instrument survey if its documented history ensures:

- that it has never been used or stored in an area controlled for contamination purposes (i.e., a contamination area, high contamination area, or airborne radioactivity area),
- that it has never come into contact with unsealed radioactive sources,
- that it has not been stored or used in a radiological buffer area (RBA) surrounding a contamination area, high contamination area, or airborne radioactivity area.

This material may be considered to be not contaminated and an instrument survey is not necessary. A material history release form should be used to document the release of material that is known to be free of contamination by its history of use. If the material history release form cannot be completed, or if the history of the material is unknown, an instrument survey must be made of the material. Material released from RBAs around contamination areas, high contamination areas, or airborne radioactivity areas should also be evaluated using an instrument survey.

The material evaluation process should also consider the nuclides to which the material was potentially exposed. If the material was exposed to significant quantities of nuclides that are difficult to detect, including tritium ^{14}C, ^{125}I, or ^{129}I, an appropriate survey methodology should be applied.

A scan survey for fixed contamination requires passing the detector of an alpha and a beta/gamma survey instrument, as applicable, over the accessible surface of the material. The detector should be moved at a constant rate that allows detection of contamination at a level equal to three times the guideline value. If a change in the audible output of the instrument is heard, the area under the window of the instrument should be re-surveyed using a stationary measurement for 3 to 5 seconds. If the increase does not persist, the scan should continue. If the elevated counts persist, the material is contaminated and should not be released. This procedure should be followed until the surface of the material has been surveyed.

The scan survey for fixed contamination ensures that none of the material's surface is contaminated above three times the guideline value. If no contamination above background is detected during the scan survey, a large-area wipe survey for removable contamination should be performed. If contamination above background is detected, then decontamination of the material should be considered and the methodology described in this document should not be applied.

Following the scan and large-area wipe surveys, a statistical survey for fixed contamination should be performed. The survey methodology should be used for both beta/gamma and alpha contamination, unless only one type of potential contaminant exists in the facility. If no measurements above background are observed, the material may be released from radiological control.

The fixed survey measurements should be chosen using random detector placements over the entire surface of the material. It may be prudent to bias some of the measurements toward those areas that are more likely to be contaminated, including handles, horizontal surfaces, stains, cracks, and other surface anomalies in which foreign material typically accumulates. This type of selection bias will further increase the confidence associated with the statistical survey method.

Measurements performed to release material should be made in a low-background area unless the MDA of the instrument in a high-background area is known and appropriate considerations are made. If material is being surveyed for release from a radiological area, performing measurements in a low-background area may not be possible. If background count rates are high enough that the release guideline values cannot be measured in the radiological area by using portable survey instruments, a survey for removable contamination should be performed to avoid spreading removable contamination from the radiological area. If the survey for removable contamination does not indicate the presence of contamination in excess of background levels, the material may be moved to an area with a lower background for an immediate fixed contamination survey.

4.2.3.1 Uranium Contamination Detection

The detection and measurement of uranium contamination is necessary to ensure control of contamination and compliance with DOE requirements. Typically, detection of uranium contamination has been performed using the alpha activity. However, for some conditions and situations, detection of the beta/gamma radiations from uranium decay products may be a more sensitive and more appropriate monitoring technique. For natural uranium, depleted uranium, and the lower levels of enriched uranium that are in equilibrium with their decay products, the detection sensitivity for the beta/gamma radiations is about five times more sensitive than by the detection of the alpha alone. If the uranium is highly enriched or has been very recently processed, detection using the alpha radiation is necessary because there may be little or no decay product radiations present.

Detection of uranium contamination may require use of beta/gamma-sensitive instruments when surveying upholstery material, rugs, cloth and wet surfaces. Because of the range and ease of shielding alpha particles, burial or surface liquid may preclude the detection of the alpha radiation. The use of GM detectors, such as the thin-window detector probe, are particularly useful in these situations. In some instances, a thin Nal detector may be better than a GM detector for detecting low-energy photons from uranium contamination.

Many of the processes used in uranium facilities may separate and/or concentrate impurities or decay products of uranium. Examples of these processes are uranium recovery from ore, reduction of green salt to metal, UF_6 conversion, casting of metal, and uranium oxidation. Radionuclides of particular importance are ^{234m}Pa and other decay products and trace impurities such as ^{99}Tc, ^{239}Pu, and ^{231}Np. In addition to the separation processes, some of the decay products of uranium may be selectively accumulated in tank and pipe liner material. Dose rates of up to 150 mR/h, attributed to radium accumulation, have been measured from neoprene liner material. Dose rates from furnace lids and crucibles have been measured as high as 30 rad/h.

Detection and measurement of uranium contamination, both surface and airborne, require a knowledge of the process and of the separation and concentration mechanisms. Depending upon the process, the time since separation, and the isotopic ratios of the uranium, contamination resulting from uranium operations may be almost totally alpha or totally beta/gamma-emitters. Consequently, detection techniques may require the capability to detect all types of radiations. Appropriate monitoring in most facilities requires both types of surveys, but on differing frequencies.

4.2.4 ALARA Guidelines

Contamination levels should be maintained ALARA to minimize the potential for the spread of contamination and to reduce the protective measures and equipment required. Control of radioactive

material at the source and prevention of the generation of contamination are generally more effective and less costly than remediation.

4.3 PERSONNEL CONTAMINATION CONTROL

Contamination control should be achieved primarily by physical design features, including engineering controls (see the discussion above), such as containment, confinement, and ventilation control. Only if the primary controls fail or if there is a potential for personnel contamination during an activity are administrative controls such as protective clothing and respirators advisable.

4.3.1 Monitoring Philosophy

Although the primary hazard to personnel from uranium is from internal exposure, contamination is also of concern because of potential skin doses. Additionally, an objective of the contamination control program is to confine uranium contamination to production/work areas and to minimize, to the extent practicable, any spread of contamination to areas outside the plant or to the public. Therefore, guidelines for allowable contamination on personnel and personal clothing/shoes both inside the plant and prior to exiting radiological areas are required. Also, a personnel monitoring program must be developed with adequate monitoring equipment and sensitivity to provide assurance that contamination is effectively controlled. The guidelines should be developed considering the following factors:

a. the need to prevent detectable activity from appearing outside the controlled area,

b. the degree of risk to the health of the employees, their families, and the public from contamination removed from the plant,

c. the technical feasibility of measurement of the guide levels,

d. commitment to the policy of keeping contamination to the minimum practical level, and

e. the presence of other radionuclides due to the presence of recycled uranium contaminants or uranium daughters.

4.3.2 Monitoring Program

Instrumentation should be provided and persons entering a uranium work station should be required to survey themselves at established frequencies. As a minimum, workers should survey their gloves and coverall sleeves each time they are withdrawn from a glove box (or similar containment system) and after each glove replacement or bag-out operation.

In addition to mandatory monitoring at the exit to areas controlled for contamination, personnel monitoring for contamination should be mandatory at the egress from controlled areas and be conducted in a verifiable manner. Assurance should be provided that personnel are monitored prior to breaks, meals, or exits from the plant site. Portal monitors, hand-and-shoe counters, and/or portable survey instruments may be used for this purpose. If employees are instructed to perform self-monitoring, the equipment should be set up in a "go/no-go" mode and employees should be clearly instructed in the required actions to take if predetermined action levels are exceeded. Frequent audits should be performed to verify that controls are adequate. Limiting the number of egress points and controlling personnel movement can minimize the numbers of locations where positive control of personnel monitoring must be maintained.

Monitoring of shoes, clothing, and hands should be required prior to leaving a work station where uranium or uranium contaminated material was handled. Following routine work, self-monitoring upon exit is usually considered adequate if the person has received proper training in the use of the instrument provided. The instrument should clearly detect an unacceptable level of contamination.

After performing work that involves a high potential for intake of radioactive material, an RCT should perform exit monitoring of the worker.

After performing work that, in retrospect, involved a high potential for intake of radioactive material, each worker should provide a swipe of the nasal passages, to be counted immediately. If respiratory protection was worn, there is no need for nasal swipes unless a breach of the respirator seal is suspected. If facial contamination is detected during the exit contamination monitoring, a nasal swipe should be taken and counted immediately. Chapter 5 provides guidance on the actions to be taken if a nasal swipe is positive.

4.3.3 Protective Clothing

Various types of protective clothing, including laboratory coats, shoe covers, gloves, coveralls, plastic or rubber suits, and air-purifying or atmosphere-supplying respiratory protective equipment, may be required for operations with transuranic radionuclides. The use of company-issue shoes and clothing for employees with work assignments in process areas can be a major aid in contamination control. Some facilities are using disposable anti-contamination clothing. This may be a cost savings from a handling standpoint. However, disposal costs must be considered.

4.3.4 Respiratory Protection

While every attempt should be made to control uranium hazards utilizing physical design features, including engineering controls, the use of respiratory protection is an essential part of the radiological control program.

As with personnel protective equipment, respiratory equipment utilized must also provide protection from the full range of airborne hazards that may be encountered in the work environment. For example, a uranium metal machining operation may have both an airborne uranium oxide hazard and an airborne hazard from solvent vapors. The respirator utilized must be effective for both types of hazard. Also, one airborne contaminant may interfere with the effectiveness of the canister in an air-purifying device that is designed for a different contaminant. For example, a corrosive gas, such as hydrofluoric acid (HF), may attack a HEPA filter and render the filter ineffective. It is important to coordinate the use of respiratory protection requirements with other health protection groups. The respiratory protection program should also be in compliance with ANSI Standard Z88.2, American National Standard for Respiratory Protection (ANSI 1992) requirements. In specifying respirators for various applications, one should always know the applicable protection factors to determine that the range of hazard that may be encountered will be covered. While the specification of respiratory protection should normally be made a result of personal and/or area sampling results, the use of respirator guides based on surface contamination monitoring results is also acceptable.

4.3.5 ALARA Guidelines

The total dose to an individual and the collective dose to the work force should be ALARA. When applied to personnel contamination or internal intakes, this generally means less-than-detectable dose with the best available commercial technology.

4.3.6 Release Criteria

The decision to release personnel with detectable uranium contamination is made on a case-by-case basis. If the individual is injured and needs prompt medical attention, medical treatment will always take precedence, with compensatory measures made for protecting medical personnel and facilities. If injuries are absent or do not require immediate attention, decontamination is preferable to ensure that the dose to the contaminated individual and the potential for inhalation by the victim and medical staff are minimized and the spread of contamination is prevented.

In a case where decontamination is incomplete due to injury to the skin or other reasons, the individual may be provisionally released with measures to prevent the spread of contamination.

4.4 DECONTAMINATION AND DECOMMISSIONING TECHNIQUES

This section concentrates on decontamination techniques to be used in the final decommissioning of a uranium-contaminated facility for unrestricted release. Some of these techniques are similar to those used during routine operations (e.g., personnel decontamination and some equipment and building surface decontamination). Contamination detection methods are similar for routine and D&D operations.

4.4.1 Personnel Decontamination

Skin decontamination should be performed by health physics technicians or other members of the health physics staff. The treatment and decontamination of wounds should be performed by medical staff.

Non-abrasive methods should be used for skin decontamination to protect the tissues from deeper contamination. Masking tape should be used to remove dry contamination. Wet decontamination should be used to remove residual contamination. The skin should be gently scrubbed with soap and water. Household bleach may be applied as needed to decontaminate more effectively. The following procedure is recommended:

a. Survey the worker to determine the contaminated areas of the skin. Have the medical staff treat and decontaminate breaks in the skin.

b. Wipe loose contamination with a gauze sponge or cotton applicators dipped in mild antiseptic detergent. Do not spread contamination to uncontaminated areas.

c. Rub the skin with the applicators to produce good sudsing.

d. Use soft bristle scrub brushes for fingernails and other difficult-to-clean areas as long as the skin barrier is maintained intact. It may be difficult to decontaminate the cuticles and under the nails.

e. Dry the skin area with cleansing tissue.

f. After the skin is thoroughly dry, survey it for any remaining contamination.

g. If no contamination is detected, apply a good-quality hand cream to prevent chapping.

Another effective non-abrasive decontamination method involves placing the contaminated hand in a cotton glove and then a Latex glove (causing the hand to perspire).

The decontamination factor is the ratio of the initial contamination level to the contamination level after decontamination methods are applied, as determined by survey instrument readings. Non-abrasive methods should be repeated until the decontamination factor between washes drops below 2 or 3 with significant contamination still remaining.

If contamination persists on the skin, a more abrasive decontamination method may be necessary. The decision to proceed with a more abrasive method should be based on the effectiveness of the decontamination. An abrasive soap should be applied with a moist gauze sponge or soft brush while rubbing the skin to develop a soapy lather. Care should be exercised to prevent damage to the skin surface. If contamination persists after using the abrasive soap, potassium permanganate ($KMnO_4$) and sodium bisulfite should be considered. Paint the contaminated skin with $KMnO_4$ using cotton-tipped applicators, allow the solution to dry, and paint it again two or three more times, allowing the solution to dry thoroughly between each application. The skin will then appear almost black. Applicators should be discarded after each use to avoid spreading contamination to the solutions. Then, rub the treated area with $KMnO_4$ using cotton applicators, until the brown discoloration is removed. Rinse the skin with water to remove the remaining $KMnO_4$, and dry the area thoroughly and survey it for contamination.

If contamination persists after all the above decontamination efforts, wrap the contaminated area to control the contamination and consult with medical personnel.

Liberal irrigation with room temperature water or saline solution (preferable) is recommended for eye, nose, and mouth contamination. These procedures are performed by the medical staff to remove contamination.

4.4.2 Equipment and Surface Decontamination

Decontamination of surface areas may be as simple as hosing off the floors with water, washing surfaces with detergent and water, or wiping with household dust cloths. Waste material generated from decontamination activities (e.g., water and wipe material) must be contained and disposed of as radioactive waste. For some locations, vacuuming the surfaces may be appropriate. If vacuuming is used, HEPA-filtered vacuum systems are required to keep airborne radioactive material out of the vacuum exhaust.

For some operations, periodic surface flushing with water may be adequate to maintain acceptable contamination levels. Precautions should ensure control and collection of run-off water so material may be recovered and waste water analyzed before discharge. Depending upon which isotope of uranium is involved, geometrically safe containers may be required for collecting and holding the liquid.

Depending upon the physical and chemical form of the uranium and the type of surface, uranium may become imbedded in the surface. Removal of embedded material may require physical abrasion,

such as scabbling, grinding, sand blasting, or chipping, or it may be accomplished using chemical etching techniques. If the surface is porous, complete replacement could be necessary. The use of high-pressure water (hydroblasting) has been quite successful for metal and concrete surfaces.

Ultrasonic cleaning techniques, electro-polishing, or chemical baths may be useful for decontamination of high-cost items if the chemicals used are compatible with the material to be cleaned.

A description of different decontamination techniques is found in DOE/EV/10128-1, DOE Decommissioning Handbook (DOE 1980), and publications by Allen (Allen 1985) and the Electric Power Research Institute (EPRI 1989). The DOE Decommissioning Handbook also includes guidance on decontamination techniques, assessment of environmental impacts, disposition of wastes, and preparation of decommissioning cost estimates.

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5.0 INTERNAL DOSIMETRY

Internal dosimetry is an essential part of a comprehensive radiological control program at every facility where uranium is handled or processed. The purpose of an internal dosimetry program is to monitor workplace activities, assess accidental or inadvertent intakes of radioactive material, and conduct internal dose assessments from bioassay measurement data.

DOE requires that facilities be designed, operated, and remediated to prevent intakes of radioactive materials. Radiological controls for the workplace should ensure that radionuclides are contained and handled properly, and that intakes are as low as reasonably achievable.

Experience has shown that the most common route for inadvertent uranium intake is inhalation. The uranium may be in natural, enriched, or depleted form, or a combination thereof. Intakes can also occur by accidental ingestion or by wound contamination. Surveillance programs should be designed to rapidly detect a release in the event of a loss of radioactive material containment. Internal dosimetry programs should be tailored to the needs of each uranium-handling facility so that inadvertent intakes are discovered and quantified and workers' dose equivalents are determined by appropriate methods.

When workers are inadvertently exposed to radioactive material, appropriate corrective action should be taken to ensure that control and containment are re-established. Prompt detection by routine workplace monitoring practices is essential to regaining control after any contamination spread or loss of containment. Prompt workplace indications of potential intake are also crucial to ensure timely initiation of special bioassay monitoring for intake and dose assessment. An early assessment of the probable severity of an intake and its corresponding dose, preferably within the first two hours of the intake, is needed for decisions on dose reduction therapy and event reporting. Uranium is both a radiological and chemical hazard. Because the total risk must be considered, both hazards must be considered. For uranium intakes, it may take many months to obtain the bioassay data necessary for final dose assessment. Until such data become available, ongoing preliminary assessments of intake and dose may be necessary to provide guidance for the administrative and medical management of the workers.

5.1 INTERNAL DOSE EVALUATION PROGRAM

Internal doses are not directly measured but are estimated or calculated based on knowledge of the material to which a worker may be exposed and its known or assumed biokinetic behavior. The common approach to internal dosimetry is to calculate an occupational intake based on worker bioassay measurements or workplace air-sample data and assumed breathing rates. Once an intake is calculated, appropriate dose equivalents to organs and tissues of concern can be estimated by using fundamental dosimetry principles, by various intake-to-dose conversion factors, which incorporate assumed biokinetic models, or by an appropriate computer code. Intake-to-dose conversion factors can be found in Federal Guidance Report No. 11 or ICRP Publication 30. Further discussion on intake and dose assessment is provided in Section 5.8.

Participation in internal dose evaluation programs (which include routine bioassay programs) is required for conditions identified in 10 CFR 835.402(c). The internal dose evaluation program must address both general workplace conditions and individual intakes. Workplace conditions are monitored

through air and surface contamination monitoring programs. Individual monitoring for intakes is commonly performed using bioassay procedures. Bioassay monitoring includes both direct (in vivo) measurements of radioactivity in the body and indirect (in vitro) measurements of material excreted or removed from the body.

10 CFR 835.402 requires participation in a bioassay program if a general employee is likely to exceed 0.1 rem committed effective dose equivalent (CEDE) from all intakes for all radionuclides in a year. Participation in a bioassay program is generally based on the possibility that a single intake causing a dose in excess of 0.1 rem CEDE might occur.

Indications of intake include (but are not limited to) detection of facial or nasal contamination, positive air monitoring or sampling results that may indicate internal exposure, or any wound in which contamination is detected or suspected. The most common internal exposure monitoring program for workers is the bioassay program, which must be designed for the specific nuclides and forms of material at a particular facility. Likely candidates for internal exposure monitoring include personnel who may be routinely exposed to surface or airborne contamination, or those identified by workplace indicators.

Workplace monitoring for potential internal exposures is performed to verify the adequacy of containment and work practices. This monitoring includes air sampling, continuous air monitoring, personal contamination surveys, and workplace contamination surveys. Facilities are to be designed and operated to minimize internal exposure. Details regarding workplace monitoring and control practices are discussed in Chapter 4, Contamination Control.

5.1.1 Performance Capabilities for Internal Exposure Monitoring

Bioassay programs must be capable of showing compliance with the 5-rem/year stochastic and 50-rem/year nonstochastic dose limits of 10 CFR 835.202. 10 CFR 835.402(c)(1) identifies 0.1 rem CEDE for all likely intakes as a level above which workers must participate in a bioassay program. Therefore, ideally, such bioassay monitoring programs should be capable of detecting individual doses at that level. To meet this requirement, reliance must be placed on workplace monitoring to identify potential intakes at the time they occur so that special bioassay monitoring can be initiated.

Performance capabilities for bioassay and internal dosimetry programs can be expressed as the minimum detectable dose, based on some combination of minimum detectable activity and frequency of measurement or time post-intake at which the measurement is made. The term "minimum detectable dose" is preferred over any variants of the occasionally encountered terms "dose-missed" or "potentially undetected dose," which were usually defined as the same thing. The connotation of the latter terms is that of an actual intake which was not detected, whereas the intent was to define a measure of program sensitivity to doses that might have gone undetected had an intake occurred. The preferred term "minimum detectable dose" (MDD) ties the concept to the recognized terminology of minimum detectable activity (MDA).

The MDD for a bioassay program must meet the aforementioned dose limit requirements of 10 CFR 835.202. A design goal of 0.1 rem CEDE from all intakes of similar nuclides in a year is desirable but unrealistic for a routine program. To meet these requirements, bioassay programs should have measurement sensitivities (i.e., MDAs for bioassay measurements) established based on the material to which workers might be exposed. Examples of such sensitivities are given in Tables 5-1 through 5-3 for pure ^{238}U monitored by urinalysis, fecal analysis, and lung counting, respectively. The bioassay goals

are calculated by multiplying the intake (nCi) by the intake retention fraction (IRF) and by a correction factor of 2,220 dpm/nCi, where intake is the dose limit divided by a calculated dose conversion factor (rem/nCi). For class D uranium, the dose limit goal is based on the 50-rem committed dose equivalent (CDE) for bone surfaces; the other dose limit goals are based on the 0.1-rem CEDE monitoring threshold. The dose conversion factors used for Tables 5-1 through 5-3 are given in Table 5-4.

There may be circumstances in which the measurement technology is not available to provide the sensitivities required for the 0.1 rem goal using routine, periodic measurements at reasonable frequencies. Therefore, because the goal of 0.1 rem CEDE cannot be met through routine bioassay, the radiation protection organization should take the following administrative actions:

- ensure that adequate control measures are applied to prevent intakes,
- document the adequate control measures for auditing purposes,
- upgrade bioassay measurement systems and workplace monitoring practices to provide state-of-the-art measurements, and
- ensure that internal dose assessments use state-of-the-art technology.

All confirmed occupational intakes of uranium, regardless of magnitude, should be assessed. The results of all bioassay and other measurements needed to demonstrate the quality of measurements and dose assessment should be recorded and maintained. The recording and reporting requirements for internal dosimetry data are set forth in Section 3.7 of this technical standard

Table 5-1. Urine Intake Retention Fractions (IRF) and Bioassay Goals for ^{238}U Inhalation of 1-μm AMAD particles[a]

Days After Intake	Class D					Class W			Class Y		
	Urine IRF			Dose Limit Goal[e] (dpm)	100-mrem CEDE Goal[f] (dpm)	Urine IRF[d]	Dose Limit Goal[g] (dpm)	100-mrem CEDE Goal[h] (dpm)	Urine IRF[d]	Dose Limit Goal[i] (dpm)	100-mrem CEDE Goal[j] (dpm)
	Wrenn[b]	Fisher[c]	ICRP[d]								
1	1.2E-01	7.5E-02	1.0E-01	1.6E+05	5.7E+03	1.6E-02	2.5E+04	5.0E+02	8.0E-04	7.4E+01	1.5E+00
2	3.7E-02	1.9E-02	4.1E-02	4.1E+04	1.5E+03	5.5E-03	8.6E+03	1.7E+02	2.7E-04	2.4E+01	5.0E-03
7	2.1E-03	2.9E-03	9.4E-03	6.2E+03	2.2E+02	2.1E-03	3.3E+03	6.6E+01	1.0E-04	9.2E+00	1.8E-01
14	1.6E-03	1.6E-03	5.1E-03	3.4E+03	1.2E+02	1.4E-03	2.2E+03	4.4E+01	6.3E-05	5.8E+00	1.2E-01
30	9.3E-04	6.2E-04	1.7E-03	1.3E+03	4.7E+01	7.2E-04	1.2E+03	2.2E+01	3.2E-05	3.0E+00	5.9E-02
60	4.2E-04	3.0E-04	4.5E-04	6.4E+02	2.3E+01	4.1E-04	6.4E+02	1.3E+01	2.1E-05	1.9E+00	3.9E-02
90	2.4E-04	2.0E-04	1.5E-04	4.3E+02	1.5E+01	2.7E-04	4.2E+02	8.4E+00	1.9E-05	1.8E+00	3.5E-02
180	1.2E-04	1.2E-04	8.5E-06	2.6E+02	9.2E+00	9.4E-05	1.5E+02	2.9E+00	1.8E-05	1.7E+00	3.3E-02
365	7.6E-05	7.6E-05	1.7E-06	1.6E+02	5.8E+00	1.1E-05	1.7E+01	3.4E-01	1.8E-05	1.7E+00	3.3E-02
730	3.9E-05	3.9E-05	1.6E-06	8.3E+01	3.0E+00	5.8E-07	9.1E-01	1.8E-02	1.7E-05	1.6E+00	3.1E-02
1825	6.1E-06	6.1E-06	1.3E-06	1.3E+01	4.7E-01	3.9E-07	6.1E-01	1.2E-02	1.1E-05	1.0E+00	2.0E-02
3650	1.3E-06	1.3E-06	9.7E-07	2.8E+00	1.09E-01	2.9E-07	4.5E-01	9.1E-03	3.6E-06	3.3E+01	6.7E-03
7300	6.2E-07	6.2E-07	5.7E-07	1.3E+00	4.7E-02	1.7E-07	2.7E-01	5.3E-03	3.7E-07	3.4E-02	6.8E-03
1825	8.5E-08	8.5E-08	1.2E-07	1.8E-01	6.5E-03	3.7E-08	5.8E-02	1.2E-03	1.7E-08	1.6E-03	3.1E-05

(a) Suitable for elemental or any long half-life uranium isotope. Goals reflect activity in 24-hour urine void corresponding to either 50-rem CDE or 0.1-rem CEDE.

(b) Wrenn-Lipsztein Model for Soluble Uranium (Wrenn et al., 1989).

(c) Fisher Modified Wrenn-Lipsztein Uranium Model, which is based on UF_6 data (Fisher et al., 1991).

(d) ICRP Publication 30 General Systemic Model (ICRP 1988a).

(e) Based on Fisher Modified Wrenn-Lipsztein Uranium Model and 962 nCi intake.

(f) Based on Fisher Modified Wrenn-Lipsztein Uranium Model and 34.5 nCi intake.

(g) Based on ICRP Model and 704 nCi intake.

(h) Based on ICRP Model and 14.1 nCi intake.

(i) Based on ICRP Model and 41.7 nCi intake.

(j) Based on ICRP Model and 0.833 nCi intake.

Table 5-2. Feces Intake Retention Fractions (IRF) and Bioassay Goals for ^{238}U Inhalation of 1-μm AMAD particles[a]

Days After Intake	Class D					Class W			Class Y		
	Feces IRF			Dose Limit Goal[e] (dpm)	100-mrem CEDE Goal[f] (dpm)	Feces IRF[d]	Dose Limit Goal[g] (dpm)	100-mrem CEDE Goal[h] (dpm)	Feces IRF[d]	Dose Limit Goal[i] (dpm)	100-mrem CEDE Goal[j] (dpm)
	Wrenn[b]	Fisher[c]	ICRP[d]								
1	6.8E-02	6.8E-02	6.8E-02	1.5E+05	5.2E+03	1.0E-01	1.6E+05	3.1E+03	1.3E-01	1.2E+04	2.4E+02
2	4.2E-02	4.2E-02	4.2E-02	9.0E+04	3.2E+03	1.3E-01	2.0E+05	4.1E+03	1.6E-01	1.5E+04	3.0E+02
7	3.7E-04	3.7E-04	3.7E-04	7.9E+02	2.8E+01	5.9E-03	9.2E+03	1.9E+02	5.4E-03	5.0E+02	1.0E+01
14	3.4E-07	3.4E-07	3.4E-07	7.3E-01	2.6E-02	1.1E-03	1.7E+03	3.4E+01	1.7E-04	1.6E+01	3.1E-01
30						8.9E-04	1.4E+03	2.8E+01	1.3E-04	1.2E+01	2.4E-01
60						5.9E-04	9.2E+02	1.9E+01	1.3E-04	1.2E+01	2.4E-01
90						3.9E-04	6.1E+02	1.2E+01	1.2E-04	1.1E+01	2.2E-01
180						1.1E-04	1.7E+02	3.4E+00	1.1E-04	1.0E+01	2.0E-01
365						8.6E-06	1.3E+01	2.7E-01	8.4E-05	7.8E+00	1.6E-01
730						5.5E-08	8.6E-02	1.7E-03	5.0E-05	4.6E+00	9.3E-02
1825									1.1E-05	1.0E+00	2.0E-02
3650									8.8E-07	8.2E-02	1.6E-03

(a) Suitable for elemental or any long half-life uranium isotope. Goals reflect activity in 24-hour fecal samples corresponding to either 50-rem CDE or 0.1-rem CEDE. A cutoff point of 1 E-08 was used for listing IRFs and corresponding dose goals were not determined for values below the cutoff point.

(b) Wrenn-Lipsztein Model for Soluble Uranium (Wrenn et al., 1989)

(c) Fisher Modified Wrenn-Lipsztein Uranium Model, which is based on UF_6 data (Fisher et al., 1991).

(d) ICRP Publication 30 General Systemic Model (ICRP 1988a).

(e) based on Fisher Modified Wrenn-Lipsztein Uranium Model and 962 nCi intake

(f) based on Fisher Modified Wrenn-Lipsztein Uranium Model and 34.5 nCi intake

(g) based on ICRP Model and 704 nCi intake

(h) based on ICRP Model and 14.1 nCi intake

(i) based on ICRP Model and 41.7 nCi intake

(j) based on ICRP Model and 0.833 nCi intake

Table 5-3. Lung Intake Retention Fractions (IRF) and Bioassay Goals for ^{238}U Inhalation of 1-μm AMAD particles[a]

Days After Intake	Class D					Class W			Class Y		
	Lung IRF			Dose Limit Goal[e] (nCi)	100-mrem CEDE Goal[f] (nCi)	Lung IRF[d]	Dose Limit Goal[g] (nCi)	100-mrem CEDE Goal[h] (nCi)	Lung IRF[d]	Dose Limit Goal[i] (nCi)	100-mrem CEDE Goal[j] (nCi)
	Wrenn[b]	Fisher[c]	ICRP[d]								
1	8.0E-02	3.0E-02	8.0E-02	2.9E+01	1.0E+00	2.1E-01	1.5E+02	3.0E+00	2.1E-01	8.8E+00	1.8E-01
2	2.4E-02	1.2E-02	2.4E-02	1.2E+01	4.1E-01	1.8E-01	1.3E+02	2.5E+00	1.8E-01	7.5E+00	1.5E-01
7	4.5E-05	3.3E-05	4.5E-05	3.2E-02	1.1E-03	1.4E-01	9.9E+01	2.0E+00	1.5E-01	6.3E+00	1.3E-01
14						1.3E-01	9.2E+01	1.8E+00	1.5E-01	6.3E+00	1.3-01
30						1.0E-01	7.0E+01	1.4E+00	1.5E-01	6.3E+00	1.3E-01
60						7.0E-02	4.9E+01	9.9E-01	1.4E-01	5.8E+00	1.2E-01
90						4.8E-02	3.4E+01	6.8E-01	1.4E-01	5.8E+00	1.2E-01
180						1.5E-02	1.1E+01	2.1E-01	1.2E-01	5.0E+00	1.0E-01
365						1.4E-03	9.9E-01	2.0E-02	1.0E-01	4.2E+00	8.3E-02
730						1.1E-05	7.7E-03	1.6E-04	7.3E-02	3.0E+00	6.1E-02
1825									2.9E-02	1.2E+00	2.4E-02
3650									9.6E-03	4.0E-01	8.0E-03
7300									4.2E-03	1.8E-01	3.5E-03
18250									3.8E-03	1.6E-01	3.2E-03

(a) Suitable for elemental or any long half-life uranium isotope. Goals reflect activity in lungs corresponding to 50-rem CDE or 0.1-rem CEDE. A cutoff point of 1 E-08 was used for listing IRFs and corresponding dose goals were not determined for values below the cutoff point.

(b) Wrenn-Lipsztein Model for Soluble Uranium (Wrenn et al., 1989).

(c) Fisher Modified Wrenn-Lipsztein Uranium Model, which is based on UF_6 data (Fisher et al., 1991).

(d) ICRP Publication 30 General Systemic Model (ICRP 1988a).

(e) based on Fisher Modified Wrenn-Lipsztein U Model and 962 nCi intake

(f) based on Fisher Modified Wrenn-Lipsztein U Model and 34.5 nCi intake

(g) based on ICRP Model and 704 nCi intake

(h) based on ICRP Model and 14.1 nCi intake

(i) based on ICRP Model and 41.7 nCi intake

(j) based on ICRP Model and 0.833 nCi intake

Table 5-4. Dose Conversion Factors for ^{238}U [a]

Solubility Class	Model	Dose Conversion Factor (rem/nCi)	
		CEDE	Bone Surfaces
D	Fisher Modified Wrenn-Lipsztein	2.9E-3	5.2E-2
W	ICRP	7.1E-3	---
Y	ICRP	1.2E-1	---

(a) Factors calculated using CINDY Version 1.4 (Strenge et al. 1992).

Radiation exposure records programs must also provide for the summation of internal and external doses, as required by 10 CFR 835.702. While the summation process is not necessarily performed under a site internal dosimetry program, it is recommended that the program coordinator recognize what is required. The following summations are identified by 10 CFR 835.702(c)(5):

- total effective dose equivalent (TEDE) defined as the summation of effective dose equivalent (deep dose equivalent) from external exposure and the CEDE,
- summation of the effective dose equivalent (deep dose equivalent) from external exposure and the CDE to organs or tissues of concern,
- cumulative TEDE received from external and internal sources while employed at the site or facility, since January 1, 1989, and
- for the embryo/fetus of a declared pregnant worker, the summation of the deep dose equivalent to the mother from external exposure during the entire gestation period and the gestation period dose equivalent to the embryo/fetus from intakes by the mother during the entire gestation period.

Doses should be calculated and recorded for any confirmed uranium intake. What constitutes a confirmed intake is discussed in Section 5.7. Along with the doses, supporting records must be maintained, including the bioassay data, assumptions, biokinetic models, and calculational methods used to estimate the doses. These may be included in letter-report dose assessments, databases, technical basis documents, and similar records, either singly or in combination.

5.1.2 Protection of the Embryo/Fetus, Minors, and Members of the Public

The TEDE limit for the embryo/fetus of a declared pregnant worker is 0.5 rem for the entire gestation period, defined as the summation of external dose received and internal dose received during the gestation period (not the 50-year committed internal dose). Internal exposure monitoring is required if an intake is likely to result in more than 10% of that limit (i.e., 50 mrem for the gestation period). Providing adequate protection to keep the mother's intakes below the occupational limits will also provide adequate protection for the embryo/fetus. Thus, special bioassay for uranium during pregnancy

is not required. As a matter of caution, some sites try to obtain baseline bioassays as soon as a pregnancy is declared, with another baseline bioassay following the end of pregnancy. Some sites also offer to restrict pregnant workers from jobs with relatively high potential for occupational intakes.

Minors and members of the public are limited by 10 CFR 835.207 and 10 CFR 835.208 to a TEDE of 0.1 rem/year. Minors are also limited to 10% of the occupational dose limits of 10 CFR 835.202(a)(3) and (a)(4). Internal exposure monitoring is required if an intake is likely to result in 50% of that limit (0.05 rem) from all radionuclide intakes in a year. Because bioassay monitoring is not likely to be sufficiently sensitive to identify such intakes on a routine basis, enhanced workplace surveillance or restriction of access may be required.

5.2 CHARACTERIZATION OF INTERNAL HAZARDS

Monitoring for uranium poses special problems for the following reasons.

- Uranium presents both chemical and radiological toxicity risks, the relative importance of which depends on its transportability from the lung.
- Uranium usually exists in mixed transportability classes.
- Small, recent intakes easily mask larger, older intakes because nearly 50% of the uranium going to blood is cleared immediately through the urine.
- An intake of class Y material potentially resulting in a CEDE of 0.1 rem generally cannot be detected by routine bioassay monitoring. Monitoring of the workplace to document the working environment and to provide immediate indication of an intake is essential.
- Low-level chronic intakes are common, so the bioassay program must monitor for long-term buildup as well as for potentially significant acute intakes.
- Individual and temporal variability in the environmental background of uranium complicates interpretation of urinalysis results.

Consequently, the proper bioassay monitoring program for uranium workers is best determined on a case-by-case basis in consultation with an internal dosimetry specialist. As part of the program technical basis, the uranium mixtures need to be determined. In addition, determinations should be made at the time of identified incidents of potential intake. Methods for such determination may include radiochemical analysis or chemistry followed by mass spectrometry.

Solubility is of major importance in uranium inhalation toxicology. Soluble uranium compounds are absorbed and rapidly transported to kidney and bone, or excreted in urine. Because uranium damages kidney tissues by the same mechanisms as other heavy metals, dissolved uranium is considered to be a chemical toxicant. Dissolved uranium also deposits in bone and is retained for long periods of time, such that sufficiently enriched uranium can deliver an accumulated radiation dose sufficient to be considered a radiological hazard to bone (Morrow 1986).

Oxides of uranium tend to exhibit inhalation class Y behavior, slightly more soluble compounds are assigned to class W, and soluble compounds are assigned to class D. Note that some compounds that

have been classified as class Y have shown a more rapid clearance from the lung than for other class Y compounds, i.e., having a 100-day effective half-life in the lung compared to the class Y compounds that have a 500-day effective half-life. This may be due to the existence of mixtures having more than one physicochemical form (ICRP 1988b; Fisher et al. 1990). A report (Forrest and Barber 1993) on class "Q" material has behavior similar to that of special class Y. Class Q material was found to consist of 8-µm size particles (instead of the 1-µm size assumed under ICRP Publication 30 methodology and to consist of two parts: 90% class W material with a 120-day effective half-life in the lung and 10% class Y material with a 500-day effective half-life in the lung. It should be cautioned that even if special class Y or class Q material is suspected, the relative transportability of the material should be determined and documented before establishing action levels (ANSI/HPS 1995). As uranium ages in a residual, loose contamination form, such as might be found in old duct work, glove boxes, or other such components, it can be expected to undergo slow oxidation to a more insoluble form. Thus, class Y forms of uranium may be reasonable assumptions of what to expect during many decommissioning operations.

For depleted uranium to present a chemical toxic hazard from inhalation, the depleted uranium would have to be subdivided into soluble particles that can be inhaled, transported into the lungs, and transferred to the blood for transport to the kidneys. Depleted uranium metal is not readily subdivided into small, respirable particles. However, depleted uranium metal can slowly oxidize under ambient environmental conditions (corrosion), resulting in formation of small particles. The rate of oxidation will vary with the amount of water vapor present and the temperature. The oxidation rate will, in turn, influence the solubility of the material inhaled.

Following an accidental release from a nuclear reactor, fission and activation products may be present in fragments of irradiated fuel, of which the matrix is predominately uranium oxide (Devell 1988; Begichev et al. 1989; Toivonen et al. 1992). Studies of the in vitro dissolution of particles released from the Chernobyl accident, seven out of ten of which consisted mainly of uranium (Cuddihy et al. 1989), were consistent in assigning all the gamma-emitting radionuclides to class W (ICRP 1996).

Particle size is an important consideration for inhalation exposures. The normal practice for an aerosol is to identify the activity median aerodynamic diameter (AMAD) and its associated particle-size distribution. Particle sizes of 10 µm or less are considered respirable. For compliance with 10 CFR 835, the common practice is to assume a 1-µm particle size for dosimetry purposes unless actual particle size information is available. Particle size data are most readily obtainable for chronic exposure situations. Unless representative air sampling is performed in the immediate proximity of a worker during abnormal working conditions, the practical likelihood of obtaining good particle-size information is slim.

5.3 SCOPE OF BIOASSAY PROGRAM

For classes D and W uranium compounds, the monitoring programs need to be designed to maintain exposures, including those from single acute intakes, below levels that will cause transient kidney damage due to the heavy metal toxicity of uranium. Typically, urine sampling is the preferred method of monitoring for classes D and W uranium. For class Y natural uranium and all classes of highly enriched uranium, radiological considerations are more limiting. In addition, local factors concerning the diversity of chemical forms of uranium must be taken into account when designing a bioassay monitoring program. For these materials, a combination of direct and indirect monitoring may be required.

5.3.1 Classification of Bioassay Measurements

Bioassay measurements can be classified according to the primary reason for their performance. This is a useful practice for historically documenting why a worker participated in a bioassay program. Numerous reasons for bioassay measurements may be defined for specific facilities; some suggested common classifications are as follows:

- **Baseline measurements** are used to establish a pre-exposure condition, either for a new employee or as a result of a new work assignment. The RCS recommends baseline measurements if workers are considered likely to receive intakes resulting in greater than 100-mrem CEDE. It is a good practice to perform such measurements for newly hired employees, intra-company transferees, or workers transferred from facilities where bioassay measurements may not have been required. In addition, baseline measurements can verify workers' status for special work assignments. For uranium bioassay, baseline measurements made before any occupational exposure can be expected to yield no detectable results using current technology.

  A special consideration is the evaluation of intakes that include natural materials such as uranium. The sensitivity of urine sampling as a uranium bioassay tool is limited by the presence of environmental levels of uranium, which is subject to some uncertainty in interpretation. Knowledge of background level of uranium excretion is an important factor in analysis and interpretation of urine or feces for uranium bioassay purposes. In ICRP Publication 23, model values for excretion of uranium by Reference Man are given as 0.05 to 0.5 µg/day in urine and 1.4 to 1.8 µg/day in feces. 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). 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) using inductively coupled plasma mass spectrometry (ICP/MS) to look for the presence of ^{236}U. Since the ^{236}U isotope does not occur in nature, it is used as a flag to indicate occupational exposure.

  Exempting workers from baseline bioassay implies accepting any detectable results as likely attributable to current occupational exposure. However, requiring baseline measurements can potentially impact the schedule of short-term jobs; the time required to obtain a chest count and a large-volume urine sample may add a day or two delay to entry procedures. Moreover, missing a baseline for a long-term employee who will be placed on a routine bioassay program is not likely to be as troublesome as not obtaining a baseline for a short-term worker who provides a termination sample that shows detection of uranium after the worker has left the site and is difficult to reach for follow-up.

- **Routine, or periodic, measurements** are performed on a predetermined schedule (e.g., an annual or quarterly frequency).

- **Special bioassay measurements** are performed as follow-up to unusual routine results or suspected intakes.

- **End of assignment or termination measurements** are performed following completion of specific work or at the time of termination of employment. The RCS (DOE 1999) recommends that workers who participate in bioassay programs have appropriate termination measurements.

Bioassay classification is important because the purpose of a sample may affect the collection and analysis or monitoring method chosen. For example, single-void urine samples are not adequate for routine monitoring of potential uranium exposure, but can provide important information for dose-reduction therapy following a suspected intake; samples representative of excretion over a 24-hour period should be collected for quantitative intake and dose assessment. The date of sample collection (and possibly the time of collection) can be very important to special monitoring performed to assess intake. However, these are much less important with regard to periodic monitoring, for which measurements are not expected to show detectable activity and when any detection whatsoever is likely to initiate investigation and special bioassay.

5.3.2 Monitoring Requirements and Selection of Employees

Workers who are considered likely to have intakes resulting in excess of 0.1 rem CEDE are required to participate in a bioassay program. The workers at highest risk of incurring an intake are the ones in closest contact with the material. Typically, these are the operators, maintenance, and radiological control personnel handling uranium or uranium-contaminated objects in the course of routine glove-box, maintenance, or decommissioning operations. In the event of containment system failure, it is these workers who will most likely incur exposure and subsequent intake. These workers should be on a routine bioassay program designed to meet the requirements of 10 CFR 835 as a kind of safety net to identify intakes which might have gone undetected by workplace monitoring.

Other workers (e.g., supervisors, inspectors, observers, guards, and tour groups) who work in or visit a uranium facility but are not directly working with the material or contaminated objects are normally at a substantially lower risk for incurring an intake. Although these people may not need to be on a routine bioassay program, they should be subject to participation in a special bioassay program if workplace indications suggest loss of control or containment.

Routine bioassay monitoring should be implemented whenever quantities of uranium handled exceed the respective quantities in Table 5-5.

Table 5-5. Minimum Uranium Bioassay Monitoring[a,b]

TYPES OF OPERATION	MASS	ACTIVITY AMOUNT[c]
Processes in open room on bench top, with possible escape from process vessels	0.5 kg[d]	320 µCi
Process with possible escape of uranium that are carried out within a fume hood of adequate design, face velocity, and performance reliability	5 kg	3,200 µCi
Process carried out within gloveboxes that are ordinarily closed, but with possible release from process vessels and occasional exposure to contaminated box and leakage	50 kg	32,000 µCi

(a) From ANSI/HPS 1995.

(b) Values chosen as conservative for any transportability class or mixture of isotopes of uranium. For a particular type of operation, the value of mass or activity that is more restrictive for the mixture should be used.

(c) Obtained from DAC values for pure ^{235}U (see Appendix A.2 of ANSI/HPS 1995).

(d) From ANSI/HPS 1995, Appendix A.1.

5.3.3 Selection of Bioassay Monitoring Techniques

Bioassay monitoring techniques fall into two broad categories: direct measurement of radioactive materials in the body (in vivo counting) and analysis of material removed from the body for laboratory (in vitro analysis). In vivo counting includes measurements of the chest, lung, skeleton, liver, and wounds. In vitro measurements include urinalysis, fecal analysis, and occasionally analysis of tissue, sputum, or blood samples. Methods for in vitro analysis include liquid scintillation counting, fluorescence measurements, gamma spectrometry, chemical separation followed by electrodeposition, and counting with radiation detectors. A brief overview of bioassay techniques and capabilities has been developed (Selby et al. 1994). Further discussion of the techniques is provided below.

In addition, to ensure that adverse chemical toxicity effects are unlikely, bioassays for uranium should be performed when intakes of 1 mg or more of soluble uranium are likely to occur in any one work day (ANSI/HPS 1995).

5.3.3.1 In Vivo Counting

Direct bioassay (in vivo counting) is the measurement of radiations emitted from radioactive material taken into and deposited in the body. Direct bioassay is appropriate for detection and measurement of photons emitted by uranium and its decay products. Lung, wound, and skeleton counting are examples of in vivo monitoring most commonly used for uranium and its progeny.

When direct bioassay is used, the detection system should be calibrated for the radionuclides to be measured in the appropriate organs. All calibration procedures, calibration records, and quality control data should be maintained.

A uranium facility should have the capability to detect and assess depositions of uranium in the lungs of affected workers. The major objective of lung counting is to provide measurements of suspected intakes triggered by workplace monitoring results. Lung measurements should be made to provide an early estimate of the magnitude of the intake and resulting lung deposition.

The most widely used systems for lung counting are high-purity germanium detectors, thin sodium-iodide detectors, phoswich detectors, and proportional counters. Multiple high-purity germanium detectors have advantages over the other detector systems because of their good resolution, allowing better identification of the radionuclide, better detectability, and better background prediction capability. The main disadvantages of germanium detector arrays are their higher cost relative to other types of in vivo detectors and their lower reliability. Germanium detectors also must be continuously cooled with liquid nitrogen.

For natural and enriched uranium, the energy most commonly used for in vivo monitoring is the 185-keV gamma that is emitted with 54% abundance from the decay of ^{235}U (ANSI/HPS 1995; Gerber and Thomas 1992). For natural uranium, approximately 50% of the activity is due to decay of ^{234}U. For enriched uranium, ^{234}U is the major contributor to total activity. Thus, one must be aware that in vivo monitoring for uranium is based on detection and measurement of a uranium isotope that contributes very little to the dose (ANSI/HPS 1995). To calculate dose, one needs to know the total uranium activity and the isotopic distribution of the material.

For natural or depleted uranium, detection of the 92.4-keV and 92.8-keV K x-rays emitted by the ^{234}Th daughter product are most commonly used (ANSI/HPS 1995; Gerber and Thomas 1992). This monitoring method would not be appropriate for freshly separated uranium as the ^{234}Th will not be in equilibrium with the ^{238}U and would potentially result in an underestimate or overestimate of the actual internal burden.

Measurement equipment to detect and measure uranium contamination in wounds should be available at all uranium facilities. Instrumentation used for this purpose may include thin-crystal NaI(Tl), intrinsic germanium, or Si(Li) detectors. Correction for depth due to absorption of photons in the overlying tissues should be considered. Collimated detectors are useful for determining the location of the uranium in wounds.

Estimates of the depth of uranium contamination in a wound may be made using solid-state germanium or Si(Li) detectors to measure the relative absorption of the low-energy x-rays emitted by uranium. Information about depth is important for determining whether tissue excision is necessary to remove the contamination.

5.3.3.2 In Vitro Analysis

The two most common forms of in vitro analysis are urinalysis and fecal analysis.

Urinalysis. Urine sampling provides useful information about the amount of uranium excreted following an intake. After chemical isolation, the uranium in urine samples may be determined by alpha

spectrometry (gas-flow proportional or surface-barrier detection), alpha counting (zinc sulfide or liquid scintillation counting), or track counting. Analytical procedures for in vitro measurement of uranium and other radionuclides have been published (Volchok and dePlanque 1983; Gautier 1983).

Urine samples should be collected away from the uranium facility to minimize cross-contamination. Samples should be collected in contamination-free containers; measures should be considered for minimizing plateout on walls of container surfaces (such as by addition of trace amounts of gold, oxalate, or nitric acid).

Fecal Analysis. Fecal analysis is a useful procedure for evaluating the excretion of uranium and many other radioactive materials because more than half of the material deposited in the upper respiratory tract is cleared rapidly to the stomach and gastrointestinal (GI) tract.

The total fecal plus urinary elimination for the first few days after exposure, combined with in vivo counts that might be obtained, may provide the earliest and most accurate assessment of intake. Fecal samples taken during the second and third day after an inhalation incident are likely to provide the most useful data because the gastrointestinal hold-up time may vary from a few hours to a few days.

Fecal sampling is primarily a monitoring procedure for confirming and evaluating suspected intakes, but is used at some uranium facilities for routine periodic monitoring as well. Workers may find fecal sampling unpleasant or objectionable, and laboratory technicians may also have aversion to fecal sample analysis. Some of these problems may be minimized if commercial fecal sample collection kits are used for convenient collection and handling of samples. Collection kits also provide a means for collecting uncontaminated samples. Fecal samples may require additional sample preparation before analysis.

5.4 ESTABLISHING BIOASSAY FREQUENCY

The bioassay measurement frequency should be based on: 1) the potential risks of an intake occurring; and 2) the sensitivity of a bioassay program to detecting potential intakes. The bioassay program sensitivity can be selected using specified intervals between measurements based on the MDD associated with an interval.

The rationale for the selected bioassay measurement frequency should also be documented. It is appropriate to evaluate the probability of intake and to modify the sampling frequency based on that probability.

The frequency of bioassay measurements should normally not be decreased because analytical results are below the detection level. The bioassay program should be maintained to confirm the proper functioning of the overall internal exposure control program and to document the absence of significant intakes of radionuclides.

5.4.1 Frequency Based on Program Sensitivity

The minimum detectable dose concept refers to the potential dose associated with an MDA bioassay measurement at a given time interval post-intake. The pattern of retention of activity in the body, the MDA for a bioassay measurement technique, and the frequency with which that technique is applied define a quantity of intake that could go undetected by the bioassay program. An intake of such a

magnitude would not be detected if it occurred immediately after a bioassay measurement and if it were eliminated from the body at such a rate that nothing was detected during the next scheduled measurement. The dose resulting from such an intake would be the MDD for that particular measurement technique and frequency.

Estimates of MDD in terms of CEDE should be documented for each measurement technique, Minimum Detectable Activity (MDA), and frequency. The MDA is defined in ANSI/HPS N13.30 (ICRP Publication 1996) as a measure of the detection limit. Analytical radiobioassay laboratories should meet the Acceptable MDAs (AMDAs) recommended in ANSI N13.30 as a minimum. The AMDAs for U bioassay are shown in Table 5-6.

Table 5-6. Categories and Performance Criteria for Uranium Bioassay

Direct Bioassay

CATEGORY	ORGAN	AMDA[(a)]
Measurement of ^{234}Th	Lung	3 nCi*
Measurement of ^{235}U	Lung	0.2 nCi

* Based on 10 mg ^{238}U.

Indirect Bioassay

CATEGORY	NUCLIDE	AMDA[(a)]
Alpha (Urine)	^{234}U, ^{235}U, ^{238}U	0.1 pCi/L
Isotope specific measurements		
Mass determination	Uranium (natural)	5 µg/L

(a) Note: The "Acceptable MDAs (AMDAs)" were removed from later drafts of the ANSI standard due to possible misinterpretation of the word "acceptable". The AMDAs have been replaced with test ranges for externally conducted quality control tests that take into consideration the need to be several times MDA or more before reasonably low coefficients of variation can be obtained for individual sample measurements. In this way, bias as well as precision can be estimated from reasonably small samples within each test category.

Retention functions specific to the various chemical forms and particle size distributions found in the facility should be used. Examples of MDD tabulations can be found in La Bone et al. (1993) and Carbaugh et al. (1994). In establishing MDD tables, it is important to consider dose contributions from all appropriate radionuclides in any mixture, rather than just the dose contribution from the bioassay indicator nuclide.

The minimum frequency for routine bioassay programs is interrelated to action levels, as specified in Table 5-7 (ANSI/HPS 1995). Special bioassays are taken as needed.

Table 5-7. Minimum Suggested Frequencies for Routine Bioassay for Uranium[a]

SOLUBILITY CLASS	SITUATION	FREQUENCY		
		URINE	FECAL	IN VIVO
	Radiological			
D		Monthly	(b)	(b)
W		Quarterly		Annually
Special Y		Quarterly		Annually
Y		(b)		Annually
	Chemical Toxicity			
D and W		Monthly		Annually[c]

(a) From ICRP Publication , 1995.
(b) The method of analysis not usually used.
(c) For Class W.

5.4.2 Frequency Based on Potential Risk of Intake

Although uranium workers are not generally considered to be at high risk of incurring intakes that might result in CEDEs of 0.1 rem or more, any uranium worker can be considered to have the potential for such an intake (see Section 5.3.2). However, having the potential for intake does not mean that they are likely to incur an intake.

Workers who have the highest potential risk for an intake are those most closely working with uranium or uranium-contaminated material. Typically, these workers are glove-box workers, maintenance workers, and operational radiological control surveillance staff. These workers should be on a routine uranium bioassay program, including urinalysis and in vivo measurements. Such programs are relatively insensitive compared to the 0.1 rem CEDE monitoring threshold and are a safety net intended to catch intakes of significance relative to regulatory limits, rather than substantially lower administrative levels. Selection of bioassay frequency depends on the facility experience with potential intakes, the perceived likelihood of intake, and the MDD of a program. Annual urinalyses and in vivo chest counts are fairly typical. More frequent (e.g., semi-annual or quarterly) measurements may permit more timely review of workplace indicators in the event that an abnormal bioassay result is obtained, but do not necessarily mean a more sensitive program.

5.4.3 Special Bioassay as Supplements to Routine Bioassay Programs

Special bioassay programs for workers with known or suspected acute inhalation intakes of uranium or other alpha-emitting radionuclides should include both urine and fecal sampling. Special bioassay measurements should be initiated for each employee in a contaminated work area when surface contamination is detected by routine surveillance if it is possible that the contamination resulted in a CEDE of 0.1 rem or greater. Excreta samples should not be collected where they may be contaminated by external sources of uranium. Ideally, total urine and feces should be collected for about a week

following intake. This permits a sensitive assessment of potential intake and internal dose. Longer-term special samples collected at various times from a month to a year following intake can help to discriminate between ingestion, class W inhalation, and class Y inhalation.

5.4.4 Long-term Follow-up Bioassay Programs

Following an intake, a long-term follow-up bioassay program may be required for a worker to compare the actual excreta or in vivo results with those projected by the evaluation. This is important to verify the accuracy of intake and dose assessments. The frequency and duration of a special program is dependent upon the projected values; it is suggested that as long as a worker continues to have detectable bioassay results, he or she should continue to be monitored. It is particularly important to have good baseline data and projections for individuals who return to uranium work.

The ability of a bioassay program to distinguish between an established, elevated baseline and a new potential intake is important in the continued monitoring of workers once an intake has occurred. Because of statistical fluctuations in low-level uranium measurements, it can be very difficult to identify a new intake by routine bioassay if a worker has an elevated baseline.

5.4.5 Other Frequency Situations

For chronic exposures to soluble uranyl compounds approaching the occupational exposure limits, more frequent bioassays should be taken. Some suggested frequencies are to sample after each work break and to sample at the beginning or end of the work week.

If exposure to pure class Y material occurs, monitoring may be done either by fecal analysis, or urinalysis methods with lower MDAs. As a minimum, the monitoring must be adequate to show compliance with the dose limits (10 CFR 835.402(d)). Increased frequency is one way to lower MDAs for urinalysis for the average of a number of measurements.

5.5 ADMINISTRATION OF A BIOASSAY PROGRAM

Administering a bioassay program requires that the policies, procedures, materials, support facilities, and staff be in place to enable a bioassay program to commence. Among the administrative items to address are the following:

- management policy requiring participation in bioassay program by appropriate workers (may be part of an overall radiation protection policy),
- implementing procedures (e.g., criteria for who should participate, scheduling, sample kit instructions, sample kit issue/receipt, follow-up to unsuccessful sample or measurement attempts, data-handling),
- arrangements with appropriate analytical laboratories, including specifications of analysis sensitivity, processing times, reporting requirements, and quality assurance provisions,
- onsite support facilities (e.g., sample kit storage locations, sample kit issue/collection stations, measurement laboratory facilities, equipment maintenance),

- staff selection, qualification, and training,
- total CEDE from all intakes during a year,
- committed dose equivalent (CDE) to organs or tissues of concern from all intakes during a year,
- magnitude of intake for each radionuclide during a year,
- data necessary to allow subsequent verification, correction, or recalculation of doses, and
- gestation period dose equivalent to the embryo/fetus from intake by the mother during the entire gestation period.

Recommendations for testing criteria for radiobioassay laboratories are in ANSI N13.30. These recommendations include calculational methods and performance criteria for bias, precision, and testing levels. The establishment of minimum detection capability must be driven by programmatic needs, ideally related to some concept of a minimum detectable dose, rather than as a single magnitude number.

Some sites have established brief flyers or brochures describing their bioassay measurements. These may be distributed to workers during classroom training, upon notification of scheduled measurements, or at the time of the measurement or sample.

The choice of the measurement technique, or of a combination of techniques, depends on the radioisotopes, physicochemical forms, and exposure pathway.

Because of the wide range of chemical and physical forms of uranium, an appropriate bioassay program is one that does not rely on assumed transportability and will provide data from which radiation dose can be calculated that will not be dependent on the chemical form. This will normally require both in vivo and in vitro bioassay. If the uranium being handled has been shown to be of medium to high transportability, then the bioassay program must be designed to demonstrate that 3 µg U/g kidney has not been exceeded.

Uranium class Y materials cannot be effectively detected at the levels listed in ICRP Publication 54 by ordinary methods available for either lung in vivo counts or urinalysis. This is shown by the fact that the DIL (based on 0.3 ALI as per ANSI/HPS 1995) was 0.06 pCi L^{-1}, which is below the MDA suggested as reasonable for routine uranium alpha urinalysis (0.1 pCi L^{-1}) in the standard.

5.5.1 In Vivo Monitoring

The scheduling and measurement process for obtaining in vivo measurements is usually straightforward. Workers are scheduled for the measurements and results are available shortly after the measurement is completed. The long counting times can impose limitations on the throughput of workers through a measurement facility, making scheduling an important issue. Procedures should be in place to ensure that workers arrive for scheduled measurements and that follow-up occurs when a measurement is not completed or a worker fails to show.

Occasionally, workers are found who are claustrophobic when placed inside in vivo counter cells. Leaving the cell door partially open may help reduce some of the anxiety, but will also likely compromise the low background for which the system is designed.

Many workers want to know the results of their measurements. While a simple statement by the in vivo measurement technician may be adequate, a form letter stating that results were normal (or showed no detection of any of the nuclides of concern) can provide permanent verification. If results are not normal, a form letter can also be used to explain what happens next.

In vivo analysis is most useful for characterizing inhalation exposure of class W or Y compounds of uranium by lung counting. MDAs are generally not sufficiently low to provide reliable information about systemic distribution of soluble uranium at occupational levels. The ^{235}U decays with emission of an energetic (186-keV) photon in high abundance that is used for in vivo monitoring of enriched uranium workers. The other long-lived uranium isotopes emit only low yields of low-energy photons (<60 keV), which are easily attenuated by body tissue and have limited usefulness for in vivo analysis. Internal exposures to aged depleted uranium can be measured in vivo by taking advantage of several photons of moderate energy (63-93 keV) emitted by the ^{234m}Pa daughter of ^{234}Th, which are both short-lived daughters of ^{238}U.

An important aspect of any in vivo measurement program is the calibration and verification testing of the measurement equipment. In vivo measurement results are highly dependent on the determination of a background result. Likewise, calibration using known activities in appropriate phantoms is also important. Phantoms are available commercially or by loan from the U.S. DOE Phantom Library, operated by the Pacific Northwest National Laboratory.[1]

5.5.2 Urine Sampling

Urine sampling programs can be effectively administered using either workplace or home collection protocols. Workplace sampling protocols must determine whether adequate precautions are taken to prevent external contamination of the sample by levels of activity well below the detection capabilities of friskers and workplace monitors. Home collection protocols have the advantage of being sufficiently removed from the workplace to render essentially nonexistent the potential of very low-level contamination of the sample from external sources of uranium. Avoidance of very minor external contamination of the samples is extremely important due to the dosimetric implications of uranium in urine.

Large-volume urine samples are necessary for bioassay monitoring due to the very small urinary excretion rates. Ideally, 24-hour total samples would be preferred; however, such samples often impose substantial inconvenience on workers, resulting in noncompliance with the instructions. As an alternative, total samples can be simulated by either time-collection protocols or volume normalization techniques.

One method of time-collection simulation (NCRP 1987b; Sula et al. 1991) is to collect all urine voided from 1 hour before going to bed at night until 1 hour after rising in the morning for two

(1) For information on or to request loans from the U.S. DOE Phantom Library, contact In Vivo Radioassay Research Facility, at the Pacific Northwest National Laboratory, telephone (509) 376-6102.

consecutive nights. This technique has been reviewed with regard to uranium (Medley et al. 1994) and found to underestimate daily urine excretion by about 14%. Such a finding is not unexpected, since the time span defined by the protocol is likely to be about 18 to 22 hours for most people.

The volume normalization technique typically normalizes whatever volume is collected to the ICRP Reference Man daily urine excretion volume of 1400 mL. Reference Woman excretion (1000 mL/d) may be used for gender-specific programs. As a matter of practicality, routine monitoring programs do not usually use gender as a basis of routine data interpretation, particularly since results are anticipated to be nondetectable under normal conditions.

A third method calls for collection of a standard volume (e.g., 1 liter) irrespective of the time over which the sample is obtained. This method uses the standard volume as a screening tool only for routine monitoring. It does not attempt to relate measured routine excretion to intake, relying on well-defined and timely supplemental special bioassay to give true or simulated daily excretion rates.

The most common sample collection containers are 1-liter polyethylene bottles. Although glass bottles are also used, they pose additional risks of breakage. Wide-mouthed bottles are preferred for convenience and sanitation. The number of bottles included in the kit should be appropriate to the protocol; for a total 24-hour protocol, as much as 3 liters can be expected. Special provisions, such as a funnel or transfer cup, may improve the esthetics of sample collection and provide for added worker cooperation.

Some concerns can exist with length of sample storage before analysis. Storage may come from delays before batching samples in-house or due to transportation times to an offsite laboratory. The longer a sample stands, the more chemical and biological change it can undergo, typically manifesting itself as sedimentation and plate-out on container walls. While samples can be preserved by acidification or freezing, good radiochemistry techniques should ensure essentially complete recovery of any plate-out or sediment. Samples sent offsite for analysis can be preserved with acid, but this method imposes hazardous material shipping requirements. Freezing samples can preserve them, but plate-out and sedimentation upon thawing should still be expected.

Precautions are necessary if a lab uses an aliquot for analysis and extrapolates the aliquot result to the total sample. The aliquoting procedure should be tested using spiked samples to determine if it is representative.

A quality control (QC) verification program should exist for laboratory analyses, including use of known blank samples and samples spiked with known quantities of radioactivity. Ideally, the samples should not be distinguishable by the analytical laboratory from actual worker samples. The number of QC verification samples may range from 5% to 15% of the total samples processed by a large-volume program; a small program focused on submittal of special samples following suspected intakes may have a much higher percentage of controls. An additional QC provision may be to request the analytical lab to provide results of their in-house QC results for independent review.

There are no standard or regulatory requirements for bioassay sample chain-of-custody provisions, nor has there been consensus on their need. Tampering with samples has not been a widely reported or suspected problem. Site-specific chain-of-custody requirements should be based on balancing the need with the resources required to implement them. Some sites have no chain-of-custody requirements associated with bioassay sample collection. At other sites, a simple seal placed on a sample container

following collection by the subject worker is an effective means of providing a small degree of chain-of custody. At the more complex level would be strict accountability requiring signature of issue, certification of collection, and signature of submittal.

Procedures describing details of the bioassay program should be documented. These procedures should include a description of sample collection, analysis, calibration techniques, QC, biokinetic modeling, and dose calculational methods used.

5.5.3 Fecal Sampling

Fecal analysis is most useful in the first few days after a known acute exposure, since a large fraction of either an ingestion or inhalation deposition is excreted in feces. Chronic inhalation exposures to class W or Y uranium can also be characterized by fecal analysis, since a large fraction of the material clears to the GI tract and is eliminated in feces. Urinalysis is the only reliable method for determining inhalation exposures to class D uranium and for monitoring the excretion of systemic uranium. It also provides complementary information, which, when used with in vivo or fecal monitoring results, contributes to greater accuracy in internal dose assessments. Because urinalysis is generally less disruptive to work schedules than in vivo monitoring and more acceptable to workers than fecal monitoring, it occupies a prominent place in most uranium bioassay programs.

Fecal analysis is often more likely to detect exposure to highly insoluble class Y material than urinalysis. The ratio between the fecal excretion level per day and the urine excretion level per day is greater than 7, as calculated for a 90-day sampling interval. All action levels are above the typically attainable MDA for fecal analysis of 0.1 pCi per L (ANSI 1996). Thus, it is recommended that facilities that have a significant class Y uranium exposure potential should have fecal analysis capabilities available to them, unless they have urinalysis methods that have MDAs well below the 0.1 pCi per sample (ANSI/HPS 1995).

A fecal sampling program must be designed to optimize worker cooperation, whether collecting samples at home or in the workplace. Since the frequency of fecal voiding varies greatly from person to person, the sample collection program must be adaptable. Flexibility in sample dates is important. It is suggested that when a fecal sample is required, the worker be provided with a kit and instructed to collect the sample, noting the date and time of voiding on the sample label. This practice can reduce the likelihood of unsuccessful samples. If multiple samples are required (for example, to collect the total early fecal clearance following an acute inhalation exposure), the worker may be given several kits and told to collect the next several voidings, noting the date and time of each.

Since the total fecal voiding should be collected, thought must be given to the kit provided. Fecal sampling kits can be obtained from medical supply companies or designed by the site. A typical kit might include a large plastic zipper-closure bag to hold the sample, placed inside a 1- to 2-liter collection bucket with a tight-fitting lid. The bucket and bag can be held in place under a toilet seat by a trapezoid-shaped bracket with a hole through it sized to hold the bucket. After sample collection, the zipper bag is sealed, the lid is snapped tight on the bucket, and the bucket placed in a cardboard box.

Following collection, the provisions for sample handling, control, analytical, and QC are similar to those described above for urine samples. One particular concern for fecal analysis is the potential difficulty of dissolving class Y uranium in the fecal matrix. While nitric acid dissolution may be adequate, enhanced digestion using hydrofluoric acid may be preferred.

5.5.4 Conditions for Adjustments of Action Levels

When workers are potentially exposed to other radiation sources or toxic agents, the action levels should be reevaluated. Since uranium has both chemical and radiological toxicity characteristics, urinalysis results should be interpreted both in terms of mass and radioactivity to ensure that the most appropriate set of action levels is used (ANSI/HPS 1995).

5.6 MODELING THE BEHAVIOR OF URANIUM IN THE BODY

A key issue in uranium dosimetry is the modeling of how the material behaves in the body. Some of the standard models are described below, with additional discussion of the biological behavior given in Section 2.4. It is important that an internal dosimetry program establishes and documents the routine models and assumptions used for dosimetry. Computer codes typically incorporate standard models but may allow the flexibility to alter parameters. When altered on an individual-specific basis, the revised models need to be addressed in the pertinent case evaluations or the technical basis.

5.6.1 Respiratory Tract

The respiratory tract model of ICRP Publication 30 is commonly used for evaluating inhalation intakes of radioactivity. The model has been widely published and included in reference books (e.g., Cember 1996; Shleien 1992) and internal dosimetry computer codes, hence it is not reproduced here. In 1994 a newer respiratory track model, *ICRP Publication 66,* was published (ICRP 1994)

Like all models, the ICRP respiratory tract model represents anticipated behavior. Once an exposure has occurred and actual data become available, deviations from the model in light of the data are appropriate.

In practice, the model has proved extremely valuable for calculating derived investigation levels and estimating intakes from bioassay data, using standard D, W, and Y classes of material. Model interpretation becomes more subjective when extensive data become available. Others (Carbaugh et al. 1991 and La Bone et al. 1992) have provided excellent examples of two cases where the standard lung model assumptions did not fit the data.

Most internal dosimetry computer codes allow adjustment of particle size and selection of solubility classes. Some codes also permit detailed adjustment of the model's individual compartment parameters; with these codes, it may be possible to arrive at various subjective interpretations to explain the same data. When adjustments are made to the standard assumptions, it is important to explain what those adjustments are and why they were made.

5.6.2 Gastrointestinal Tract

The gastrointestinal tract model of ICRP Publication 30 is also widely promulgated and used for evaluating ingestion intakes. It is also coupled to the respiratory tract for inhalation intakes. The model is particularly subject to individual variations in fecal voiding frequency, so judgment must be used in its application to human data.

A key parameter of the model for internal dosimetry is the f_1 factor for absorption to blood of material in the small intestine, that is, the fraction of a stable element reaching the body fluids following

ingestion. The GI tract absorption factors for different solubility classes of uranium are given in Table 5-8. The parameter f_1 has large uncertainties and likely varies with age as well as with diet (Wrenn et al. 1989) and with the chemical and physical form of the uranium. Systemic burdens and committed doses are directly proportional to f_1 for ingestion of long-lived uranium, provided that f_1 is greater than 0.01. Fractional absorption from the GI tract is highly variable, but values of 0.05 for soluble compounds and 0.005 for insoluble ones provide sufficient protection for all but perhaps very high concentrations in ingested materials.

Table 5-8. GI Tract Absorption Factors for Uranium[a]

INHALATION CLASS	CHEMICAL FORM[b]	ABSORPTION FACTOR *(f_1)*
D	UF_6, UO_2F_2, $UO_2(NO_3)_2$	0.05
W	UO_3, UF_4, UCl_4	0.05
Y	UO_2, U_3O_8	0.002

(a) From ICRP 1979; ICRP 1988a; ANSI/HPS 1995.
Note that some compounds that have been classified as class Y have shown a more rapid clearance from the lung than for other class Y compounds. This may be due to the existence of mixtures having more than one physico-chemical form (ICRP 1988a; Fisher et al. 1990).

(b) The solubility of uranium oxides is very dependent on heat treatment. The rate of oxidation may also affect the solubility. Although references assign inhalation classes to various uranium compounds, it is recommended that solubility studies be performed to characterize the actual materials present. For example, depending on factors such as the heat treatment and rate of oxidation of the materials, UO_2 could be class W or Y, U_3O_8 could be class W or Y, and UO_3 could be class D or W.

5.6.3 Systemic Retention and Excretion of Uranium

There are two ways in which the systemic retention of soluble uranium may be calculated: using the ICRP Publication 30 model or the Fisher-Modified Wrenn-Lipsztein model. The latter, which comprises modifications of the ICRP Publication 30 lung model and of the Wrenn-Lipsztein urinary excretion model, is used for class W and class Y uranium. Professional judgment must be used in selecting one of these functions as the primary model to be used in routine bioassays and as the model of choice in individual dose assessments.

5.6.3.1 ICRP Publication 30 Uranium Systemic Retention Function

The systemic retention function per unit intake ($r_{as}[t]$) for uranium given by the ICRP in Publications 30 and 54 (ICRP 1979; ICRP 1988b) is as follows:

$$r_s^a(t) = 0.54e^{-0.693\frac{t}{0.25}} + 0.24e^{-0.693\frac{t}{6}} + 0.2e^{-0.693\frac{t}{20}} + 0.001e^{-0.693\frac{t}{1500}} + 0.023e^{-0.693\frac{t}{5000}} \quad (5.1)$$

where *t* is the time in days after intake of uranium. Of the uranium entering the transfer compartment, 0.54 is directly excreted ($T_{1/2_b}$ of 0.25 d), 0.2 and 0.023 are translocated to mineral bone ($T_{1/2_b}$ of 20 and 5000 d, respectively), 0.12 and 0.00052 are translocated to the kidneys ($T_{1/2_b}$ of 6 and 1500 d, respectively), and 0.12 and 0.00052 are translocated to all other tissues ($T_{1/2_b}$ of 6 and 1500 d, respectively).

5.6.3.2 Fisher-Modified Wrenn-Lipsztein Uranium Systemic Retention Function

The Fisher-Modified model (Fisher et al., 1991) consists of modifications of clearance half-times from two biokinetic models: the ICRP Publication 30 respiratory tract model (ICRP 1979) and the Wrenn-Lipsztein recycling model for systemic uranium (Wrenn, et al. 1989). As shown in Table 5-9, the clearance half-time of uranium from ICRP Publication 30 lung compartment *e* to the blood is changed from 0.5 d to 0.03 d or 45 min. The clearance half-time of uranium from the kidney to urine in the Wrenn-Lipsztein urinary excretion model is changed from 15 d to 6 d.

Table 5-9. Mathematical Model to Describe Clearance from the Respiratory Tract for the Fisher-Modified Wrenn-Lipsztein Uranium Urinary Excretion Model[a)]

		Class D		Fast or Slow
Region	FR	T½ (d)	F	
N-P	0.30	0.01	0.5	F
		0.01	0.5	F
	0.08	0.01	0.95	F
		0.2	0.05	F
P	0.25	0.03[(b)]	0.8	S
		NA	NA	F
		NA	NA	S
		0.5	0.2	S
L	NA	0.5	1.0	S
		NA	NA	---

(a) F_R = Regional fractions; T½ = Removal half-times (d);
F = compartmental fractions for each of the three classes of retained materials; and NA = not applicable.
(b) Modified from ICRP 30 (ICRP 1979) value of 0.5 d.

The function for the Fisher-Modified Wrenn-Lipsztein Model is as follows:

$$r_s^a(t) = 0.673e^{-0.693\frac{t}{0.25}} + 0.007e^{-0.693\frac{t}{1.1}} + 0.08e^{-0.693\frac{t}{6}} + 0.07e^{-0.693\frac{t}{26}} + 0.15e^{-0.693\frac{t}{300}} + 0.02e^{-0.693\frac{t}{3700}} \quad (5.2)$$

where $r_{as}(t)$ is the systemic retention per unit intake and t is the time in days after intake of uranium. Of the uranium entering the transfer compartment, 0.673 is directly excreted ($T_{1/2_b}$ of 0.25 d), 0.007 is translocated to red blood cells ($T_{1/2_b}$ of 1.1 d), 0.08 is translocated to kidneys ($T_{1/2_b}$ of 6 d), 0.07 is translocated to soft tissues ($T_{1/2_b}$ of 26 d), and 0.15 and 0.02 are translocated to mineral bone ($T_{1/2_b}$ of 300 and 3700 d, respectively). The parameter values for this function are summarized in Table 5-10.

Table 5-10. Fisher-Modified Wrenn-Lipsztein Uranium Model Parameter Values[(a)]

Organ(s)	Coefficient	Fraction	Biological Half-Life (d)
Excretion	0.673	1.0	0.25
Red blood cells	0.007	1.0	1.1
Kidneys	0.08	1.0	6
Soft tissues	0.07	1.0	26
Bone	0.17	0.88	300
		0.12	3,700

(a) (Fisher et al., 1991)

5.6.3.3 Urinary Excretion of Uranium

There currently are three possible urinary excretion functions for uranium compounds: the ICRP Publication 30, Wrenn-Lipsztein, and the Fisher-Modified Wrenn-Lipsztein models. Professional judgement must be used in selecting one of these functions as the primary function to be used in routine bioassays in this internal dosimetry program and as the function of choice in individual dose assessments. The value of fractional urinary excretion used to evaluate bioassay data is $f_u = 1.0$ (ICRP, 1988b).

5.6.3.4 ICRP Publication 30 Uranium Urinary Excretion Function

The derivative of the ICRP uranium retention function with respect to time multiplied by a urinary excretion fraction (f_u) of 1.0 describes the urinary excretion of uranium. The urinary excretion is as follows:

$$\dot{e}^{i}_{u}(t) = 1.50e^{-0.693\frac{t}{0.25}} + 2.77\times10^{-2}e^{-0.693\frac{t}{6}} + 6.93\times10^{-3}e^{-0.693\frac{t}{20}} + 4.62\times10^{-7}e^{-0.693\frac{t}{1500}} + 3.19\times10^{-6}e^{-0.693\frac{t}{5000}} \quad (5.3)$$

where $e_{i_u}(t)$ is the fractional urinary excretion rate per unit intake as a function of time t after a single intake of uranium. The parameter values for this function are summarized in Table 5-11.

Table 5-11. ICRP Publication 30 Uranium Urinary Excretion Parameter Values[a]

Compartment	Coefficient (d^{-1})	Biological Half-Life (d)
1	1.5	0.25
2	2.77×10^{-2}	6
3	6.93×10^{-3}	20
4	4.62×10^{-7}	1,500
5	3.19×10^{-6}	5,000

(a) (ICRP 1979 and 1988b)

5.6.3.5 Fisher-Modified Wrenn-Lipsztein Uranium Urinary Excretion Model

As discussed above, this model (given in Equation 5.2) represents a modification of the clearance half-times in the ICRP Publication 30 respiratory tract model and in the Wrenn-Lipsztein recycling model for systemic uranium. In an effort to simplify this recycling model, Fisher et al. also derived a five-compartment model to fit the curve predicted by the above-mentioned Fisher-Modified Wrenn-Lipsztein function. The replacement function that describes this five-compartment model is as follows:

$$\dot{e}_u^i(t) = 0.86e^{-0.693\frac{t}{0.25}} + 0.0048e^{-0.693\frac{t}{6}} + 0.00069e^{-0.693\frac{t}{26}} + 0.00017e^{-0.693\frac{t}{300}} + 2.5\times10^{-6}e^{-0.693\frac{t}{3700}} \quad (5.4)$$

where $e_{iu}(t)$ is the urine excretion rate per unit intake and t is the time in days after intake of uranium. The parameter values for this replacement function are given in Table 5-12.

Table 5-12. Parameter Values for the Replacement Function for the Fisher-Modified Wrenn-Lipsztein Uranium Model[a]

Compartment	Fractional Uptake from Transfer Compartment	Excretion Constant (d^{-1})	Residence Halftime (d)
Transfer	0.673	2.77	0.25
Kidneys and soft tissues	0.15 0.07	0.116 0.0267	1.1 0.02 67
Bone volume	0.15 0.02	0.00231 0.000187	300 3,700

(a) (Fisher et al., 1991)

While Equation 5.4 represents a replacement function for the Fisher-Modified Wrenn-Lipsztein model, it does not have output values that are identical to the model given by Equation 5.2, especially between 1 and 5 d.

Standard models for the systemic retention of uranium are commonly used for internal dosimetry because in vivo detection of uranium within the individual systemic compartments is not usually possible. Three models were proposed by the ICRP over a 10-year period. Each of them has had a wide application, and the ICRP has suggested that results derived using one model do not need to be rederived for compliance purposes using the newest model. Studies by the U.S. Transuranium Registry (Kathren 1994) have indicated that alternate compartments and clearance half-times may be more appropriate.

For convention, this document will use the ICRP 30, Part 4 systemic retention parameters for uranium internal dosimetry. The ICRP model (ICRP 1979) for uranium is a "once-through" model: 53.6% of uranium entering the transfer compartment (the blood) is assumed to be excreted directly in urine; the remainder is distributed among the bone (22.3%), kidney (12%), and other soft tissue (12%).

Others (Durbin 1986 and Wrenn et al. 1985) have described a recycling model based on extensive review of available data. In this model, 67.3% is excreted in urine, 17% taken up by bone volume, 7% by soft tissue, and 8% by the kidney.

These two models are the only ones to have been widely applied to evaluate exposure to uranium in recent years. The recycling model is a more physiological representation, but the ICRP model is more widely accepted.

The appropriate toxicokinetic model for uranium entering the blood after exposure is the recycling model. The different physical and chemical forms of the uranium are unimportant. The parameters of the model may not be appropriate at high uranium concentrations.

Uranium is transported through the bloodstream as a carbonate ion ($UO_2[CO_3]_2^{2-}$). The chemical form of uranium that enters the blood is dependent on the chemical form of the uranium that was inhaled. A substantial portion of uranium filtered by the kidneys is temporarily retained in the renal tubules before passing to the urinary bladder contents (ICRP 1995).

The skeletal behavior of uranium is in some ways qualitatively similar to that of the alkaline earths, with UO_2^{2+} exchanging with Ca^{2+} on the bone mineral surfaces. There remain substantial uncertainties regarding the long-term retention of uranium in bone, as well as soft tissues (ICRP 1995).

Urinary excretion of uranium is assumed to arise from:

- uranium moving directly from plasma to the urinary bladder contents, accounting for 63% of uranium leaving the circulation or

- uranium moving to the urinary bladder contents after temporary residence in the renal tubules, accounting for 12% of uranium leaving the circulation, with a half-life of 7 days (ICRP 1995).

5.6.4 Chemical Toxicity

Inhalation is the most important route of uranium intake for industrial workers. The retention of uranium in the lungs is influenced by the dissolution rate of the uranium compound inhaled. The major portion of dissolved uranium is quickly absorbed into extracellular fluid (Durbin 1986). Ingestion is an important route of uranium intake for the general public (Wrenn et al. 1985). Saliva contains digestive enzymes and high concentrations of HCO_3^- and CO_3^{2-} at pH 6 to 7.4. As a result, uranyl bicarbonate complexes would be expected to form. When no digestion is occurring, gastric juice contains high concentrations of HCO_3^- and CO_3^{2-}. Under these relatively alkaline conditions, dissolved uranium available for absorption would be expected to form uranyl bicarbonate and carbonate complexes.

When food is ingested, digestive enzymes are secreted, along with HCl to maintain the pH at about 2, which is the optimal pH for enzyme activity. Under these conditions, dissolution of less soluble uranium compounds would increase. Uranium carbonate complexes are unstable at low pH.

Deposition of uranium in the kidney is not uniform and is located in distal tubules. Uranium bicarbonate complexes are transported to the tubules when the complex dissociates as a result of decreased bicarbonate concentrate and decreased pH. Uranium is excreted from the kidney as an equilibrium between uranyl bicarbonate complex concentration and cell-bound uranyl ion becomes re-established.

Uranyl ion transported in blood is initially deposited on endosteal bone surfaces (Neuman 1953; Priest et al. 1982). It becomes incorporated within the bone volume so that a diffuse distribution is achieved by 72 days after deposition (Rowland and Farnham 1969). Uranyl ion is cleared from bone slowly by ion exchange with Ca^{2+} as bone remodeling progresses (Durbin, 1986).

The kidney is the primary target of the chemical toxicity of uranium. The critical level of the metal above which damage may be expected has not been rigorously defined. The suggested guidance level of 3-µg U/g kidney is not adequately documented by experimental data; however, it provides a basis for preventing an increased frequency of end-stage renal disease in uranium workers.

Data on human exposures and the effect of various intakes of uranium are summarized in Table 5-13. These data indicate that a single intake of 8 mg of natural uranium would be well below the level that could cause permanent kidney damage in most individuals, and that 4 mg intake would likely cause no observable effects. The urine levels for situations in which chemical toxicity might be of concern are based on interpretation of the data (McGuire 1991).

Table 5-13. Health Effects from Acute Intake of Soluble Uranium[a]

HEALTH EFFECTS	URANIUM per kg BODY WT (mg U kg^{-1})[b]	URANIUM IN 70 kg PERSON (mg)	URANIUM INTAKE BY 70 kg PERSON (mg)[c]
50% Lethality	1.63	114	230
Threshold for permanent renal damage	0.3[d]	21	40
Threshold for transient renal damage	0.058	4.06	8
No effect	0.03	2.1	4

(a) (ANSI 1995); based on review (McGuire 1991).
(b) Based on review (Just and Emler 1984), except where noted.
(c) For this table, intake is defined as the total amount of material inhaled into the body. It includes material immediately exhaled in addition to material absorbed within the body. For small uranium particles in soluble form, about half of the intake will be absorbed by the body according to ICRP 30 (1979).
(d) See discussion in (Just and Emler 1984).

The Canadian guidance (Health and Welfare Canada 1987) suggests chemical toxicity is the dominant consideration over radiological toxicity only for more soluble class D compounds. The higher specific activity over natural uranium (2.5×10^7 Bq/kg) could bring the equilibrium mass burden equivalent to 6.5 Bq above the lowered permissible chemical burden. Then, under continuous exposure conditions, both classes D and W natural uranium could exceed permissible chemical levels in the body after some months of continuous exposure at 0.3 DAC levels (ANSI/HPS 1995).

Also, it has been shown that below an enrichment of 20% ^{235}U by weight, a 900-mg kidney burden could be exceeded in a single intake without exceeding the ALI. For a 900-mg kidney limit, radiological considerations limit the DILs for enrichments greater that 20%, although possible chemical toxicity in the long-term must also be considered even for these higher enrichments (ANSI/HPS 1995).

5.6.5 Natural Uranium Balance in Man

Uranium is present in trace quantities throughout the environment. As a result, man ingests about 2 µg of natural uranium each day in food and fluids. A similar quantity is excreted each day in the feces and urine. The uranium balance for reference man is presented in Table 5-14.

Table 5-14. Uranium Balance for Reference Man[a]

Intake:			
	Food and fluids:	1.9	µg/day
	Inhalation:	7.0 E-3	µg/day
Losses:			
	Feces:	1.4 - 1.8	µg/day
	Urine:	0.05 - 0.5	µg/day
	Other (hair)	0.02	µg/day

(a) (ICRP 1992).

The range of intake and losses has been observed to vary over several orders of magnitude, depending upon the uranium concentration in foods and in the water supply.

5.6.6 Mother-to-Fetus Transfer

The embryo/fetus is included as part of the 10% of the systemic uptake that is uniformly distributed in all "other" soft tissues except the liver and gonads. Methods for evaluating embryo/fetal uptake and dose have been described in NUREG/CR-5631 (Sikov et al. 1992) and its 1993 addendum (Sikov and Hui 1993). For uptakes occurring during the first two months of pregnancy, the activity in the embryo/fetus is assumed to have the same concentration as in the mother's "other soft tissue." For later uptakes, the embryo/fetal concentration gradually increases relative to the maternal concentration, but is assumed to remain uniformly distributed in the embryo/fetus. At three months, the embryo/fetal concentration is one-and-a-half times the mother's "other" soft tissue concentration. At six months, it is twice the mother's, and at eight months, it is three times the maternal "other" concentration. Following transfer to the embryo/fetus, uranium activity is assumed to remain uniformly distributed, without clearance, until birth.

The Nuclear Regulatory Commission has developed simplified methods for assessing the gestation period dose to an embryo/fetus in Regulatory Guide 8.36. Application of these methods shows that very large maternal intakes of uranium are required to produce uptakes that would deliver 500 mrem, or even 50 mrem to the embryo/fetus. The NUREG/CR-5631 Addendum notes that maternal inhalation intakes of nominally 100 times the annual limit on intake (ALI) are required to give a 50-mrem embryo/fetal dose. For ingestion intakes, a 1000 ALI maternal intake of uranium is required to give a 50-mrem dose to the embryo/fetus. Thus, providing adequate radiation protection to limit maternal intake of uranium to the occupational limits will adequately provide for the protection of the embryo/fetus.

5.7 INTERPRETATION OF BIOASSAY RESULTS

Bioassay measurements detecting uranium in workers can be initially interpreted as indicating that occupational intakes may have occurred. Standard bioassay procedures are not sufficiently sensitive to detect differentiate occupational intakes from the range of environmental background levels in vivo or in excreta. For example, there may be significantly elevated uranium bioassay results in certain populations who obtain their drinking water from wells. Since most uranium bioassay measurement

procedures include counting for radioactivity as the final step in the measurement process, they are also subject to the statistics associated with the counting process.

Two key questions associated with bioassay data are: 1) When does a sample result indicate the presence of something (i.e., when is the analyte detected); and 2) What is the overall capability of the bioassay method for continual assurance of detection of the analyte?

The decision level L_c (also called the critical level for detection) is the level for a given measurement that indicates the likely presence of the analyte. The L_c is dependent on the probability of obtaining false positive results (type I, or alpha, error) that is acceptable to the program. A 5% probability of false-positive results is a common design parameter of measurement programs, implying that for a large number of measurements, 5% of the time results will be indicated as positive when in fact there is no activity present. The L_c is calculated from results of analyses of blank samples. Once a measurement is performed, it is appropriate to compare it with the L_c to determine whether or not the result is "positive" (i.e., the analyte is detected).

The MDA is the level at which continued assurance of detection can be provided. The MDA is a function of the probabilities of both false positive and false negative (type II, or beta) errors and is typically based on a 5% probability for each kind of error. The MDA is also determined from analysis of blank samples, but is substantially higher than the L_c. The MDA is appropriate for use in designing bioassay programs and as the basis for estimating minimum detectable intakes and doses as indicators of program sensitivity. The MDA should not be used as a comparison with actual measurements to determine whether or not activity is present (i.e., <MDA is not an appropriate use of the concept).

Methods for calculating both L_c and MDA are given in ANSI N13.30.

As an alternative to the L_c and MDA of classical statistics, there have been proposals (Miller et al. 1993) to use Bayesian statistical methods for evaluating bioassay data.

General follow-up actions to abnormal bioassay measurements should include data checks, timely verification measurements, work history reviews, and performance of special in vivo measurements or excreta sample analyses for intake and dose assessments.

5.7.1 In Vivo Count Results

In vivo uranium measurements are generally relatively insensitive with regard to levels of occupational exposure concern. This applies particularly to routine chest or lung counting, skeleton counting, and liver counting. For that reason, any detection of uranium should be investigated. The investigation should address the validity of the measurement by reviewing the spectrum and its associated background subtraction. These reviews are particularly important if the result is near the L_c. Follow-up to a positive result should include a confirming measurement. Ideally, this should be an immediate (same day) recount of equal or higher sensitivity. The farther removed in time a verification measurement is from the original measurement, the more important it becomes to factor in potential lung clearance in comparing the two measurements. A follow-up measurement taken 30 days after an initial high-routine may not be capable of providing verification if the material of concern exhibits class W behavior.

Chest-wall thickness has a significant impact on chest counting. Corrections are commonly made using a height-to-weight ratio or ultrasonic methods (Kruchten and Anderson 1990).

Corrections may be required to address apparent detection in one tissue resulting from photon crossfire from another tissue. For example, chest counting is performed primarily to estimate activity in the lung. Yet, there is substantial bone over the lungs (rib cage, sternum) and behind the lungs (vertebrae). Plutonium and uranium are both bone-seeking radionuclides which will deposit on those bone surfaces and can interfere with chest counting. It is possible for a person having a systemic burden of uranium from a wound in the finger to manifest a positive chest count from material translocated to the skeleton, axillary lymph nodes, or liver (Carbaugh et al. 1989; Graham and Kirkham 1983; Jeffries and Gunston 1986). Interpreting such a chest count as a lung burden can render dose estimates somewhat inaccurate.

When comparing in vivo measurements made over many years, it is important to make sure that the measurements are, in fact, comparable. One consideration is to make sure that corrections have been consistently applied to all similar measurements. It is not unusual for measurement systems to be replaced or to change the algorithms used for calculating results over time. Step changes in data can occur and should be addressed in monitoring long-term detectable trends (Carbaugh et al. 1988).

In vivo wound counting for uranium is usually one facet of special bioassay. While a portable alpha survey meter may show if surface contamination is present at the wound site or contamination of the wounding object, alpha detectors are not capable of measuring imbedded activity or activity masked by blood or serum. Thus, uranium facilities should have available a wound counter utilizing a thin sodium iodide or semi-conductor (e.g., planar germanium) detector. Such detectors are capable of measuring the low-energy photons emitted from uranium. The ability to accurately quantify wound activity is highly variable, depending on the calibration of the equipment and how deeply imbedded material is in the wound. If the object causing a wound and blood smears taken at the time of a wound show no detectable activity, then a wound count also showing no detectable activity is probably sufficient to rule out an intake. If the wounding object or the blood smears show detectable activity, special urine samples should be obtained regardless of the wound count result. In this latter circumstance, lack of detectable activity on a wound count could be attributable to deeply imbedded material at the wound site or to rapid transportation of material from the wound to the systemic compartment.

5.7.2 Urine Sample Results

Detection of uranium activity in a routine or special urine sample using commonly available radiochemical measurement techniques should be investigated as a potential intake. A data review should be made to determine if the sample result was correctly determined, and batch QC sample data should be verified.

If the result is near the L_C, it is possible that statistical fluctuation of the measurement process could account for the apparent detection. Recounting the final sample preparation once or twice can be a helpful technique to verify a result or classify it as a false-positive. If the first recount also detects the analyte, it can be concluded that the sample does contain the analyte (the likelihood of two consecutive false positives at a 5% type I error per measurement is 0.0025, or 0.25%.) If the first recount does not detect the analyte, a second recount can be performed as a tie-breaker.

An investigation should be initiated for any abnormal uranium urinalysis result. "Abnormal" for a person with no prior history of intake should be interpreted as any detectable activity.

Once an intake is confirmed, sufficient samples must be obtained to establish a reasonably anticipated baseline against which future measurements can be compared. This is important both to provide future verification of the accuracy of the assessment and to identify potential additional intakes.

The statistical fluctuation of low-level measurements can be particularly troublesome for long-term excretion patterns. Factors of two can be easily expected due to day-to-day variability and imprecise adherence by the worker to urine collection protocols.

5.7.3 Fecal Sample Results

Fecal samples are much more sensitive to detection of intakes than are urine samples and, consequently, are an important part of follow-up bioassay monitoring for potential intakes initially identified by workplace indications. Pitfalls to the data interpretation include highly variable individual fecal voiding patterns, ranging from more than one per day to one every few days. This makes it extremely important to know what time interval is represented by a collected fecal sample. While a single set of fecal data can be normalized to a daily excretion rate for Reference Man, it is not likely to improve the quality of assessment.

The preferred fecal sampling protocol following an intake is to collect all the early fecal clearance (meaning total feces for the first 5 to 7 days). This method will allow a good estimation of inhalation or ingestion intake, but does not readily permit discrimination of inhalation from ingestion, or identify whether inhaled material exhibits class D, W, or Y clearance patterns. For optimum interpretation, total fecal collection should be interpreted in light of early urine and in vivo data for preliminary estimates. The urine data is likely to be particularly valuable in conjunction with fecal data to classify an intake as class W or Y. Longer-term follow-up fecal samples at nominally 30, 60, and 90 days post-intake should substantially improve the classification of material as class W or Y.

Fecal sampling can also be applied to monitor excretion at long times post-intake. One caveat in such sampling is that a worker still active in a uranium facility may be incurring very minor chronic exposure, which can significantly interfere with long-term interpretation of acute exposure data. Papers (Bihl et al. 1993) have discussed experience with a routine fecal sampling program.

5.7.4 Use of Air Sample Data in Internal Dosimetry

Results of air sampling and continuous air monitoring implying more than 40 DAC-hours exposure should be used to initiate special bioassay to assess intakes of uranium. Although bioassay data are the preferred method for assessing intakes and internal doses, air sample data can be used if bioassay data are unavailable or determined to be inadequate or nonrepresentative. Air sample data can be used to calculate an exposure to airborne material either in terms of DAC-hours or potential radioactivity intake as follows:

$$\text{DAC-hours} = \frac{\text{Air Concentration}}{\text{DAC}} \times \text{Duration (hours)} \qquad (5.5)$$

$$\text{Intake} = \text{Air Concentration} \times \text{Breathing Rate} \times \text{Time} \quad (5.6)$$

If air sample results are $H_{E,50}$ representative of air breathed by individuals, then doses can be calculated using the 5-rem stochastic limit for CEDE ($H_{E,50}$)or the 50-rem nonstochastic limit for committed tissue dose equivalent ($H_{T,50}$) and the respective stochastic or nonstochastic DAC or ALI conversion factor, as shown below:

$$H_{50} = (\text{DAC-hours}) \times \frac{\text{Dose Limit}}{2000\ \text{DAC-hours}} \quad (5.7)$$

$$H_{50} = \text{Intake} \times \frac{\text{Dose Limit}}{\text{ALI}} \quad (5.8)$$

If respiratory protection is worn by workers, the appropriate respirator protection factor may be applied to the above calculations (i.e., dividing the calculated result by the protection factor.)

General air sampling programs should be augmented by breathing zone sampling when air concentrations to which individuals are exposed might be highly variable. Breathing zone sampling may include both fixed-location and personal (lapel) air samplers. Personal air samples are more likely to be representative of actual exposure conditions than are samples collected at fixed locations, and they can be particularly useful for assessing potential intakes involving short-term exposure to well-monitored air concentrations.

5.8 DOSE ASSESSMENT

Dose assessment involves collecting and analyzing information concerning a potential intake and developing a conclusion regarding the magnitude of intake and its associated committed dose equivalents. Dose assessments are conducted by investigating the nature of a potential intake and by analyzing bioassay measurement results or other pertinent data. Biokinetic models are used in conjunction with bioassay data to evaluate the intake, uptake, and retention of uranium in the organs and tissues of the body. Intake estimates can then be used to calculate committed effective and organ dose equivalents. It is essential that good professional judgement be used in evaluating potential intakes and assessing internal doses. A number of considerations for dose assessments have been identified (Carbaugh 1994).

Computer codes are commonly used for assessment of intakes, dose calculation, and bioassay or body content projections. An overview of what should be considered in selecting a computer code, as well as descriptions of a number of internal dosimetry codes available in 1994, has been developed (La Bone 1994). Internal dosimetry code users should understand how the code works and be aware of its limitations. Computer codes merely provide the logical result of the input they are given. Use of a particular computer code does not necessarily mean a dose estimate is correct.

As used in this section, the definition of "intake" is the total quantity of radioactive material taken into the body. Not all material taken into the body is retained. For example, in an inhalation intake, the ICRP Publication 30 respiratory tract model predicts that, for 1-µm particles, 63% of the intake will be

deposited in the respiratory tract; the other 37% is immediately exhaled (ICRP 1979). For a wound intake, material may be initially deposited at the wound site. Once the material has been deposited, it can be taken up into systemic circulation either as an instantaneous process (e.g., direct intravenous injection of a dissolved compound) or gradually (e.g., slow absorption from a wound site or the pulmonary region of the lung). Both the instantaneous and slow absorption processes are often referred to as uptake to the systemic transfer compartment (i.e., blood). Once material has been absorbed by the blood, it can be translocated to the various systemic organs and tissues.

An understanding of this terminology is important to review of historical cases. Before DOE Order 5480.11, many sites reported internal doses not as dose equivalent estimates but as an uptake (or projected uptake) expressed as a percentage of a maximum permissible body burden (MPBB). The standard tabulated values for MPBBs were those in ICRP Publication 2 (ICRP 1959). Many archived historical records may have used this approach. DOE Order 5480.11 (now superseded) required calculation of dose equivalent. Now, 10 CFR 835 has codified the calculation of intakes and committed doses.

5.8.1 Methods of Estimating Intake

There are several published methods for estimating intake from bioassay data (Skrable et al. 1994; Strenge et al. 1992; ICRP 1988b; King 1987). These methods each employ an idealized mathematical model of the human body showing how materials are retained in and excreted from the body over time following the intake. An intake retention function (IRF) is a simplified mathematical description of the complex biokinetics of a radioactive material in the human body. These functions are used to predict the fraction of an intake that will be present in any compartment of the body, including excreta, at any time post-intake. Intake retention functions incorporate an uptake retention model that relates uptake to bioassay data and a feed model that relates intake to uptake and bioassay data. ICRP Publication 54 (ICRP 1988b) and others (Lessard et al. 1987) contain compilations of IRFs.

In its simplest form, a compartment content at any time post-intake (Q_t) can be expressed as the product of intake multiplied by the intake retention function value for compartment Q at time t post-intake, or:

$$Q_t = \text{Intake} \times \text{IRF}(Q_t) \tag{5.9}$$

Results predicted by the model can then be compared with the observed bioassay data. Such results are often referred to as expectation values.

Simple algebraic manipulation of the model allows calculation of intake from the compartment content at time t, as shown below:

$$\text{Intake} = \frac{Q_t}{\text{IRF}(Q_t)} \tag{5.10}$$

When multiple data points are available for a compartment, the intake can be estimated using an unweighted or weighted least-squares fitting procedure, as described by Skrable (Skrable et al. 1994) and Strenge (Strenge et al. 1992) or as can be found in most statistics textbooks. As an alternative, data can be fit by eye to a graphical plot; however, the apparent fit can be misleading if data have been logarithmically transformed.

Intake can also be estimated from air sample data, as described in Section 5.7.4. This method is appropriate if bioassay data are not available or insufficiently sensitive. Intake estimates based on air samples and bioassay data are also appropriate as a check on each other. Valid bioassay data showing detectable results should be given preference over intake estimates based on air sample results.

5.8.2 Alternate Methods of Intake Assessment

Historically, intake as described in the foregoing section was not always calculated when assessing uranium exposures. Estimates of uptake using recognized methods (Langham 1956, Healy 1957, Lawrence 1987) focused on assessing the magnitude of radioactivity retained in the body, rather than intake (which includes material not retained and of no dosimetric significance). These methods were (and are) dosimetrically sound in so far as estimates of deposition and uptake are concerned.

5.8.3 Estimating Effective Dose Equivalent from Intakes of Uranium

The committed dose equivalent resulting from an intake of uranium may be calculated by multiplying the estimated intake (I) by an appropriate dose conversion factor (DCF):

$$H_{50} = I \times DCF \tag{5.11}$$

Dose conversion factors can be obtained from tabulated data in Federal Guidance Report No. 11 , ICRP Publication 30, Part 4, in the Supplement to Part 1 of ICRP Publication 30, or calculated directly using computer programs. Substituting the ICRP Publication 48 (ICRP 1986) model parameters of 50% skeleton and 30% liver translocation for the assumptions in ICRP Publication 30, Parts 1 or 4, has little impact on the $H_{E,50}$ per unit intake, but does alter the committed organ dose equivalent per unit intake. Such substitution of models is acceptable, provided that the model is documented and consistently applied.

Values for simplified dose conversion factors can be obtained by dividing a dose limit by the corresponding value for the ALI. A caution must be observed with this approach: not all tabulated values of ALIs are the same. The ALIs are commonly rounded in most tabulations to one significant figure (e.g., as in ICRP Publication 30 and Federal Guidance Report No. 11). Substantial variation can occur as a result of units conversion. For example, Federal Guidance Report No. 11 lists the ALI for ^{239}Pu class Y inhalation as both 6 x 10^{-4} MBq (600 Bq) and 0.006 µCi (740 Bq). Such rounding errors can introduce significant discrepancies in dosimetry calculations. This method also raises a question about which ALI should be used if compliance monitoring is being based on comparison with secondary limits, such as the ALI rather than the primary dose limits.

Where individual-specific data are available, the models should be adjusted. However, the general lack of capability to monitor organ-specific retention for uranium (i.e., content and clearance half-times) makes the use of default models most practical.

Ideally, one should obtain as much bioassay information as possible to determine the intake and track the retention of uranium in the body to reduce the uncertainty associated with the daily variation in the measurements. A regression analysis should be used to fit the measurement values for estimating the initial intake and clearance half-times.

5.9 REFERENCE AND ACTION LEVELS

Reference and action levels are essential to operation of a routine internal dosimetry program. Because a wide range of levels can be defined by various facilities and organizations, this document does not attempt to prescribe particular level titles. As used in this document, reference and action levels are simply workplace or bioassay measurements, or associated calculated doses, at which specific actions occur.

Notification levels based on workplace indicators for reacting to a potential intake are suggested in Table 5-15. The intent of these notification levels is to provide guidance for field response to any potential intake of radioactive material with a potential for a dose commitment that is >100-mrem CEDE. Table 5-16 suggests notification levels to the occupational medicine physician for possible early medical intervention in an internal contamination event. These tables, derived from Carbaugh et al. (1994), are based on general considerations and significant experience with past intakes of radioactive material and, because they are based on field measurements, do not correspond with any exact dose commitment to the worker.

Table 5-15. Uranium Levels for Internal Dosimetry Notification

Indicator	Notification Level
Nasal or mouth smears	Detectable activity
Facial contamination (direct measurement)	200 dpm
Skin breaks or blood smears	Any skin break while handling material other than sealed sources
Head, neck contamination	2,000 dpm
Contamination in respirator	Detectable activity inside respirator after use
Hands, forearms, clothing[a]	10,000 dpm
Airborne radioactivity	Acute intake equivalent to 40 DAC-hours after accounting for respiratory protection factor[b]

(a) Clothing contamination levels apply to exposure without respiratory protection, such as on inner coveralls or personal clothing.

(b)

$$\text{DAC-hours} = \frac{\text{airborne concentration}}{\text{DAC}} \times \text{hours of intake}$$

Table 5-16. Uranium Contamination Levels for Notification of Occupational Medicine Physician

Indicator	Medical Notification Level (dpm)
Nasal or mouth smears	1,000
Facial contamination	25,000
Skin breaks or wounds	100

The decision to administer treatment and the treatment protocol are solely the responsibilities of the physician in charge. The basic principle is that the proposed intervention should do more good than harm (Gerber and Thomas 1992).

Guidelines for the medical intervention of a radionuclide intake can be found in several publications. NCRP Report No. 65 (NCRP 1980) and the joint publication of the Commission on European Communities (CEC) and the DOE Guidebook for the Treatment of Accidental Internal Radionuclide Contamination of Workers (Gerber and Thomas 1992) both contain detailed guidance in intervention and medical procedures useful in mitigating radiation overexposures. The CEC/DOE Guidebook has been based on the ALI for action levels, rather than on CEDE, to overcome the problem of uncertainties in dose per unit intake. The ICRP recommends in Publication 60 a limit of 2-rem/y (20-mSv/y) on effective dose. Thus, the ALIs found in ICRP Publication 61 (1991b) and used in the CEC/DOE Guidebook noted above are those which would provide a CEDE of 2-rem/y instead of current U.S. regulations of 5-rem/y.

Guidance in the CEC/DOE Guidebook can be summarized as follows:

- When the estimated intake is below 1 ALI, treatment should not be considered.
- When the estimated intake is between 1 and 10 times the ALI, treatment should be considered. Under these situations, short-term administration will usually be appropriate, except for intake of materials poorly transported from the lung (class Y).
- When the estimated intake exceeds 10 times the ALI, then extended or protracted treatment should be implemented, except for materials poorly transported from the lung.
- For poorly transported material in the lung, lung lavage is the only recommended treatment, and it is only a consideration for intakes exceeding 100 times the ALI.

Because the dose associated with the ALI in the CEC/DOE Guidebook is 2-rem CEDE and because the upper administrative control level suggested by the RCS is 2 rem, intervention levels of 2 rem and 20 rem might be used for guidance in the manner presented in the CEC/DOE Guidebook:

- When the CEDE estimated intake is below 2 rem, treatment is not generally recommended.

- When the CEDE for an estimated intake is between 2 rem and 20 rem, treatment should be considered. Under these situations, short-term administration will usually be appropriate.

- When the CEDE equivalent for an estimated intake exceeds 20 rem, then extended or protracted treatment is strongly recommended, except for poorly transported material in the lung.

A useful method to enhance excretion of uranium via the kidneys is the formation of radionuclide complexes using sodium bicarbonate. This type of complexation appears to be the only current method that has a reasonable chance of reducing or preventing kidney damage during the early period after incorporation of this chemotoxic heavy metal.

An initial prophylactic chelation therapy may be appropriate because bioassay measurements (particularly urinalysis) cannot usually be completed within the response time required for effective chelation therapy. Urinalysis becomes very helpful following administration of chelation therapy because there is a direct correlation between urinary excretion and dose averted because of uranium excreted. This provides a method of measuring the effectiveness of chelation therapy and determining if it is worthwhile to continue therapy. It is probably that the efficacy of treatment will decrease with continued administration as uranium is removed and the rate of transfer into the systemic compartment decreases.

5.10 RESPONSE TO SUSPECTED INTAKES

Experience has shown that most intakes of uranium are accidental. Uranium facilities and operating procedures are designed to prevent intakes. Nonetheless, it is important for management to prepare for the possibility that workers might receive an intake of uranium--even though the probability of an incident may be very small. Prompt and appropriate action following an accidental intake of uranium will allow for therapeutic measures to be taken to minimize the internal contamination and lessen the potential for harmful effects. The health physicist and medical staff should work closely to ensure that the proper course of action is followed.

All employees suspected of having received an intake of uranium should be referred for special bioassay measurements. Because a fraction of an intake by inhalation may be retained in the nasal passages for a few hours after exposure to airborne radioactive materials, any level of contamination on a nasal swab indicates an intake that should be followed up by a special bioassay measurement program. However, lack of detection on nasal smears cannot be taken as evidence that an intake did not occur either because the nasal passages can be expected to clear very rapidly or, alternatively, because the worker could be a mouth-breather. Special bioassay should also be initiated if uranium contamination is found on the worker in the vicinity of nose or mouth.

Developing specific field criteria to identify the need for medical response can be challenging. Inhalation intake estimations based on DAC-hours 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 5-17. Another method is to develop field observation criteria (e.g., nasal smear or skin contamination criteria) which might imply that an action level has been exceeded. This latter approach is highly subjective with any number chosen likely to be arguable. Knowledge of facility operations, material forms, and past experience will likely play a key role in development of such criteria.

For acute intakes, direct bioassay measurements should be taken before, during, and after the period of rapid clearance of activity. Urine and fecal samples collected after known or suspected inhalation incidents should also be used to estimate the magnitude of the intake. Initial assessments of intakes from contaminated wounds are based primarily on wound count and urinalysis data.

If a significant intake is indicated, the worker should not return to further potential exposure to uranium until the intake has been thoroughly assessed and a predictable bioassay pattern established. This is particularly important because a new intake of a very low level may confound the interpretation of bioassay measurements for previous intakes of uranium.

Table 5-17. Early Bioassay Measurement Results Corresponding to the Therapeutic Intervention Action Levels Used at the Hanford Site (Carbaugh et al., 1995)

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

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

The health physicist must make important decisions for prompt action at the site of an accidental or suspected intake of uranium or other radioactive materials. Often, these decisions must be based on limited data. Information that may be available for initially estimating the amount and type of intake may include the following:

- levels of measured contamination in the work area,
- skin contamination levels, affected areas, and whether the skin is damaged or punctured,
- wound contamination levels,
- chemical form of the material involved,
- results of air monitoring,
- nasal smear activity levels, and
- sputum and/or mouth contamination.

The special bioassay monitoring program is initiated following a known or suspected intake. This information is needed for dose assessment and future exposure management. The intake is confirmed if follow-up bioassay measurements indicate positive measurement results. Additional bioassay measurements may be needed to quantify the intake and provide data for determining the effective dose

equivalent. The frequency of bioassay monitoring will depend on the specific case to be evaluated. Selection of the appropriate sampling frequency is based on the previously discussed performance capabilities for workplace monitoring programs, consultations with internal dosimetry specialists, and the cooperation of the affected employee.

5.10.1 Emergency Action Planning

The management at the uranium facility should be prepared to follow an emergency action plan for response to an uranium intake. If a worker accidentally inhales or ingests uranium or is injured by a uranium-contaminated object, the action plan should be initiated immediately. A rapid response is important because any delay in implementing appropriate action could lessen the effectiveness of decorporation therapy and increase the probability for internalized uranium to deposit in the kidneys or on bone surfaces.

5.10.2 Medical Emergency Response Plan

The health physicist and medical staff must establish an emergency action plan for the appropriate medical management of an accidental intake of uranium. The elements of the plan should include the following:

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

The sequence and priority of the emergency action plan may vary with the magnitude and type of accidental conditions and their severity. An initial early assessment of the incident should focus, first, on treatment of life-threatening physical injuries and, second, on the radioactive contamination involved. Minor injuries should be treated after decontamination.

A rapid estimate of the amount of internal contamination by uranium or other alpha-emitters may not be possible. If a significant intake (meaning one that exceeds 10 times the ALI) is suspected, medical staff should proceed with decorporation therapy after first treating major injuries.

5.10.3 Responsibilities for Management of Internal Contamination

Responsibilities should be assigned for action in response to an accidental internal uranium contamination. The affected worker has the responsibility to inform the health physicist, radiological control technician (RCT), or his immediate supervisor as soon as an intake is suspected. The health physicist or RCT should make an initial survey of the extent of the contamination and immediately contact his supervisor and, when action levels are exceeded, contact a member of the medical staff. Monitoring and radiation safety support to the medical staff and supervisors should continue during the

management of the contamination incident. Care should be taken to limit the spread of radioactive contamination.

The health physicist should immediately begin to gather data on the time and extent of the incident. Contamination survey results should be recorded. Radionuclide identity, chemical form, and solubility classification should be determined. Nasal smears should be obtained immediately if an intake by inhalation is suspected. When action levels are exceeded, all urine and feces should be collected and labeled for analysis. Decontamination should proceed with the assistance of the medical staff. Contaminated clothing and other objects should be saved for later analysis.

5.10.4 Immediate Medical Care

The medical staff should provide immediate emergency medical care for serious injuries to preserve the life and well-being of the affected worker. Minor injuries may await medical treatment until after an initial radiation survey is completed and the spread of contamination is controlled. However, the individual should be removed from the contaminated radiation area as soon as possible. Chemical contamination and acids should be washed immediately from the skin to prevent serious burns and reactions.

A chelating agent should be administered immediately following an accidental intake of uranium. Sodium bicarbonate should be available for treating internal uranium contamination. The worker to be treated must first be informed of the proposed use of a chelating agent, instructed on the purpose of administering the chelating agent, and warned about the possible side-effects. The worker must then give signed consent before chelation therapy may be initiated. Even though sodium bicarbonate therapy is the only method available for reducing the quantity of uranium retained in the body, the affected worker has the right to refuse its use.

The recommended therapy for decorporation is a systemic administration of 250 mL of isotonic (1.4%) solution of sodium bicarbonate by slow intravenous injection (Gerber and Thomas, 1992). The sodium bicarbonate reacts with uranyl ions, UO_2^{++}, in body fluids to form an anionic complex, probably $UO_2(CO_3)_3$, which is rapidly excreted in urine. Treatment may be continued if bioassay indicates that decorporation therapy continues to enhance the urinary excretion of uranium. However, if treatment is extended over the days following the incident, the dosage should be adapted to prevent contraindications of alkalosis (bicarbonate solution is alkaline) and respiratory acidosis (Gerber and Thomas 1992).

5.10.5 Contaminated Wounds

Medical treatment for contaminated wounds may include flushing with saline and decorporating solutions, debridement, and surgical excision of the wound. These measures are all the responsibility of trained medical staff operating under the direction of a physician. Radiological control personnel can provide valuable assistance by prompt assessment of materials removed from the wound and identification of magnitude of residual activity as decontamination proceeds. Decontamination should continue until all radioactivity has been removed or until risk of permanent physical impairment is reached.

6.0 EXTERNAL DOSIMETRY

The external dosimetry program is an integral part of the external dose control program. DOE G 441.1-4, External Dosimetry Program Guide, provides detailed guidance for implementing an external dosimetry program that meets the requirements of 10 CFR 835. The reference section of that Guide lists specific documents applicable to external dosimetry. Because the requirements and recommendations are explicitly given in these documents, they will not be discussed in any great detail in this chapter. Rather, the emphasis will be on items that are unique to uranium facilities and the radiological aspects for safe handling of uranium.

Measuring the external radiation exposure and the resultant dose is complicated by the many radiations involved in uranium handling. Chapter 2 of this Technical Standard discusses the radioactive decay schemes for and radiations emitted by the uranium isotopes and their radioactive daughter products. Uranium has a wide distribution of beta and gamma energies, with a 2.29-MeV beta as the most significant of these. The dose rate from photons is relatively low. Uranium also emits alpha particles that may generate ~2 MeV neutrons as a result of interactions with the nuclei of fluorine or other low-Z atoms. The magnitude of the neutron fluence depends on the enrichment of the uranium and on the interacting chemical.

The elements of the external dose control program are: detection and characterization of the beta, gamma, and neutron radiation fields; measurement and quantification of these fields; measurement of personnel dose; and determination and establishment of dose control practices.

6.1 DOSE LIMITS

10 CFR 835 specifies the applicable limits used for control of external radiations. Table 6-1 lists the appropriate depths in tissue for measurement of doses to the whole body, lens of the eye, "unlimited areas of skin," and extremities.

Table 6-1. Effective Depth of Tissue for Various Organs

	Depth of tissue, mg/cm^2
Deep (penetrating)	1000
Lens of eye	300
Shallow (skin, extremities)	7

6.1.1 Limiting Quantities

In 1977, the ICRP introduced a major revision in recommended radiation protection practice with the introduction of ICRP Publication 26 (ICRP 1977). The new methodology establishes a "risk-based" system of dose limitation. The ICRP introduced the terms *stochastic* and *nonstochastic* for radiation effects and set limits for both types of effect. Stochastic effect is defined as one for which the probability of the effect occurring (as opposed to the degree or severity of effect) is a function of radiation dose. Nonstochastic effect is defined as one for which the severity of the effect is a function of the dose; a threshold may exist. Limits were established such that the risk of stochastic effects occurring was equivalent to about the same risks faced by workers in "safe" industries who were not occupationally

exposed to radiation in the workplace. Limits were also established for nonstochastic effects that prevented these effects from occurring even if the exposure occurred at the annual limit over the lifetime of the worker.

For stochastic effects, the ICRP specified in Publication 26 that radiation exposure be limited by the *effective dose equivalent,* H_E, which can be expressed by the relation:

$$H_E = \Sigma w_T D_T Q_T \quad (6.1)$$

where: $\Sigma w_T = 1$
w_T = tissue weighing factor for the relevant organ or tissue T
D_T = absorbed dose in the tissue or organ of interest
Q_T = the quality factor averaged over the tissue or organ of interest.

Table 6-2 lists the weighing factors, taken from 10 CFR 835. Effective dose equivalent has the benefit that it is additive, and internal and external radiations can be added numerically to derive an overall estimate of risk.

Table 6-2. Tissue Weighing Factors

Tissue or Organ	Tissue Weighing Factor, w_T
Gonads	0.25
Breast	0.15
Bone marrow (red)	0.12
Lungs	0.12
Thyroid	0.03
Bone surfaces	0.03
Remainder[(a)]	0.30
Whole body[(b)]	1.00

(a) Remainder means the five other organs or tissues with the highest dose (e.g., liver, kidney, spleen, thymus, adrenal, pancreas, stomach, small intestine, and upper large intestine). The weighing factor for each remaining organ is 0.06.
(b) For the case of uniform external irradiation of the whole body, a weighing factor equal to 1 may be used in determining the effective dose equivalent.

The methodology of ICRP-26 has been incorporated into 10 CFR 835. Table 6-3 lists the annual radiation dose limits for DOE activities. However, DOE contractors usually establish lower annual administrative control levels, typically 500 mrem/year or less.

In practice, it is difficult to measure the effective dose equivalents specified in Table 6-3 because it is necessary to know not only the type of radiation but also its energy and direction. If the flux, energy, and direction of incidence are known, it is possible to calculate effective dose equivalent using fluence to effective dose equivalent conversion coefficients presented in ICRP Publication 51 (ICRP 1987), which presents the effective dose equivalent as a function of energy for various irradiation

geometries. Conversion coefficients for mono-directional beams of neutrons can be found in an article by Stewart (Stewart et al. 1994). Conversion coefficients for photons in various irradiation geometries, including planar sources, can be found in a report by Zankl (Zankl et al. 1994). This approach will provide more accurate values of effective dose equivalent, as opposed to numerically setting the value of effective dose equivalent equal to dose equivalent.

Table 6-3. Radiation Dose Limits for DOE and DOE Contractors

Type of Radiation Exposure	Limit
Occupational Exposures of Adults	
Stochastic Effects	5-rem total per year (sum of effective dose equivalent from external exposures and CEDE received during year)
Non-Stochastic Effects	
Lens of eye	15-rem dose equivalent per year
Extremity	50-rem dose equivalent per year
Skin	50-rem dose equivalent per year
Individual organ or tissue	50-rem dose equivalent per year (sum of dose equivalent from external exposures and CDE received during the year)
Occupational Exposures of Minors	
Stochastic Effects	0.1-rem per year (sum of effective dose equivalent from external exposures and CEDE received during year)
Non-Stochastic Effects (Lens of eye, extremity, skin, individual organ or tissue)	10% of occupational dose limits for adults
Embryo/fetus of a Declared Pregnant Worker	0.5-rem effective dose equivalent per gestation period
Planned Special Exposure	Same as routine occupational dose limits in a year (but accounted for separately) 5 times the routine occupational dose limits over an individual's lifetime

6.1.2 Operational Quantities

Because of the difficulties in determining effective dose equivalent from direct measurements, the concept of *operational quantities* has been introduced to be more closely related to measurable quantities. Operational quantities include *ambient dose equivalent* used for area monitoring and *personal dose equivalent* used for personnel dosimetry. Operational quantities are designed to be a conservative estimator of effective dose equivalent, i.e., the values of the operational quantities will be equal to or higher than the effective dose equivalent specified for the limiting quantities.

The ambient dose equivalent, $H^*(d)$, is the dose equivalent at a depth, d, in a 30-cm-diameter sphere of tissue, where: a) the radiation field has the same fluence and energy distribution as the point of reference for the measurement; and b) the fluence is unidirectional (i.e., the sphere can be viewed as being in an aligned radiation field). Most survey instruments are designed to measure ambient dose

equivalent, and international standards are based on the ambient dose equivalent concept. The depth of interest is typically 1 cm of soft tissue, as specified in 10 CFR 835.

The personal dose equivalent, $H_p(d)$, is the dose equivalent in soft tissue at the appropriate depth, d, below a specified point on the body. Obviously, personnel dosimeters should be calibrated in terms of personal dose equivalent.

In reality, most instruments and personnel dosimeters used at DOE facilities are calibrated in terms of dose equivalent. For example, consider the case in which personnel neutron dosimeters are calibrated on acrylic plastic phantoms at a specified distance from a calibrated neutron source. For DOELAP testing, the dose equivalent at this point has been calculated in accordance with NBS Special Publication 633, Procedures for Calibration of Neutron Personnel Dosimeters. These calculations are based on the Grundl-Eisenhauer energy spectrum and the conversion coefficients from NCRP Report 38 (NCRP 1971), which are for the "old" values of dose equivalent from cylindrical phantom calculations.

In most instances, the present methods based on dose equivalent overestimate effective dose equivalent. In cases where personnel are approaching dose limits, it may be prudent to more accurately evaluate effective dose equivalent using special calibrations. Depending on the irradiation geometry and energy, effective dose equivalent may be as much as a factor of two less than dose equivalent.

6.2 RADIATIONS IN URANIUM FACILITIES

As outlined in Section 2.0 of this TS, the uranium isotopes are primarily alpha-emitters and their progeny emit a wide variety of radiations, including alpha and beta particles, as well as more penetrating x rays and gamma rays. Alpha-neutron interactions (and the small cross-section for spontaneous fission) add the potential for neutron exposure to the radiation mix. This section outlines methods to calculate the dose equivalents from radiations emitted by uranium and its progeny. Examples of measured dose rates are also included.

The design of an external dose control program, including instrument and dosimeter selection, is dependent upon the type and intensity of the radiation fields to which the workers will be exposed. Many factors can affect the radiation field:

- enrichment (mix of uranium isotopes),
- emissions from parent radionuclide(s),
- emissions from daughter radionuclide(s),
- emissions from impurity radionuclide(s),
- type of radiation emitted (beta, gamma, etc.),
- energies of emitted radiation,
- specific activity of the source material,
- self-shielding of source material,
- shielding provided by process equipment,
- shielding provided by protective clothing, or
- distance and geometry factors.

The ratio of uranium isotopes in a specific process (a function of enrichment) will determine the source term by which the radiation fields can be predicted. This mix of uranium isotopes and daughter radionuclides may be estimated by using an equation developed to predict specific activity as a function

of enrichment. Figure 2-2 (Chapter 2) shows the estimated activities of the uranium isotopes as a function of enrichment as predicted by the reference equation.

Radiation fields from uranium are frequently dominated by contributions from daughter product or impurity radionuclides. For example, nearly all of the beta radiation field from depleted uranium comes from the daughter radionuclide ^{234m}Pa, and to a lesser extent from ^{234}Th. During melting and casting operations, these daughter elements may concentrate on the surface of the castings and equipment, producing beta radiation fields up to 20 rad per hour.

Figure 6-1. Beta Radiation Readings at Surface of Uranium Metal vs. % Enrichment by Weight

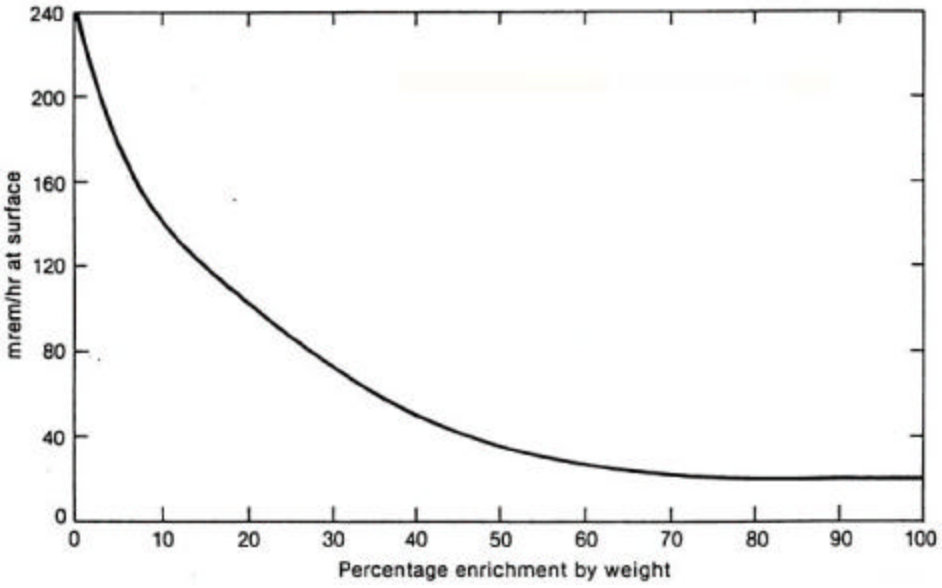


Figure 6-2. Absorbed Dose Rate as a Function of Depth in Mylar

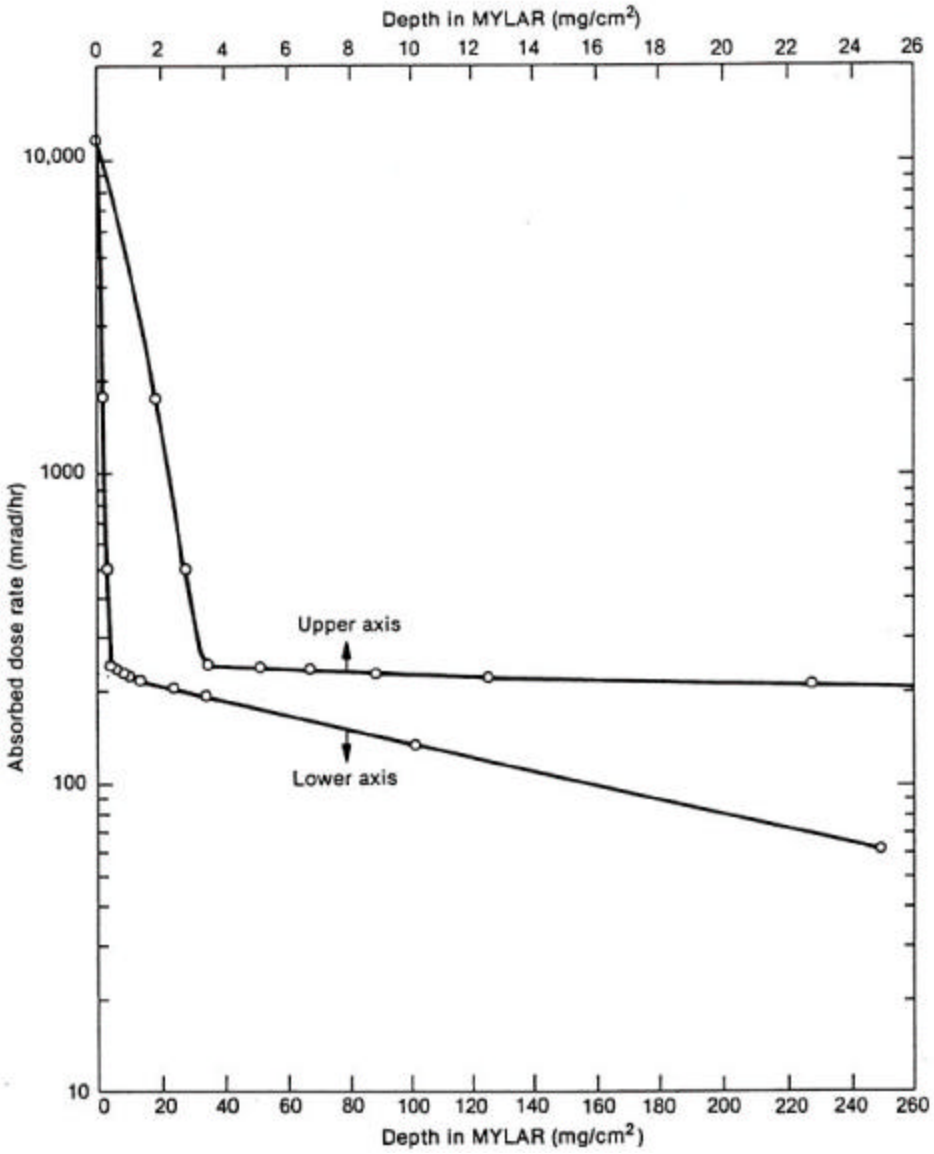
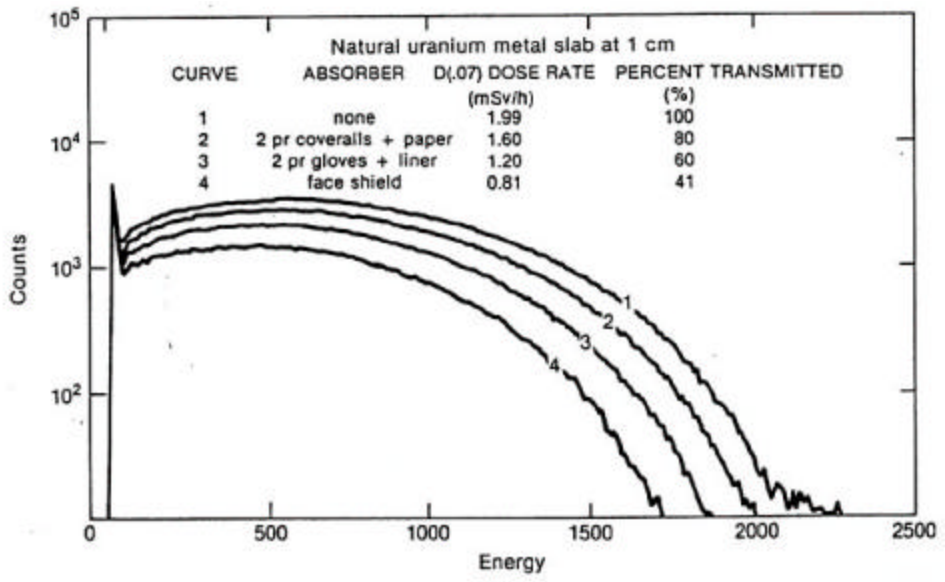


Figure 6-3. Changes in Beta Energy Spectra and Shallow Dose Rate From a Natural Uranium Metal Slab Source Caused by Protective Apparel (Note the bremsstrahlung peak in the low-energy ranges.)



6.2.1 Alpha and Beta Doses

Uranium is an alpha-emitter and is of concern if inhaled or ingested into the body. However, the skin is an effective barrier to alpha particles, and alpha radiation emitted from external contamination is only a problem if there is a wound or break in the skin.

Beta radiation fields are usually the dominant external radiation hazard in facilities requiring contact work with unshielded forms of uranium. Figure 6-1 gives the estimated beta dose rates from a semi-infinite slab of uranium metal of various enrichments. For uranium enrichments up to 30%, the beta radiation field is dominated by contributions from ^{238}U decay products. Thus, for uranium of these enrichments, one is dealing essentially with 2.29-MeV (E_{max}) beta particles from ^{234m}Pa, the most energetic contributor to the beta exposure.

Beta doses to the skin, extremities, and (sometimes) the lens of the eye can be limiting in facilities that process unshielded depleted, natural, or low-enrichment uranium. Absorbed dose rates as a function of depth were measured by P. Plato (Plato 1979) with an extrapolation chamber in a tissue equivalent medium (Mylar) (See Figure 6-2). Skin doses at less than 4 mg/cm^2 resulting from alpha particles are of no concern from an external radiation exposure standpoint. Potentially significant skin exposure from uranium occurs primarily from the ^{234m}Pa betas at tissue depths of 4 mg/cm^2 and greater.

Processes that separate and sometimes concentrate beta-emitting uranium daughters are not uncommon in DOE uranium facilities. Surface beta dose rates on the order of 1 to 20 rad per hour have been observed in such circumstances. Exposure control is complicated by the fact that considerable contact work takes place in facilities that process uranium metal. Beta particles are shielded by rubber gloves or other protective devices or are usually absorbed within the dead layer of skin. The actual beta dose to live tissue would depend on the energy of the beta particles and the thickness and types of intervening shielding.

The data in Figure 6-3 were obtained with a tissue equivalent plastic scintillation detector and demonstrate the spectral changes and the resultant exposure rates under typical protective clothing. It can be seen from Figures 6-2 and 6-3 that significant fractions of the uranium beta radiation will penetrate typical protective clothing worn in facilities which process uranium.

6.2.2 Gamma Doses

Gamma radiation from uranium is normally not the controlling challenge to radiation protection. For example, the contact beta radiation field from depleted uranium is approximately 240 mrem/h, while the contact gamma radiation field is less than 10 mrem/h. Although gamma radiation fields from uranium are not usually the dominant concern, significant gamma fields can exist in areas where large quantities of uranium are stored. Bremsstrahlung from the 2.29 MeV ^{234m}Pa beta can contribute up to 40% of the photon dose from uranium metal. Neutron fields from enriched uranium fluoride compounds can also add to this area of concern. Care should be taken that dose-equivalents from such fields are kept to levels that are ALARA.

Although beta radiation fields from unshielded uranium tend to present the most intense radiation problem, storage of large quantities of uranium can create widespread, low-level (<5 mrem/h) gamma radiation fields. Such fields can create ALARA problems--particularly when significant numbers of people must work in adjacent areas.

6.2.3 Neutron Dose Equivalents

In uranium processes that create fluoride compounds (UF_4, UF_6, etc.), the a-n reaction with this light nuclide can result in neutron radiation fields, the intensity of which are a function of the compound, mixing, storage configuration, and enrichment. As indicated in Section 2.0, low enriched UF_6 (< 5%) in large storage containers can result in neutron radiation in the 0.2 mrem/h range, while highly enriched (> 97%) UF_6 can create fields in the 4 mrem/h range. At high enrichments, the neutron fields can be up to a factor of 2 higher than the gamma fields and be the limiting source of whole body exposure. Neutron radiation from uranium metals and low enriched compounds is considerably lower than the gamma component and, consequently, is not limiting.

Neutron dose equivalent rates can be calculated accurately with computer codes, such as MCNP (Breismeier 1986). The MCNP code has the advantage that it can calculate both neutron and photon doses through shielding and in complex arrays. The Monte Carlo codes can also calculate the effects of neutron multiplication in systems containing large amounts of uranium. However, neutron dose equivalent rates can also be calculated from simple empirical formulas. Unlike gamma doses, there is very little self-shielding for neutrons in sub-kilogram masses of uranium.

Table 6-4 lists spontaneous fission yields for uranium isotopes that may be found in facilities within the DOE complex. These data are taken from NUREG/CR-5550 (NRC 1991) and are believed to be more current then the previously published PNL values (PNL 1988b). As a rule of thumb, nuclides with even numbers of protons and neutrons have the highest spontaneous fission neutron emission rates. The spontaneous fission rate for odd-even nuclides is about 1000 times less, and the rate for odd-odd

nuclides is about 100,000 less. Spontaneous fission neutrons are emitted with a Maxwellian energy distribution given by the equation:

$$N(E) \doteq (\sqrt{E})\exp(E/1.43\ MeV) \tag{6.2}$$

where N(E) is the number of neutrons as a function of the energy E in MeV.

Table 6-4. Spontaneous Fission Neutron Yields

Isotope	Total Half-Life	Spontaneous Fission Half-Life, years	Spontaneous Fission Yield, n/sec-gram
^{232}U	71.7 y	8×10^{13}	1.3
^{233}U	1.59×10^{5} y	1.2×10^{17}	8.6×10^{-4}
^{234}U	2.45×10^{5} y	2.1×10^{16}	5.02×10^{-3}
^{235}U	7.04×10^{8} y	3.5×10^{17}	2.99×10^{-4}
^{236}U	2.34×10^{7} y	1.95×10^{16}	5.49×10^{-3}
^{238}U	4.47×10^{9} y	8.20×10^{15}	1.36×10^{-2}

Energetic alpha particles can overcome coulomb barriers in low-atomic-number elements and create an unstable nucleus that emits neutrons. Because of the high alpha activity of uranium, this can be a significant source of neutrons. There are two nuclear reactions that are of importance:

$$\alpha + {}^{18}O \rightarrow {}^{21}Ne + n \tag{6.3}$$
$$\alpha + {}^{19}F \rightarrow {}^{22}Na + n. \tag{6.4}$$

Table 6-5 lists the alpha-neutron yields for oxides and fluorides for the uranium isotopes. Note that the neutron yields are normalized per gram of nuclide, not per gram of compound. These data are taken from NUREG/CR-5550 (NRC 1991).

The total neutron yield per gram of uranium can be found by summing the contributions from:

- spontaneous fission (from Table 6-4)
- alpha-neutron reactions in oxides or fluorides (from Table 6-5)
- neutrons from low-atomic-number impurities (from Table 6-6).

Multiplying the specific neutron yield (neutrons/second-gram of uranium) by the mass of uranium (grams) gives S, the neutron emission rate (neutrons/second).

Table 6-5. Neutron Yields from Alpha-Neutron Reactions for Oxides and Fluorides

Isotope	Alpha Decay Half-Life	Alpha Yield, alpha/s-g	Average Alpha Energy, MeV	alpha, n Yield in Oxides, n/s-g	alpha, n Yield in Fluorides, n/s-g
^{232}Th	1.41×10^{10} y	4.1×10^{3}	4.00	2.2×10^{-5}	
^{232}U	71.7 y	8.0×10^{11}	5.30	1.49×10^{4}	2.6×10^{6}
^{233}U	1.59×10^{5} y	3.5×10^{8}	4.82	4.8	7.0×10^{2}
^{234}U	2.45×10^{5} y	2.3×10^{8}	4.76	3.0	5.8×10^{2}
^{235}U	7.04×10^{8} y	7.9×10^{4}	4.40	7.1×10^{-4}	0.08
^{236}U	2.34×10^{7} y	2.3×10^{6}	4.48	2.4×10^{-2}	2.9
^{238}U	4.47×10^{9} y	1.2×10^{4}	4.19	8.3×10^{-5}	0.028

Table 6-6. Neutron Yields for Trace Impurities in Uranium

Element	Neutron Yield per 10^6 Alphas at 4.7 MeV (^{234}U)
Li	0.16 ± 0.04
Be	44. ± 4
B	12.4 ± 0.6
C	0.051 ± 0.002
O	0.040 ± 0.001
F	3.1 ± 0.3
Na	0.5 ± 0.5
Mg	0.42 ± 0.03
Al	0.13 ± 0.01
Si	0.028 ± 0.002
Cl	0.01 ± 0.01

6.3 RADIATION DETECTION AND EVALUATION

This section describes the response of portable instruments, personnel dosimeters, and nuclear accident dosimeters to the radiations emitted by uranium, which are primarily alpha and beta particles and photons. Neutron emissions may range from negligible to significant. Data are also included on special spectrometry instruments used to calibrate dosimeters in the field.

6.3.1 Portable Survey Instruments--Beta Radiation Response

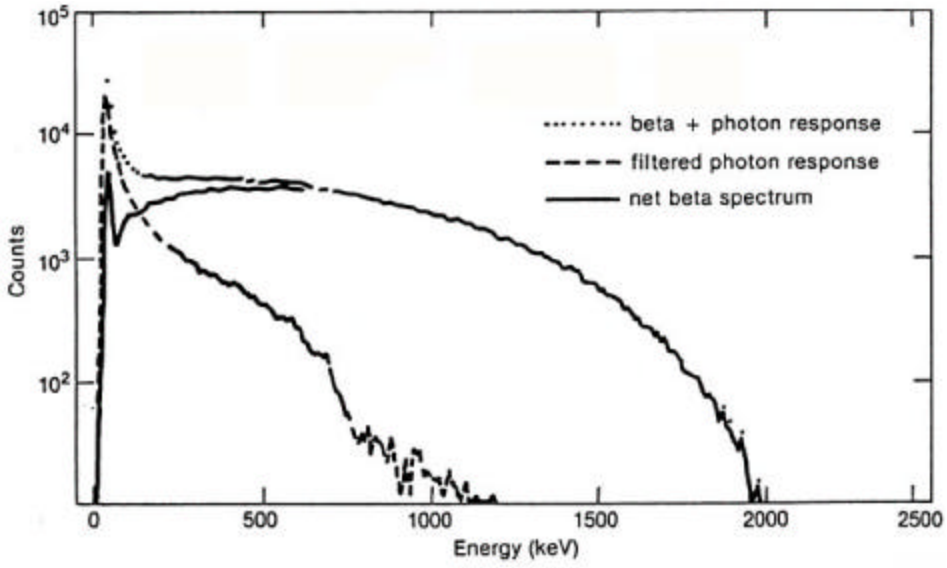
The primary exposures of concern when handling bare uranium materials come from the beta radiation. The accuracy and precision of survey instruments used for measurement of beta radiation fields depend upon some or all of the following factors:

- beta energy response,
- angular response of instrument,
- source-detector geometry factors, and
- detector construction (window thickness, etc.).

6.3.1.1 Energy Dependence

Most commercially-available radiation survey instruments under-respond to beta radiation fields from uranium. Figures 6-4 and 6-5 show the beta and gamma spectra measured with a tissue equivalent plastic scintillation. Table 6-6 presents typical survey instrument response to uranium fields specifically. At best, typical "beta correction factors" (true dose rate/indicated dose rate) are on the order of 1.5 to 2. This under-response is due primarily to a) the angular response of the detector and b) attenuation of the dose-rate by the detector window and the sensitive volume of the detector.

Figure 6-4. Meter Readings for a Depleted Uranium Ingot



6-5. Meter Readings for an Open Drum of UF_4 (green salt)

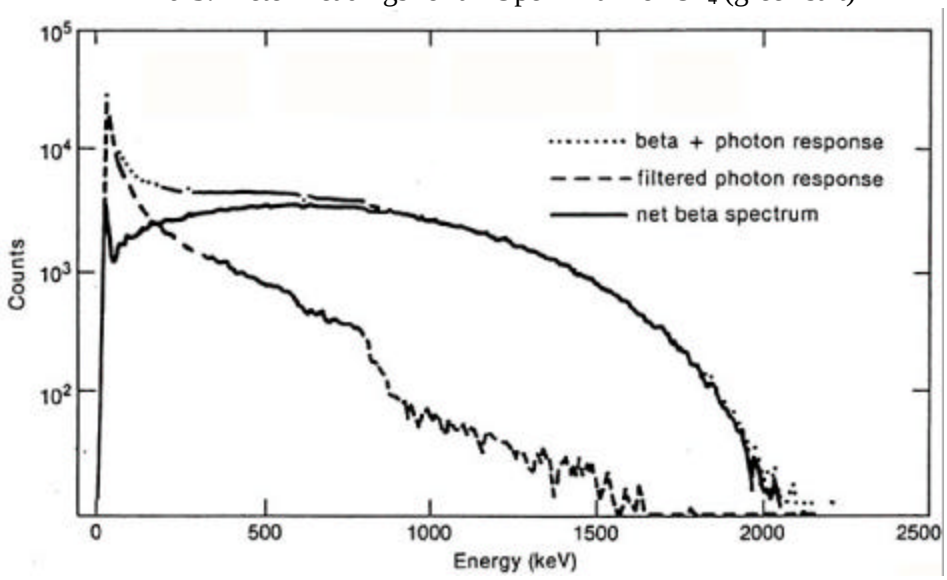


Table 6-7. Instrument Response to Uranium Beta Fields

Instrument	Window, mg/cm²	Beta Correction Factor[a]	Exposure Geometry
Victoreen 471	1.1	1.4	30 cm from U foils
Eberline RO-2	7	2.0	30 cm from U foils
Eberline RO-2A	7	4.0	Contact with DU slab
Aluminum-walled GM Detector	30	1.7	30 cm from U foils
Victoreen Radector III	34	14	Contact with DU slab
HPI-1075	7	1.8	Contact with DU slab
Teletector	30 (low range)	50	Contact with DU slab
Eberline PIC-6A	30	40	Contact with DU slab
British BNL-3	7	1.3	1.5 cm from 100 cm² DU

(a) True reading/measured value.

Currently, skin dose measurements are related to the dose at a depth of 7 mg/cm^2 in tissue. Window thicknesses of commonly available survey instruments typically range from on the order of 7 mg/cm^2 to several hundred mg/cm^2.

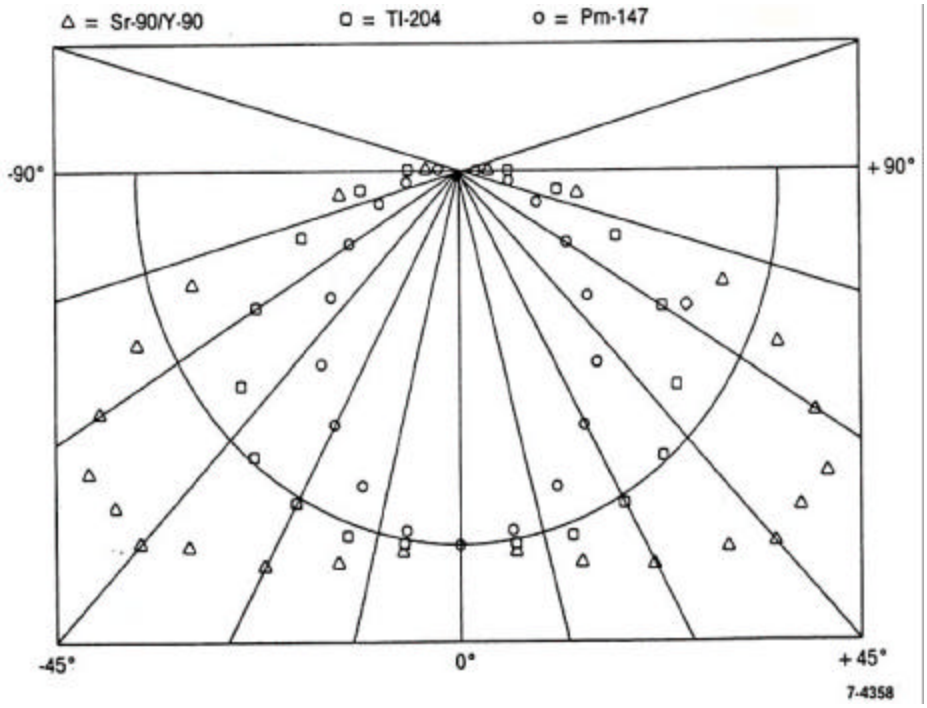
Even if the window provides only minimal attenuation, the attenuation of the beta field through the sensitive volume of large detectors remains a problem. The detector indicates the average dose-rate throughout the sensitive volume. The "true" dose-rate is that which occurs at the plane of the detector incident to the radiation source. The instrument will under-respond by the ratio of this average dose-rate to the incident dose-rate. This sensitive volume under-response is a function of the beta energy distribution and of the size and shape of the sensitive volume.

6.3.1.2 Angular Response

The construction of most survey instruments (e.g., "cutie pie") leads to a severe angular dependence when measuring beta radiation fields. This angular dependence results from the attenuation of the beta field by the walls of the detector as the window is moved away from the source.

Figure 6-6 demonstrates the response of a tissue equivalent response (a 5 mg/cm^2 detector under a 5 mg/cm^2 window and mounted in a TE phantom) to off-axis (non-incident) ^{90}Sr/^{90}Y betas (energies similar to those from uranium). Skin tissue dose response is greater to off-axis betas; survey instruments, which effectively shield these high angle particles, will under-respond compared to skin tissue.

Figure 6-6. Measured Angular Response of the INEL TE Survey Meter to Parallel Beams of Beta Particles From Three Standard Beta Sources



6.3.1.3 Source-Detector Geometry

Measurements taken close to small beta sources may be inaccurate due to non-uniform irradiation of the sensitive volume of the detector. Uranium in most DOE facilities tends to present wide-area sources of beta radiation. However, adjustments would need to be made if significant non-uniform irradiation was encountered.

6.3.1.4 Detector Construction and Use

Characteristics of instrument construction may significantly affect their response and use. For example, many survey instruments have "beta windows" that are intended to discriminate between beta and gamma radiation. Obviously, measurements of beta dose-rate must be made with the beta window open. It should be noted, however, that a number of instruments have beta windows that are only a few hundred mg/cm^2 thick. Such windows can transmit a significant fraction of the dose-rate from high-energy beta-emitters (e.g., ^{234m}Pa). Thus, up to 10% or 20% of the "gamma only" reading may be due to the higher-energy betas penetrating the so-called beta window.

Occasionally, survey instruments are placed in plastic bags or covered to protect them from becoming contaminated. Bagging the instrument places additional absorber between the radiation field and sensitive volume of the detector. Calibration of the instrument (or application of a correction factor) should take this additional shielding into account.

6.3.2 Portable Survey Instruments--Gamma Radiation Response

Although the external dose resulting from gamma and x-ray radiation from bare uranium is a small fraction of the total, it represents the "penetrating" or whole body dose source and is the only source of radiation from contained facilities (i.e., those having glove boxes, etc.). Survey instruments are typically calibrated with ^{137}Cs (0.663-MeV) photons. Typical portable survey instruments demonstrate a fairly flat energy response above 250 keV, while the response below 250 keV can be variable to a greater or lesser degree depending upon the instrument design. Figures 6-7 and 6-8 show average response of a group of commercial survey instruments. Figure 6-9 shows a typical gamma spectrum from a uranium oxide source while Table 6-8 illustrates the wide variation that can occur in the photon spectra at various locations in a single plant. This demonstrates the desirability of using ion chambers or compensated beta instruments for dose-rate measurements. It also indicates the need to have knowledge of the energy response of the instrument used and the value, or at least qualitative knowledge, of the photon spectra at the various work stations.

Figure 6-7. Average Ion Chamber Survey Meter Response by Group to X or Gamma Photon Radiation

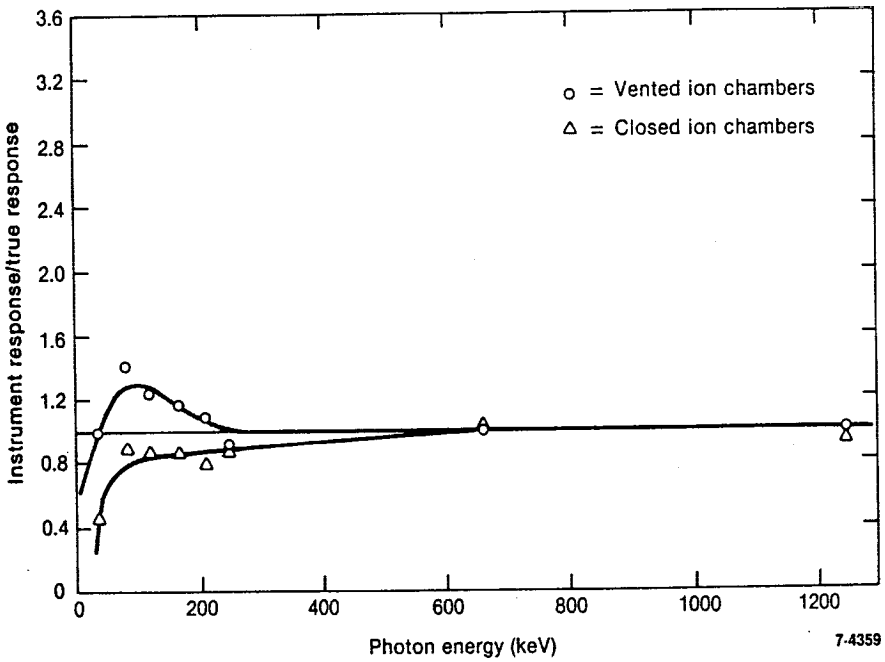


Figure 6-8. Average GM Survey Meter Photon Energy Response by Group

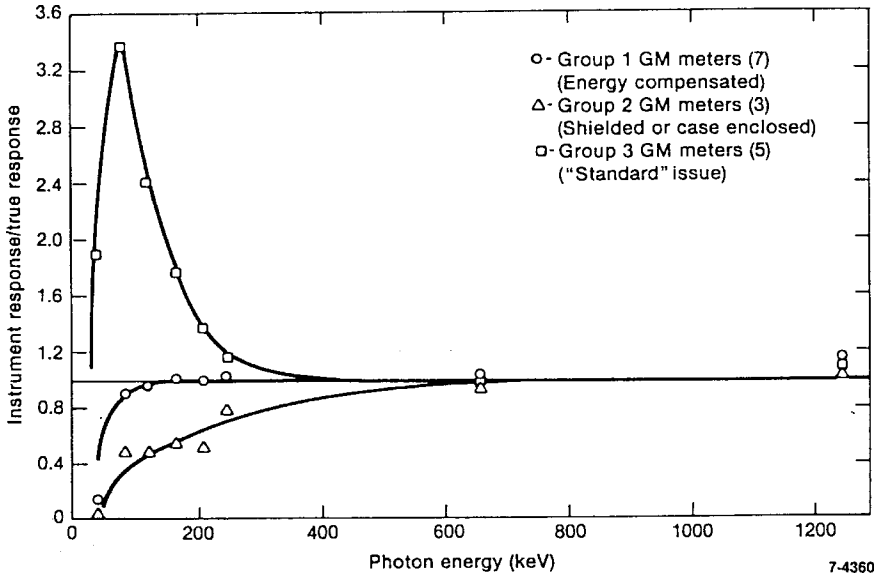


Figure 6-9. High Resolution Gamma Spectrum of Slightly Enriched Uranium Oxide (1% U-235) recorded with Ge (Li) Detector

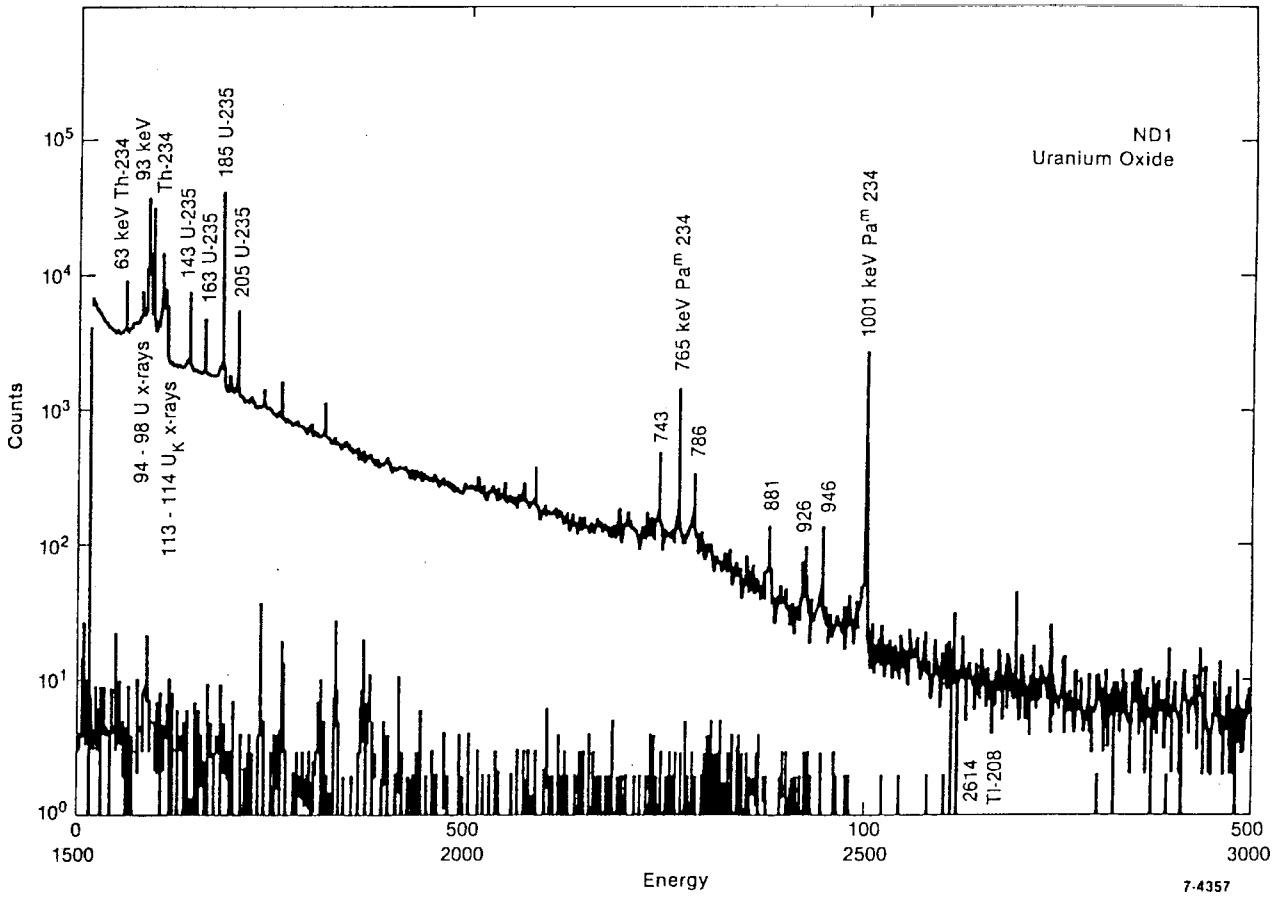


Table 6-8. Gamma Flux and Ratios at Various Locations and Sources at Fernald Plant

	Integrated Gamma Flux, photons/cm^2/sec		
Source Description or Location	**30 to 225 keV**	**675 to 1050 keV**	**Ratio**
Crucible load station - 55-gal. drum	990	348	2.8
Beside UO_3 barrel	538	159	3.4
Open UO_3 barrel	919	232	4.0
Tube-cutting work station, metal	253	58	4.4
Outside Plant 9 south entrance, near exhaust fan	776	165	4.7
Box of black top crop at 25 cm	848	154	5.5
Lathe work station	424	76	5.6
Background outside Building 3045	35	5	7.0
Near "thorium" hopper	424	58	7.3
Plant 9 west wing, SW hot area	708	72	9.8
Crucible burnout station	776	69	11.2
Plant 9 HP change room	5	<0.4	12.5
Background 75 ft from Bldg. 3045	25	2	12.5
Graphite crucible (G-8010) 30 cm	183	11	16.6
Graphite crucible (3898) 30 cm	310	18	17.2

6.3.3 Portable Survey Instruments--Neutron Response

The need for neutron surveys at uranium facilities depends on the quantity of uranium present, its form, and the potential for (a,n) reactions, such as occurs with uranium fluoride. In facilities where such monitoring is required, selection of instruments with appropriate energy characteristics is important because of the energy and angular dependence with most instruments. Fortunately, uranium compounds emit neutrons in the MeV range, where problems with energy and angular dependence are minimal. Calibration with sources that emit neutron energies similar to those in the facility will assist in accurately measuring the radiation fields and selecting appropriate factors in calculating personnel doses.

6.4 PERSONNEL DOSIMETRY

It is important to verify and document that personnel dosimetry systems provide accurate measurements and records of the occupational radiation doses received by workers in uranium facilities. To provide a level of confidence in dosimetry services in DOE facilities, the DOELAP accreditation program has been established. 10 CFR 835 requires participation in the DOELAP program (or specific exceptions or other approvals) by all DOE facilities that are subject to the individual external dose monitoring requirements. Previously, the National Institute of Standards and Technology (NIST) established the National Voluntary Laboratory Accreditation Program (NVLAP) for testing and accreditation of dosimeter processors serving the commercial nuclear power industry and medical facilities. The DOELAP standard includes some tests that differ from those in ANSI N13.11 (ANSI 1983a), on which the NVLAP program is based. Both DOELAP and NVLAP accreditation programs use performance tests that evaluate the accuracy and precision of personnel dosimetry measurements. The accuracy is determined by comparing the measured dose equivalent to the "conventionally true dose equivalent" derived from calibration standards directly traceable to NIST in carefully controlled conditions.

DOE G 441.1-4, External Dosimetry Program Guide, provides detailed guidance for developing and implementing an external dosimetry program that will comply with the requirements of 10 CFR 835. This section will focus on dosimetry problem areas specific to uranium facilities and possible solutions.

Personnel dosimeters produce the data that become the formal or "legal" record of personnel exposure. However, these detectors experience many of the same energy dependence and angular response problems encountered by survey instruments. The most difficult problem is relating badge results to the shallow or skin dose.

Thermoluminescent dosimeters (TLDs), currently the dosimeter of choice in most DOE uranium facilities, provide the most accurate and precise means of measuring doses received by workers. Film badges and nuclear track detectors are other types of dosimeters. Although the following discussion focuses on the more widely used TLD detector systems, the basic principles apply to film badges, with the added uncertainties associated with the increased susceptibility of film to environmental influences, such as temperature, humidity, pressure, etc. Great care is necessary to ensure that the shallow and deep doses are accurately discriminated and measured.

An ideal dosimeter would directly measure doses at 7, 1000, and perhaps 300 mg/cm^2 (shallow, deep, and lens of eye doses). In practice, the dose at such depths in tissue must be inferred from a combination of measurements with different filters. TLD and film elements are mounted in a badge arrangement, which is covered by at least 10 to 30 mg/cm^2 of Mylar, paper, or other covering for mechanical and/or protective reasons.

6.4.1 Energy Dependence

Personnel dosimeters are beta energy-dependent for the same reason that survey instruments are beta energy-dependent. That is, the reading obtained from the dosimeter is proportional to the average rate of energy deposition through the "sensitive volume" or body of the element. If this average energy deposition is less than the deposition at 7 mg/cm^2, then the dosimeter will under-respond.

TLD chips of lithium fluoride (0.32 cm x 0.32 cm) are about 240 mg/cm^2 thick. Significant attenuation of the beta field takes place through the body of the chip. As a result, these types of TLD chips under-respond to uranium decay betas by a factor of about 2.

Other TLD systems minimize this problem by adhering a thin layer of TL powder onto a plastic backing. Current TLD personnel dosimeters typically use multiple detectors (typically, four) under different filter thicknesses. The different responses of each element are used as input to an algorithm which provides an estimate of the effective radiation energy and the doses at depths of interest.

Detectors that are very thin minimize energy-dependence. Film detectors demonstrate a high energy-response dependence for low-energy photons, as well as beta energy-response dependence (though the beta response is less variable than that of TLD chips).

Current systems could potentially provide accurate and precise information; however, their complexity can lead to problems. Calibration of these systems should be performed by a person with specific expertise in the detector's system and knowledge of badge response to high beta or mixed beta and gamma radiation fields.

6.4.2 Angular Dependence

The dosimeter elements must be mounted in a badge or element holder. The assembled badge usually displays severe angular dependence. Fortunately, in most cases, a worker's normal movements will tend to average out some of this dependence. Some badge holder arrangements can flip the badge completely over so that the "beta window" of the badge is facing the worker, not the source. The design of the badge holder or strict administrative controls should be utilized to minimize this problem.

6.4.3 Dosimetry Practices

Beta and gamma fields in working areas should be well-characterized. See previous figures and tables as examples. An attempt should be made to correlate survey instrument and dosimeter badge results. Badge reading frequency should be long enough to accumulate a significant dose (100-mrem range) and short enough to allow adequate control. Dosimeter change frequencies can vary with the specific work-site conditions.

Although multiple badging is not usually necessary, it should be considered for use in very high beta fields produced by separated uranium decay products. The dosimetry system used shall meet or be specifically excepted from DOELAP standards (10 CFR 835.402(b)) and be specifically designated for measuring both shallow and deep doses from uranium.

Dosimetry systems should be capable of providing routine results within a reasonable time period. The system of badge collection and re-distribution should be well defined and minimize the possibility of lost badges.

Badge reading systems should have established "action levels" to alert technicians or operators of unusual results. Such results should include readings or TLD element ratios in excess of certain levels. If possible, the system should automatically save glow curves of any unusual results.

The potential for badge contamination should be minimized. Where the potential for badge contamination exists, badges should be frequently checked for contamination.

6.4.4 Extremity Dosimetry

Doses to the extremities from uranium processing and handling can involve significant exposures to the skin of the hands and forearms. Doses over small areas of the skin, including those from hot particles, are discussed in detail in DOE G 441.1-4 and will not be discussed here.

Measurement of the dose to the hands and/or forearms typically are made with TLD chips or TL powder in finger rings or wrist dosimeters. Such devices do not allow for all of the sophisticated energy discrimination just discussed. The non-homogeneity of beta radiation fields coupled with the angular dependence of commonly-available extremity dosimeters can result in a probability of underestimating the dose. However, by carefully considering the typical exposure conditions at the work site (handling metal pieces, glove box work, etc.) and calibrating the dosimeters with appropriate sources (uranium plaque sources, etc.), extremity doses can be measured with acceptable accuracy for protective purposes.

Care should be exercised in preventing "obvious" underestimations of extremity dose. For example, finger rings worn on the "top" of the finger (opposite the palm side of the hand) will not measure the dose received by the palm side when handling metal rods, etc. Dosimeters worn on the wrist have been shown to underestimate the beta dose to the fingers and palm. Reference to the Bibliography information sources will provide further information in current techniques and considerations.

The general methods used to calibrate dosimeters are given in the National Bureau of Standards Special Publication 633, Procedures for Calibrating Neutron Personnel Dosimeters. Two laboratories conduct the performance test irradiations for the DOELAP and NVLAP programs: Pacific Northwest National Laboratory of Richland, Washington, and the Radiological and Environmental Laboratory (RESL) of Idaho Falls, Idaho. Processors submit dosimeters for testing to the performance testing laboratories in the categories listed in Table 6-9. If the dosimeter processor passes certain accuracy and tolerance testing criteria, a team of dosimetry experts visit the processor and assess the operation of the dosimetry program, including dosimetry records and data retrieval systems, before the dosimeter processor is certified. These requirements are given in DOE STD-1111-98, Department of Energy Laboratory Accreditation Program Administration (DOE 1998b) and its associated guidance documents.

Table 6-9. Performance Test Categories, Radiation Sources, and Test Ranges for the DOELAP and NVLAP Programs

Category	Radiation Source	Test Range
Low-energy photons (high dose)	IST x-ray Beam code M150	0.1 - 50 Gy
High-energy photons (high dose)	^{137}Cs	0.1 - 50 Gy
Low-energy photons (low dose)	NIST x-ray Beam codes: M30 M50[(a)] S60 S75[(a)] M100[(a)] M150 H150[(b)]	0.3 - 100 mSv
High-energy photons (low dose)	^{137}Cs	0.3 - 100 mSv
Low-energy photons (monoenergetic)	15 - 20 keV[(b)] 55 - 65 keV[(b)]	0.3 - 50 mSv
Beta particles	^{204}Tl ^{90}Sr/^{90}Y Natural or depleted uranium (slab)[(b)]	1.5 - 100 mSv 1.5 - 50 mSv
Neutrons	^{252}Cf moderated ^{252}Cf unmoderated[(b)]	2.0 - 50 mSv
Photon mixtures Photon/beta mixtures Photon/neutron mixtures		2.0 - 50 mSv

(a) Category unique to the NVLAP program.
(b) Category unique to the DOELAP program. Note also that ^{241}Am (59-keV photons) may be used in place of the mono-energetic photon (55 - 65 keV) fluorescent x-ray source.

At present, only personnel dosimeters for whole body irradiations are required to be tested, but a DOE working group developed an extremity dosimetry performance testing standard. Extremity dosimeters may be voluntarily tested. DOE also conducts an inter-comparison of calibration sources used for radiation protection purposes, but in the near future DOE secondary calibration laboratories will be established to increase the consistency of radiation protection instrument calibrations to national standards.

There is some question about the correct quality factor to apply to extremity neutron dosimeters. Most quality factors are defined in terms of linear energy transfer (LET), so a numerical value for quality factor can be readily derived by calculation or measurement of the neutron energy spectra. However, the relationship between quality factor and LET was derived from biological experiments on cancer induction, especially leukemia in blood-forming organs. There are no blood-forming organs in the extremities, so there is no biological basis for large values of quality factors for extremity exposures. However, regulatory agencies typically apply quality factors derived for whole-body exposures to the extremities; thus, for compliance purposes, qualify factors should be applied for extremity exposures.

6.4.5 Dose to Lens of Eye

It is sometimes assumed that if the skin dose limit is not exceeded, the dose limit to the lens of the eye will not be exceeded. Such assumptions should be well supported by calculations or (preferably) actual measurements. See Figure 6-3 for data indicating significant uranium beta penetration of even face shields. It is suggested and is a common practice in most fabrication areas to require the use of safety glasses, a practice which tends to mitigate this concern.

6.5 EXTERNAL DOSE CONTROL

Reduction of personnel doses to levels that are ALARA is largely a matter of common sense applied to the principles of time, distance, and shielding. The first step in any dose control program is to adequately identify, characterize, and measure the radiation fields. Only after this step has been performed can optimum dose control be achieved for a given amount of time, money, and energy. However, other considerations may be just as important. Good housekeeping practices are vital to keep dose rates low. Even invisible dust layers on the interior surfaces of glove boxes can increase radiation fields. Storing gloves inside the glove box when not in use and placing lightweight "pie plate" shields over the glove-port openings are examples of practices that can significantly reduce dose rates.

6.5.1 Time

As a general rule, a reduction in exposure time will yield a reduction in doses. Any operation that involves high dose rates (more than a few mrem/hour) or extended exposures should be reviewed for possible reductions in a worker's exposure time. Traffic and material flow in proposed facilities should be closely examined for opportunities to reduce exposure time.

6.5.2 Distance

Beta dose rates from uranium and its decay products decrease rapidly with distance from the source due to geometry and air shielding while gamma and neutron radiation decrease less with distance due to scattering buildup. Because uranium facilities usually involve a high percentage of contact work, considerable dose reduction can result from simple techniques to make operations semi-remote and allow workers to function. Even short distances can effect significant dose reductions.

6.5.3 Shielding

Shielding is probably the most widely used (and most effective) method of reducing beta doses from uranium. Relatively lightweight, cheap, and flexible shielding (e.g., plastic or rubber mats) has been used effectively. Figure 6-3 demonstrates the spectral basis for shielding and lists a few protective clothing reduction factors. Table 6-10 lists the thicknesses of common shielding materials necessary to stop essentially all of the beta particles from uranium (i.e., ^{234m}Pa). Generally, the less dense shielding materials are used whenever possible to eliminate bremsstrahlung as well as beta radiation fields.

Protective clothing commonly worn in the nuclear industry can also afford beta dose reduction. Figure 6-3 and Table 6-11 list approximate dose reduction factors provided by such clothing. Particular attention should be paid to the use of gloves for "hands-on" work. Although lightweight rubber gloves provide some reduction, consideration should be given to using heavy leather or even leaded gloves for operations that do not require manual dexterity. Such gloves can be particularly effective in handling materials emitting high beta fields from unsupported uranium decay products.

Table 6-10. Uranium Beta Shielding

Material	Approximate Material Thickness Required to Stop ^{234m}Pa Betas, cm
Air	850
Aluminum	0.41
Lead	0.10
Lucite	0.92
Pyrex Glass	0.49
Polyethylene	1.2
Stainless Steel (347)	0.14
Water	1.1
Wood	1.7 (approx.)
Uranium	0.06

Table 6-11. Uranium Beta Dose Reduction Factors

Item	Fraction of Beta Dose Remaining
Vinyl surgeon's gloves	0.95
Latex surgeon's gloves	0.87
Lead loaded, 10-mil lead equivalent	0.77
Lead-loaded, 30-mil lead equivalent	0.13
Pylox gloves	0.62
Leather, medium weight	0.62
White cotton gloves	0.89
"Tyvek" coveralls	0.98
"Durafab" paper lab coat	0.96
65% Dacron/35% cotton lab coat	0.91

Contamination build-up inside of work gloves has led to unacceptable hand doses in some facilities. Re-use of leather or cloth gloves should be reviewed carefully for such build-up. Workers should wear thin, anti-contamination gloves inside the heavy gloves.

Dose to the lens of the eye can be effectively reduced through the use of ordinary glasses, safety glasses, or face shields. Such eye protection should be required when workers are dealing with the high beta fields from concentrated uranium decay products.

6.5.4 Geometry

The beta radiation field from uranium is strictly a surface phenomenon. Dose reduction programs can take advantage of this fact in some circumstances. For example, large plates or sheets of uranium metal, if stored in racks 'ledge on," will present less of a beta (and gamma) radiation field.

6.6 RECORDKEEPING

10 CFR 835 establishes specific requirements for maintenance of records associated with area and individual monitoring. DOE G 441.1-11, Occupational Radiation Protection Record-keeping and Reporting Guide, and the RCS provide guidance for achieving compliance with these requirements. There are no occupational radiation protection recordkeeping requirements that are unique to uranium facilities.

7.0 NUCLEAR CRITICALITY SAFETY

This chapter emphasizes present-day criticality concerns from the standpoint of what nuclear criticality safety and radiological control personnel in a uranium facility need to know for the DOE mission to be accomplished in a safe and cost-effective manner. It provides an overview of the administrative and technical elements of current nuclear criticality safety programs. It does not provide a definitive discourse on nuclear criticality safety principles or repeat existing guidance. For radiological control personnel who require a greater understanding of nuclear criticality safety, the listed references provide a source of detailed requirements and information.

Health physicists and other radiation protection personnel have the technical responsibility to understand nuclear principles and the impact of these principles, in the form of the radiological conditions that exist in DOE facilities as the result of the processing, handling, and storage of radioactive and/or fissile materials. Radiation protection personnel provide an additional knowledgeable resource to help recognize workplace situations that might lead to the violation of a nuclear criticality control parameter that could contribute to an inadvertent nuclear criticality event. There have been occasions in which radiation protection personnel have observed and stopped unsafe actions by facility personnel that, if allowed to continue, might have resulted in a nuclear criticality accident. Radiation protection personnel must also be aware of the potential impacts of their actions that would be viewed as routine for normal radiation protection practice, but which could result in the violation of a nuclear criticality safety control parameter. Finally, radiation protection personnel are the focus of emergency response actions should an inadvertent nuclear criticality occur. These actions include use of emergency instrumentation, accident dosimetry, radiological dose assessment, and recovery.

This section reviews 1) nuclear criticality safety regulations and standards, including TSARS, applicable to DOE facilities, 2) criticality control factors, 3) past criticality accidents and associated lessons learned, 4) roles, responsibilities, and authorities of radiological control staff with regard to nuclear criticality safety, and 5) the content of an acceptable nuclear criticality safety program.

7.1 REGULATIONS AND STANDARDS

Nuclear criticality safety program requirements for DOE facilities are presented in DOE O 420.1A, Facility Safety (DOE 2002). There are two objectives for nuclear criticality safety in the Order: 1) nuclear criticality safety is comprehensively addressed and receives an objective review, with all identifiable risks reduced to acceptably low levels and management authorization of the operation documented, and 2) the public, workers, property, both government and private, the environment, and essential operations are protected from the effects of a criticality accident.

The following standards of the American Nuclear Society provide recommendations for criticality operations, alarms, storage of fissionable materials, programs, training and documentation:

a. ANSI/ANS-8.1, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (ANSI 1983b). This standard provides the basic criteria and limits for operations with fissionable materials outside reactors except for critical experiments. It also provides requirements for establishing the validity and the areas of applicability for any calculational method used in assessing nuclear criticality safety.

b. ANSI/ANS-8.3, Criticality Accident Alarm System. This standard provides the performance criteria for the location, selection, design, operation, and testing of nuclear criticality detection

and alarm systems. Paragraphs 4.1.2, 4.2.1 and 4.2.2 shall be followed as modified in Section 4.3.3.c and e of DOE 420.1. ANSI/ANS- 8.7 - 1975, R87, Guide for Nuclear Criticality Safety in the Storage of Fissile Materials and ANSI/ANS- 8.15 - 1981, R87, Criticality Safety Control of Special Actinide Elements provide additional guidance.

c. ANSI/ANS-8.19, Administrative Practices for Nuclear Criticality Safety (ANSI 1984). This standard provides the elements of an acceptable nuclear criticality safety program for operations outside reactors.

d. ANSI/ANS-8.20, Nuclear Criticality Safety Training (ANSI 1991). This standard provides the criteria for the administration of a nuclear criticality safety training program for personnel who manage, work in, or work near facilities, or work outside of reactors, where the potential exists for nuclear criticality accidents. This standard does not meet the training needs of nuclear criticality safety personnel.

e. ANSI/ANS-10.3, Guidelines for the Documentation of Digital Computer Programs (ANSI 1986b). This standard presents guidelines for documenting computer codes (i.e., user documentation) for engineering and scientific applications.

f. ANSI/ANS-10.4, Guidelines for the Verification and Validation of Scientific and Engineering Computer Programs for the Nuclear Industry (ANSI 1987c). The objective of this standard is to identify processes that will enhance the reliability of computer codes used in the nuclear industry and reduce the risk of incorrect application.

7.2 CRITICALITY CONTROL FACTORS

As noted in ANSI/ANS-8.1, the critical mass is a function of the radionuclides in the material as well as its density, chemical and physical form, shape, and surroundings (i.e., moderators, reflectors, neutron absorbers). Nuclear criticality safety is achieved by controlling the quantity and distribution of fissionable materials and other materials capable of sustaining a chain reaction and the quantities, distributions, and nuclear properties of all other materials with which fissionable materials are associated. For new facilities, DOE requires that design considerations for establishing the controls shall include mass, density, geometry, moderation, reflection, interaction, material types, and nuclear poisons (neutron absorbers). Passive engineered controls such as geometry control is the preferred method. The use of administrative controls is to be minimized.

Nuclear criticality control factors can be classified as engineered (e.g., geometry controls and volume controls) or administrative (e.g., mass limits and operating procedures).

7.2.1 Controllable Factors

Some of the criticality safety controls used to prevent a nuclear criticality accident are described below.

7.2.1.1 ^{235}U Enrichment

Enriched uranium is normally required to provide sufficient fissionable material to sustain a critical or sustained nuclear reaction in a small enough mass to meet the needs of the system. Handling of natural (0.7% ^{235}U) or depleted (<0.2% ^{235}U) uranium is generally safe at DOE uranium-processing

facilities because deliberate engineering efforts, such as moderation with heavy water, reactor-grade graphite, etc., are required to create a critical mass with natural uranium. However, safe-handling measures should always be observed when handling uranium of any enrichment.

7.2.1.2 Mass

The minimum mass of uranium that will sustain a chain reaction under specified conditions is called the minimum critical mass. The minimum critical mass depends on ^{235}U enrichment and other factors, such as the amount of moderator. For example, the minimum critical mass of an aqueous mixture of ^{235}U with full water reflection is approximately 800 grams. The single parameter subcritical limit for this condition is 700 grams of ^{235}U (ANSI/ANS 1983b).

7.2.1.3 Density or Concentration

Density or concentration is defined as mass per unit volume (grams/liter, etc.). A uniform solution or slurry less than 10.8 gm ^{235}U /l will be subcritical at any volume, while a concentration four or five times greater could result in the minimum critical mass (ANSI/ANS 1983b).

7.2.1.4 Moderation and Reflection

A moderator is a material that slows down fast neutrons. The most effective moderators are those materials having a low atomic weight, such as hydrogen, deuterium, beryllium and carbon. The moderator concentration is usually expressed as the ratio of the number of hydrogen atoms to the number of fissionable atoms of the isotope; thus, the extent of moderation in an aqueous solution of ^{235}U may be expressed as the $H/^{235}U$ ratio. The ratio $H/^{235}U$ may range from zero for metal, or a dry unhydrated salt, to several thousands for a dilute aqueous solution. Over this concentration range and with the assumed spherical geometry, the critical mass may vary from a few tens of kilograms (with little hydrogen) through a minimum of a few hundred grams (at optimum moderation) to infinity in a very dilute solution where the neutron absorption by hydrogen makes a chain reaction impossible. A moderated and/or reflected system allows a smaller mass of ^{235}U to become critical.

A reflected system is an assembly where the fissionable material is partly or wholly surrounded by another material having a greater neutron scattering cross-section than air. (Technically, air is a reflector, but its effect is usually negligible). In a reflected system, a fraction of the neutrons leaving the fissionable material (core) is reflected back into the fissionable material where they may induce additional fissions. The effect of a reflection is to reduce the minimum critical mass. A good reflector is a material that has a low neutron absorption cross-section. Water, concrete, graphite, and stainless steel are typically "good" reflectors, although any material will serve as a reflector. A "fully reflected" system is one where the fissionable material is totally surrounded by a reflector such that increasing the reflector thickness results in little or no decrease in the critical mass. For example, experiments at various laboratories have shown that increasing the thickness of water surrounding the fissionable material beyond 8 inches does not significantly decrease the critical mass (Paxton et. al.1986).

7.2.1.5 Geometry or Shape

Leakage of neutrons from a system depends on the shape of the system and on the neutron-reflecting properties of surrounding materials. The shape and size of containers are determined by considering the ratio of surface area (S) to volume (V). The ratio S/V is maintained at a value that prevents a chain reaction regardless of the quantity of material contained.

7.2.1.6 Interaction or Arrays

Interaction is the exchange of neutrons between separate containers containing uranium material. An increase in the exchanged neutrons increases the fission reaction rate. Units that are subcritical individually can be made into a critical array if brought near each other.

7.2.1.7 Neutron Poisons (Absorbers)

Neutron absorbers (poisons) are nonfissionable materials that capture neutrons, thus reducing the number of neutrons available for a fission reaction. Cadmium, boron, and chlorine are examples of neutron absorbers. Boron in borosilicate glass Raschig rings and chlorine in polyvinyl chloride (CPVC) rings are poisons used in some applications.

7.2.1.8 Monitoring for Deposits for Nuclear Safety Control

One concern in many older facilities is the potential for accumulation of uranium compounds in ventilation ductwork and process piping. A program must be in effect to routinely monitor such equipment to identify uranium compound deposits in quantities that may present nuclear criticality safety concerns. The need for such a program should be determined by nuclear criticality safety specialists, based on the enrichment of material processed (both past and present) and the geometry of the ductwork or piping. Such a review and survey should also be conducted prior to shutdown and decommissioning of uranium facilities. In general, the use of NaI detectors, in conjunction with single or multichannel analyzers, can often provide adequate sensitivity to determine holdup deposits. If intervening shielding reduces sensitivity and/or background gamma radiation levels are too great, neutron detectors may be effective in identifying uranium deposits, particularly for highly enriched uranium. Since the hold-up measurements are generally taken in "cpm" for maximum sensitivity, it is useful to have a correlation from "cpm" to exposure or dose units to facilitate an understanding of the relative radiological hazard.

7.2.2 Double Contingency Principle

DOE O 420.1A mandates the application of the double contingency principle in nuclear criticality safety.

The double contingency principle, as defined in DOE O 420.1A, requires that process designs incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before an inadvertent, unplanned criticality could occur. Protection, or defense in depth, shall be provided by either a) the control of two independent process parameters (which is the preferred approach, if practical) or b) a system of multiple controls on a single parameter. In all cases, no single failure shall result in the potential for a criticality accident. The basis for selecting either approach shall be fully documented.

The two parameters that are controlled in the double contingency analysis process shall not be subject to common mode failures. Judgment is required in determining whether the two events are related and, consequently, whether they represent two contingencies or a single contingency. For example, exceeding a storage limit and then flooding an area with water would constitute two independent events. However, a fire followed by the flooding of a storage area with fire suppression water would constitute a single event.

The double contingency principle is to be applied to all nuclear criticality safety analyses for processes, systems and equipment, storage, and transportation of fissionable materials. Should contingencies be determined to be related, efforts shall be made to separate the contingencies.

7.2.3 Administrative Practices

Administrative practices consist of personnel, programs, plans, procedures, training, audits and reviews, and quality assurance practices used to administer a nuclear criticality safety program. Administrative controls are used in addition to physical design features, including engineered controls, to ensure nuclear criticality safety. ANSI/ANS-8.19 outlines administrative practices. An effective nuclear criticality safety program requires a joint effort by managers, supervisors, workers, and nuclear criticality safety staff and relies on conformance with operating procedures by all involved personnel. The following sections describe the key elements of a nuclear criticality safety program.

7.2.3.1 Nuclear Criticality Safety Program

Management should develop a nuclear criticality safety policy and ensure it is distributed to fissionable material workers. They should also delegate authority to implement the policy, monitor the nuclear criticality safety program, and periodically participate in audits of the program. Supervisory staff should ensure that nuclear criticality safety procedures are written and staff are trained in those procedures. The nuclear criticality safety staff should provide technical guidance for equipment and process design and for operating procedure development. The nuclear criticality safety staff should perform a nuclear criticality safety evaluation before starting a new operation with fissionable materials or before changing an existing operation. An independent expert should evaluate the technical adequacy of the nuclear criticality safety program periodically.

7.2.3.2 Nuclear Criticality Safety Organization

Like the radiation protection program, the nuclear criticality safety organization should report to the highest level of facility management independent of operations. Management should clearly communicate nuclear criticality safety organization responsibilities and authorities to other facility personnel. Organizational and procedural documents should clearly define lines of interaction and interfaces with other facility organizational components. Management should assign the responsibility for nuclear criticality safety in a manner that is consistent with other safety disciplines. The organization should also have an independent nuclear criticality safety review committee and have access to consultants to assist in the conduct of the criticality safety program.

7.2.3.3 Plans and Procedures

Facility nuclear criticality safety plans and procedures are critical components of the overall facility operation. The purpose of procedures is to facilitate the safe and efficient conduct of operations. These documents provide the means by which the program is conducted and prescribe how nuclear criticality safety is to be achieved. The plans and procedures describe administrative activities and the technical aspects of nuclear criticality safety analysis. The processes of procedure development, review, training, and approval should have sufficient controls to ensure that nuclear criticality concerns are properly addressed. These controls include periodically reviewing and reaffirming procedures, and properly investigating procedure deviations and reporting them to facility management and, if appropriate, to DOE. The controls should also ensure such deviations do not recur.

Procedures should exist that address the determination and posting of nuclear criticality safety parameters. These procedures should include a description of how the limits are to be determined and how workstations are to be posted as to form, geometry controls, mass limits, moderator limits, etc.

Management should provide fire-fighting guidelines to ensure fire-fighting techniques do not violate a criticality control limit that might lead to an inadvertent nuclear criticality event. These guidelines should include the posting of specific rooms with acceptable fire-suppression techniques that can be used for a specific location or the use of notations on facility fire pre-plans (operating procedures) located at fire stations.

Recovery procedures should be in place to provide for the recovery from a nuclear criticality control limit violation. A limit violation involves exceeding the fissionable material mass limit or the moderator liquid limit, or violating any other criticality control in an operations procedure. This process should separately address both static and dynamic cases, as
responses to these violations may be quite different.

Management should develop and implement nuclear criticality safety training plans and procedures for all personnel working with or near fissionable materials, as required by ANSI/ANS-8.20 and DOE-STD-1136-99 Guidance for Nuclear Criticality Safety Engineer Training and Qualification (DOE 1999n). This program and its associated procedures describe the program, training requirements, recordkeeping, content, responsibilities, and objectives of a facility nuclear criticality safety program.

Inspections and audits are performed to assess the success of the nuclear criticality safety program. Qualified individuals who are independent of the operation should perform the inspections and audits. The audits and inspections should verify that operating procedures and other safety standards are being followed and identify any weaknesses in the nuclear safety program. Deficiencies should be formally addressed, tracked, reported, and resolved.

7.2.3.4 Nuclear Facility Safety Analysis

Documented Safety Analyses (DSA) document the analysis and potential consequences of accidents and abnormal occurrences at nuclear facilities. Per 10 CFR 830.204 (DOE 2001d), with respect to a nonreactor nuclear facility with fissionable material in a form and amount to pose a potential for criticality, the DSA defines a criticality safety program that: ensures that operations with fissionable material remain subcritical under all normal and credible abnormal conditions, identifies applicable nuclear criticality safety standards, and describes how the program meets applicable nuclear criticality safety standards.

7.3 CRITICALITY ACCIDENT EXPERIENCE

Criticality accidents, sometimes called criticality excursions, can either be single pulse, multiple pulse, or "steady state" (continuous) excursions.

7.3.1 Types of Criticality Accidents

In a pulse-type criticality accident, there is an initial pulse of 10^{16}-10^{18} fissions over a short time-period (less than 1 second), sometimes followed by additional lower-intensity pulses. In a fissionable

material solution, the pulse or spike is terminated by the heating and consequent thermal expansion of the solution and by bubble formation that serves to reconfigure the fissionable mass into a noncritical configuration (Paxton 1966). If the initial pulse results in a loss of solution from the container (e.g., by splashing) or redistribution of material, the criticality event may conclude without further pulses. However, if there is no loss of material as the solution cools, it may form a critical mass once again and pulse with slightly lower fission yield.

Criticality accidents can result in lethal doses of neutron and gamma radiation at considerable distances from the accident site (on the order of tens of meters). There can also be high level beta-gamma residual radiation levels from fission products after the excursion is concluded. The heat generated during the excursion can melt parts of the system that contain the fissionable material (Moe 1988).

Moe reviewed estimated prompt radiation doses from excursions in a moderated system and a metallic system, as well as dose rates from residual contamination left by a criticality excursion. Assuming a burst of 10^{18} fissions in an unshielded, water-moderated system, the total absorbed dose is estimated to be >600 rad up to 6 m and >100 rad up to about 15 m. The gamma/neutron ratio of the total absorbed dose was 2.8. An excursion of 3×10^{15} fissions in a metallic, partially reflected ^{239}Pu assembly, assuming no shielding, yielded total absorbed doses of >600 rad up to approximately 10 m and >100 rad up to approximately 25 m. The gamma/neutron absorbed dose ratio was 0.1. In general, for a moderated system, the gamma dose would be expected to be higher than the neutron dose and, for a metal system, the neutron dose would be expected to be higher than the gamma dose.

Moe (Moe 1988) noted that for an excursion of $>10^{18}$ fissions, dispersion of the fissionable material and fission products would occur, resulting in heavy local contamination and subsequent high residual dose rates. This dose rate was estimated at >1000 rad/h at 100 ft shortly after the burst and >10 rad/h at 30 ft an hour after the burst. This is the basis for instructing workers to immediately run from the work area when the criticality alarm is sounded. Seconds can save significant dose, if not from the excursion itself, then from any residual radiation that is in the area.

Additional guidance for estimating dose following a criticality accident may be found in NUREG/CR-5504 An Updated Nuclear Criticality Slide Rule (NRC 1994).

7.3.2 Summary of Past Criticality Accidents

Current criticality safety practice has been influenced both by the overall experience of the nuclear industry and by the analysis of the accidental criticality excursions that have occurred. Los Alamos National Laboratory has published LA-13638 Review of Criticality Accidents (McLaughlin et al. 2000) which provides a description of 60 criticality accidents. According to LA-13638, there have been 22 criticality accidents in chemical process facilities. Twenty-one of the 22 occurred with fissile material in solutions or slurries, one occurred with metal ingots. No accidents occurred with powders.

Overall, the consequences from the 22 accidents have been 9 deaths, 3 survivors with limbs amputated, minimal equipment damage, and negligible loss of fissionable material. One of these incidents resulted in measurable exposure to the general public (well below allowable worker annual exposures). All accidents have been dominated by design, managerial, and operational failures. The focus for accident prevention should be on these issues.

7.4 CRITICALITY ALARMS AND NUCLEAR ACCIDENT DOSIMETRY

Guidelines for criticality alarm systems and nuclear accident dosimetry are presented in this section. Criticality alarm systems provide rapid warning to individuals in the immediate accident location and nearby locations to evacuate to a predesignated assembly location. Specific requirements for the criticality alarm system are found in ANSI/ANS-8.3. Key requirements that may be of interest for the radiological control staff are summarized in Section 7.4.1. Paxton noted lives have been saved in past criticality accidents by radiation alarms coupled with effective evacuation procedures. Nuclear accident dosimetry, discussed in Section 7.4.2, provides the means for determining the dose to workers in the vicinity of the excursion.

7.4.1 Criticality Accident Alarm System (CAAS)

As specified in ANSI/ANS-8.3, the need for a CAAS shall be evaluated for all activities in which the inventory of fissionable material in individual unrelated work areas exceeds 700 g of ^{235}U, 520 g of ^{233}U, 450 g of ^{239}Pu or 450 g of any combination of these three isotopes.

a. If the fissionable material mass exceeds the ANSI/ANS-8.3 limits and the probability of criticality is greater than 10^{-6} per year, a CAAS shall be provided to cover occupied areas in which the expected dose exceeds 12 rad in free air. Nuclear accident dosimetry shall also be provided, as otherwise required. The CAAS should include a criticality detection device and a personnel evacuation alarm.

b. If the fissionable material mass exceeds the ANSI/ANS-8.3 limits and the probability of criticality is greater than 10^{-6} per year, but there are no occupied areas in which the expected dose exceeds 12 rad in free air, then only a criticality detector system (i.e., nuclear accident dosimetry) is needed.

c. If the fissionable material mass exceeds the ANSI/ANS-8.3 limits, but a criticality accident is determined to be impossible or less than 10^{-6} per year (per a Documented Safety Analysis), then neither a criticality alarm nor nuclear accident dosimetry is needed.

The alarm signal shall be for immediate evacuation purposes only and of sufficient volume and coverage to be heard in all areas that are to be evacuated. Information on sound levels of the alarm can be found in ANSI/ANS-8.3. The alarm trip point shall be set low enough to detect the minimum accident of concern. The minimum accident of concern may be assumed to deliver the equivalent of an absorbed dose in free air of 20 rad at a distance of 2 meters from the reacting material within 60 seconds. The alarm signal shall activate promptly (i.e., within 0.5 second) when the dose rate at the detectors equals or exceeds a value equivalent to 20 rad/min at 2 meters from the reacting material. A visible or audible warning signal shall be provided at a normally occupied location to indicate system malfunction or loss of primary power. Each alarm system should be tested at least once every three months. An evacuation drill shall be conducted at least annually.

Criticality accident alarm systems may consist of one to several detectors per unit. In multi-detector units (e.g., three detectors), at least two detectors shall be at the alarm level before initiating the alarm; in redundant systems, failure of any single channel shall not prevent the CAAS from functioning.

7.4.2 Nuclear Accident Dosimetry

Nuclear accident dosimetry shall be provided for installations that have sufficient quantity of fissionable material such that the excessive exposure of individuals to radiation from a nuclear criticality accident is possible (10 CFR 835.1304(a)).

Requirements for nuclear accident dosimetry programs at DOE facilities are found in 10 CFR 835. A nuclear accident dosimetry program shall include the following:

a. a method to conduct initial screening of individuals involved in a nuclear accident to determine whether significant exposures have occurred

b. a system of fixed nuclear accident dosimeter units. Sometimes referred to as area dosimeters, the dosimeters should be capable of yielding estimated radiation dose and the approximate neutron spectrum at their locations

c. personal nuclear accident dosimeters (PNADs)

d. methods and equipment for analysis of biological materials (such as ^{24}Na activity in blood and ^{32}P activity in hair)

7.4.2.1 Initial Screening Evaluation

A nuclear accident dosimetry program should provide absorbed dose information within 24 hours after the incident. A method should be established for immediately obtaining preliminary dose estimates to distinguish exposed persons from the unexposed and should permit the detection of doses in excess of approximately 10 rad (see ANSI N13.3 (ANSI 1969)). Discussions on initial screening evaluations to segregate exposed from unexposed individuals (sometimes referred to as "quick sort techniques") are found in several references (Moe 1988; Delafield 1988; Petersen and Langham 1966; Hankins 1979; Swaja and Oyan 1987).

A common initial screening method is to provide all workers in areas requiring nuclear accident dosimetry with an indium foil in their personnel dosimeter or security badge. During a criticality excursion, the foil will become activated by neutrons per the ^{115}In(n,gamma) ^{116m}In reaction and can be measured with a portable beta-gamma survey instrument or ion chamber. The ^{116m}In has a 54-minute half-life and releases a 1-MeV beta (maximum energy) and a 1.3-MeV gamma (80% of the time).

An alternate screening is to measure body activity due to neutron activation of the sodium in the blood via the ^{23}Na(n, gamma)^{24}Na reaction. Sodium-24 has a 15-hour half-life and releases a 1.4-MeV beta (maximum energy) and two gammas (1.37 MeV and 2.75 MeV). A beta-gamma survey meter is used to measure the ^{24}Na activity in the blood by placing the detector probe against the individual's abdomen and having the individual bend forward to enclose the detector (Moe 1988). Alternatively, the probe can be positioned under the armpit with the open window facing the chest area. Moe noted this

method is less sensitive than the use of indium foils and even a small reading can indicate a significant exposure. An approximate equation to calculate worker dose (D) based on body weight and instrument reading is shown in Equation 7.1:

$$D(Gy) = \frac{80 \text{ (instrument reading in mR/h)}}{\text{Body weight (lb)}} \qquad (7.1)$$

Differences in incident neutron energy spectrum, orientation, and measurement techniques relative to conditions used to develop activity-dose correlations can cause significant errors in estimated radiation dose based on quick-sort surveys. Swaja and Oyan showed radiation doses estimated from induced body activity can vary by a factor of approximately 2 because of neutron energy spectrum or orientation effects and by as much as 30% due to probe position. Doses based on indium foil activity can vary by a factor of approximately 9 due to neutron energy spectrum effects, a factor of 3 depending on foil orientation relative to the incident field, and a factor of approximately 2 due to probe window setting. Swaja and Oyan recommended those count rates above background during quick-sort techniques should be initially interpreted only as an indication that the person has been exposed.

7.4.2.2 Fixed and Personnel Nuclear Accident Dosimeters

A comprehensive nuclear criticality dosimetry system should consist of stationary (fixed-location, area) dosimeters, neutron and gamma dosimeters worn by personnel (i.e., PNADs), and specialized laboratory equipment to evaluate the dosimeters.

Fixed nuclear accident dosimeter units should be capable of determining neutron doses in the range of 10 rad to 10,000 rad with an accuracy of ±25%. They should also be capable of providing the approximate neutron spectrum to permit the conversion of rad to rem. The gamma-measuring component of the dosimeter should be capable of measuring doses in the range of 10 rem to 10,000 rem in the presence of neutrons with an accuracy of about ±20%. The number of fixed dosimeter units needed and their placement will depend on the nature of the operation, structural design of the facility, and accessibility of areas to personnel. Generally, dosimeters should be placed so there is as little intervening shielding and as few obstructions as possible. The number and placement of dosimeters should be periodically reverified to reflect changes in building design and operations. Ease of dosimeter recovery after a criticality event should be considered in their placement, including the possible need for remote retrieval.

PNADs should be worn by all individuals who enter a controlled area, with locations requiring an installed criticality alarm system. The PNADs should be capable of determining gamma dose from 10 rad to 1000 rad with an accuracy of ±20% and neutron dose from 1 rad to 1000 rad with an accuracy of ±30% without dependence upon fixed-unit data.

The general criteria of ANSI N13.3 for nuclear accident dosimeters are reviewed below. Dosimeters, both fixed and personnel, should be protected against radioactive contamination to avoid false measurements. Periodic inventory methods should be established and audits made to ensure the dosimeters are not removed or relocated without appropriate approvals. Techniques for estimating the effect of body orientation at the time of the exposure should also be developed.

- *Neutron-Measuring Component of Dosimeter.* Criticality accidents create a wide range of neutron energies. Since the neutron dose per unit fluence is strongly dependent on neutron

energy, knowledge of the neutron energy spectrum is important in accident dosimetry. In criticality accidents, neutrons with energies greater than 100 keV contribute most of the dose; therefore, measurement of the fast neutron dose is of the most importance. See Delafield (Delafield 1988) for a review of the different types of neutron dosimeters available for accidents.

- *Gamma-Measuring Component of Dosimeter.* Delafield noted the ratio of the gamma rays to neutron dose will vary according to the type of critical assembly and whether or not additional shielding is present. For unshielded assemblies, the gamma-to-neutron ratio can range from 0.1 for a small heavy-metal system up to approximately 3 for a small hydrogen-moderated solution system. A concrete or hydrogenous shielding material will increase the gamma-to-neutron ratio. Gamma dose can be determined by TLD, film, or radiophotoluminescent glass.
- *Dosimeter Comparison Studies.* Sims and Dickson (Sims and Dickson 1979; Sims 1989) present a summary of nuclear accident dosimetry intercomparison studies performed at the Oak Ridge National Laboratory Health Physics Research Reactor. The more recent summary showed that of the 22 studies conducted over 21 years, 68% of the neutron dosimeter results were within the ±25% accuracy standard and 52% of the gamma dosimeter results were within the ±20% accuracy standard. Most measurements that failed to meet the accuracy standards overestimated the actual dose. Some of their other findings include the following:

  a. Doses from hard neutron energy spectra are more accurately measured than those from soft energy spectra.

  b. The threshold detector unit (TDU) is the most accurate type of nuclear accident neutron dosimeter; however, its use is declining due to increasingly strict control of small quantities of fissionable materials.

  c. Activation foils (ACT) are the most popular nuclear accident neutron dosimeter.

  d. For gamma dosimeters, TLDs are the most popular and the least accurate, and film is the least popular and the most accurate.

7.4.2.3 Biological Indicators

Earlier in this section, a quick-sort method was described that uses neutron activation of sodium in the blood as an indicator of worker exposure. More sophisticated laboratory analysis of blood samples can be performed to obtain a more accurate estimate of worker dose (Delafield 1988; Hankins 1979). The use of neutron activation of sulfur in hair ($^{32}S(n,p)^{32}P$) is another method to estimate absorbed dose for workers involved in a criticality accident. The orientation of the subject can also be determined by taking samples of hair from the front and back of the person. Hankins described a technique for determining neutron dose to within ±20-30% using a combination of blood and hair activations. The evaluation was independent of the worker's orientation, of shielding provided by wall and equipment, and of neutron leakage spectra.

7.5 RESPONSIBILITIES OF RADIOLOGICAL CONTROL STAFF

The radiological control staff should have a basic understanding of program structure, engineering criteria, and administrative controls as related to nuclear criticality safety and reviewed in earlier sections of this chapter. However, the health physicist's primary responsibilities with regard to nuclear criticality safety include emergency instrumentation and emergency response actions.

7.5.1 Routine Operations

During routine operations, the radiological control staff's primary responsibility related to nuclear criticality safety will include calibrating, repairing, and maintaining the neutron criticality alarm detectors and nuclear accident dosimeters, and maintaining appropriate records. The radiological control staff should be knowledgeable of criticality alarm systems, including alarm design parameters, types of detectors, detector area coverage, alarm set-points, and basic control design. The staff should also be familiar with locations and scenarios for designing the fixed nuclear accident dosimetry program and formulating plans for emergency response.

The radiological control staff should maintain an adequate monitoring capability for a nuclear criticality accident. In addition to the criticality alarm systems and the fixed nuclear accident dosimeters discussed above, remotely operated high-range gamma instruments, personal alarming dosimeters for engineering response/rescue teams, neutron-monitoring instrumentation (in case of a sustained low-power critical reaction), and an air-sampling capability for fission gases shall be maintained.

Other support activities may include assisting the nuclear criticality safety engineer or operations staff in performing radiation surveys to identify residual fissionable materials remaining in process system or ventilation ducts.

7.5.2 Emergency Response Actions

The priorities of the radiological control staff during a criticality event should be to rescue personnel, prevent further incidents or exposures, and quickly identify those who have been seriously exposed. To support these emergency response actions, the radiological control staff should be trained in facility emergency procedures. These emergency procedures include evacuation routes, personnel assembly areas, personnel accountability, care and treatment of injured and exposed persons, means for immediate identification of exposed individuals, instrumentation for monitoring the assembly area, and re-entry and formation of response teams.

Emergency response procedures for conducting the initial quick-sort of workers should specify measurement techniques and require that surveyors record methods and instrument settings used for quick-sort operations to ensure proper interpretation of the results. Surveyors/analysts should compare field results to pre-established activity-dose relationships developed as part of emergency response procedures to determine if a worker was exposed. Other indicators, such as a discharged self-reading dosimeter, could also indicate a possible exposure.

As an immediate follow-up action on workers identified as being exposed during a quick-sort procedure, a more accurate dose estimate should be made using PNADs, fixed-location accident dosimeters, or biological activity analyses (^{24}Na in the blood or ^{32}P in the hair). The more accurate analyses should include: 1) better definition of source characteristics, 2) location of moderating materials, and 3) location and orientation of the person(s) at the time of exposure and action of the person following the irradiation. If the radiological control staff are involved in the rescue and initial monitoring procedures, they can provide valuable information to support this analysis, particularly regarding the location and orientation of workers to the excursion.

Radiological control staff should be responsible for retrieving fixed nuclear accident dosimeters and ensuring that PNADs from any exposed workers are submitted for analysis.

7.5.3 Special Considerations During Decommissioning Activities

Before decommissioning or disposal of any facilities or equipment, an evaluation should be performed to assess the potential holdup of fissionable material in any equipment. These types of measurements may require the assistance of radiological control staff.

Some strippable coatings and surface-fixing films are effective neutron moderators. Nuclear criticality safety specialists should be consulted when using these coatings to decontaminate surfaces because criticality could be a concern, depending on the geometry of the removed coating when in the disposal unit.

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8.0 WASTE MANAGEMENT

A material is a waste once there is no identified use or recycle value for it. Normally, wastes are considered by their physical form as either solids, liquids, or gasses, except that containerized liquids are considered solid waste under some of the current regulations. Although these forms are each processed differently, there are interrelationships. For example, it may be possible to reduce solid waste by replacing disposable protective clothing with reusable clothing that must be laundered. The laundry will produce liquid waste. In treating liquid waste, solids may be generated, e.g., filters or ion exchange resins. By careful engineering, waste generation, and treatment alternatives, a site can minimize the total waste volume and elect to generate types of waste that can be disposed of. The following sections address potentially contaminated waste and waste terminology and handling of airborne waste, solid waste, and liquid waste. The treatment of excess materials to reclaim uranium is not a waste treatment process and is not discussed here.

8.1 POTENTIALLY CONTAMINATED WASTES

Wastes are generated within a plant or facility as a consequence of creating the uranium product(s) for which the plant was designed. Uranium may be entrained in the air, may contaminate equipment, materials, or other scrap, or may be contained in low concentrations in liquid wastes and effluents. Wastes resulting from operation of a uranium facility may include radioactive, nonradioactive, and mixed materials in the form of liquids and gaseous effluent or solids requiring disposal.

Uranium recovery operations and processes are an operational feature of most major facilities handling large quantities of material for at least two major purposes, i.e., to salvage valuable material and to reduce effluent concentrations and volumes to acceptable levels.

The facility and all waste systems must be designed to minimize wastes that result in the release of radioactive materials, during normal plant operation, the occurrence of a Design Basic Accident (DBA) meeting the regulatory limits, and conditions in which dose is kept as low as reasonably achievable. Waste systems include retention containers, cleanup systems for liquids and solids, and analytical equipment.

Accounting for waste management for solid and liquid wastes is discussed below.

8.1.1 Solid Waste

Facilities should provide for the safe collection, packaging, inventory, storage, and transportation of solid waste that is potentially contaminated with radioactive materials. Such provisions include adequate space for sorting and temporary storage of solid waste, equipment for assay of the waste, and facilities for volume reduction appropriate to the types and quantities of solid waste expected. All packages containing potentially contaminated solid waste should be appropriately monitored, both before being moved to temporary storage locations and before being loaded for transport to a disposal site.

8.1.2 Liquid Waste

Industrial wastes such as discharge from mop sinks, overflow from positive pressure circulating waste systems, and process steam condensate (if existing) should be analyzed, collected and transferred to a liquid waste treatment plant or similar treatment area if mandated by the chemical analysis. Provisions should be made for continuous monitoring and recording of radioactivity, flow volume, and

pH. The radioactivity monitor should have an alarm located in the liquid waste treatment plant or area. Consideration should be given to retention systems.

Liquid process wastes should be collected and monitored near the source of generation before batch transfer through appropriate pipelines or tank transfer to a liquid waste treatment plant or area. These wastes should be individually collected at the facility in storage tanks that are equipped with stirrers, sampling and volume-measuring devices, and transfer systems. Waste storage tanks and transfer lines should be designed and constructed so that they are fully inspectable and that any leakage can be detected and contained before it reaches the environment.

Sanitary wastes include the nonradioactive wastes usually found at a facility, e.g., discharges from noncontaminated chemical laboratories, showers, and lavatories. The sanitary waste system and the uranium-handling area should not be connected. Sanitary sewers should discharge into an onsite, approved sanitary-sewage treatment system. Current Federal, state, and local codes regarding the discharge of sanitary wastes must be met.

8.2 DESIGN OF WASTE PROCESSING SYSTEMS

Process system designs may be characterized by their design objectives and the effluents of concern.

8.2.1 Objectives

A principle design objective for process systems is to minimize production of wastes at the source. One of the primary design objectives of any Waste Management Program is to provide facilities and equipment to handle the wastes generated and further reduce the amounts and volume of the waste. Volume-reduction facilities and equipment for liquid and solid wastes are required, as is air filtration to reduce the concentration of contaminants in the air effluent.

8.2.2 Effluents

Airborne and liquid effluents released uncontrolled to the environment are of particular concern when societal emphasis on environmental pollution control is high. Process and monitoring equipment are critical to maintaining acceptable operations.

Effluents (both radioactive and nonradioactive) from the uranium-handling facility include air and other gaseous exhausts and liquid wastes. The contamination in the effluents should be kept ALARA, commensurate with best available technology at the time of design. Emphasis should be placed on reducing total quantities of effluents (both radioactive and nonradioactive) released to the environment. Filter systems should be designed so that the effluent concentrations of uranium should not exceed the inhaled air Derived Air Concentration Guide (DCG) for releases, as described in DOE 5400.5, Ch.2, Radiation Protection of the Public and Environment (DOE 1993a) for uncontrolled areas measured at the point of discharge (e.g., exhaust ducts and stacks) during normal operations. Consideration should be given to recirculation systems for process ventilation where feasible. Provisions should be made for retention systems for liquid effluents. All effluent streams should be sampled or monitored as appropriate to ensure accurate measurements of all releases under normal and DBA conditions.

8.3 TREATMENT

The following sections provide information about treating airborne, liquid, and solid wastes.

8.3.1 Airborne Wastes

Ventilation control systems within a plant are designed to move air from outside "clean" areas to process areas and then to air-cleanup systems. Occupied area off-gas systems are also vented to the atmosphere and may have cleanup systems of their own. Process off-gas treatment systems consist of any or all of the following:

Wet scrubbers are generally used in dusty process off-gas situations, in which large amounts of uranium are present. The scrubbers are capable of removing and processing large quantities and serve as a prefilter to the remaining cleanup units.

Prefilter systems other than the wet scrubber are bag filters or other rough/coarse filters. The prefilters are used to remove significant quantities of particulate material from the air off-gas and are generally placed before high-efficiency particulate air (HEPA) filters in order to extend the life of the more expensive filters.

HEPA filters generally are the final filter in the process off-gas and serve to reduce the particulate effluent to insignificant or permissible levels. They may be placed in series to provide the required filtering efficiency. See section 8.3.3 for disposition of HEPA filters.

8.3.2 Liquid Waste

Because liquid effluents are generally released to the environment, liquid wastes are of equal concern with airborne wastes. Liquid effluents become available for dispersion and reconcentration in food chains, and may otherwise result in population exposure potential. In the case of liquid wastes, the concern for chemical pollutants is generally of equal concern to that of radiological contaminants. Liquid process wastes are generally collected in hold tanks, monitored, processed or treated, and released.

Hold tanks are used to collect liquid effluent prior to release in order that analyses can be performed to establish that the concentrations or total quantities are below permissible levels prior to release. The liquid can be processed or treated to remove radioactive material or neutralize chemicals.

Settling basins are frequently used to provide a means of reducing effluents further before releasing them to offsite areas.

Filtration is a simple method of removing insoluble particulate materials entrained in the liquid streams. For some processes, it is an effective and inexpensive method. The particulate material collected and filter must be periodically removed and treated as solid waste.

Ion exchange is a cleanup system for removing soluble ions from the liquid streams by collecting the material on resin columns. The contaminants must be periodically removed by a regeneration process and the materials processed, concentrated, etc., or by replacing the resin completely and treating it as solid waste.

Conversion to solid forms is a function of nearly all the processes mentioned which converts the materials removed from the liquid and airborne waste streams to more manageable forms for handling and permanent disposal.

8.3.3 Solid Waste

Solid waste come from a variety of sources in the plant from machining chips to contaminated clothing. The solid wastes should be concentrated (if possible and/or practical), packaged, and stored on the plant site for an interim time period prior to permanent disposal. Careful documentation is necessary to establish: a) quantities and nature of the waste being disposed, and b) compliance with the Resource Conservation and Recovery Act (RCRA) and other disposal and shipping/handling requirements.

Onsite volume-reduction facilities, such as incinerators, compactors, or chemical leach from metallic waste sources, can result in volume reduction in the range of 1 to 400 or more.

8.4 MONITORING

Monitoring the airborne effluents is an important aspect of control and documentation. Monitoring should be done in the stack at the discharge point and at the boundary of the uncontrolled area. In addition, total activity discharged and total mass of uranium discharged should be determined and documented to ensure that concentration requirements are not exceeded. If stack monitoring cannot be performed (e.g., in instances where the facility design is such that there are no stacks), then the reason for the monitoring method selected should also be documented.

Monitors are of two general types: continuous and passive. Continuous monitors are constructed with a radiation detector which is placed in a shielded container such that it "views" the activity as it is being collected on a filter from a sample of the stack effluent. The continuous level of radioactivity on the filter is recorded and set up in such a way that preset levels trigger an alarm. This type of monitor is less sensitive but provides an alarm in the event of mishap or equipment failure in time to take effective mitigating action.

Passive monitors consist of a continuous (isokinetic, if practical) sample collected of the effluent in the stack. The filter is periodically removed and submitted to radiological and/or chemical analyses. The sensitivity or level of detection is lower for passive sampling systems than for continuous stack samplers, and provide after-the-fact information only.

8.4.1 Air and Gaseous Effluents

All air and other gaseous effluents from confinement areas should be exhausted through a ventilation system designed to remove particulates. All exhaust ducts (or stacks) that may contain fissile contaminants should be provided with two monitoring systems. One should be of the continuous type (CAM) and the other a passive sampler. These systems may be a combination unit. The probes for sampling purposes should be designed for isokinetic sampling and located according to good industrial hygiene practices. The design of effluent monitoring systems should appropriately meet the requirements of ANSI N42.18, Specification and Performance of Onsite Instrumentation for Continuously Monitoring Radioactive Effluents (ANSI, 1974). Nuclear criticality safety should be considered in the design of equipment used to treat and clean up radioactive gaseous effluents.

8.4.2 Liquid Effluents

Emphasis should be placed on reducing total quantities of liquid effluents released to the environment. The contamination in the effluents should be ALARA, commensurate with the latest accepted technology at the time of design. All effluent streams should be sampled or monitored, as appropriate, to ensure accurate measurement of all releases under normal and DBA conditions. The

design of effluent monitoring systems should appropriately meet the requirements of ANSI N42.18 (ANSI, 1974).

8.4.3 Water Collection System

Collection systems should be considered and provided where practical for water runoff from nuclear facilities containing radioactive material, such as from firefighting activities. Nuclear criticality, confinement, sampling, volume determination, and retrievability of liquids and solids should be required in the design of collection systems. The size of the collection system for firefighting water should be based on the maximum amount of water which would be collected in fighting the Design Basis Fire (DBF). The configuration of the system components should be based on conservative assumptions as to the concentration of fissile material which might collect in the system. Recirculating systems should also be considered when there is no possibility of contamination.

For special facilities that process, handle, or store uranium, the water runoff collection system should be designed with the following nuclear criticality safety considerations: 1) the maximum uranium mass loading that could be in the runoff system; 2) the most disadvantageous uranium concentrations, particle size, and uranium dispersion in the water slurry; and 3) the change in concentration of uranium and geometric configuration of the slurry as the uranium settles out of the water.

8.5 WASTE MINIMIZATION

Uranium facilities should have a waste-minimization program. The objective of such a program is the cost-effective reduction in the generation and disposal of hazardous, radioactive, and mixed waste. The preferred method is to reduce the total volume and/or toxicity of hazardous waste generated at the source, which minimizes the volume and complexity for waste disposal.

The waste minimization program applies to all present and future activities of the facilities that generate hazardous, radioactive, and/or mixed wastes. Furthermore, waste minimization is to be considered for all future programs and projects in the design stages, and should be included in all maintenance and/or construction contracts.

All managers of facilities or activities that generate hazardous, radioactive, and mixed waste are responsible for:

- minimizing the volume and toxicity of all radioactive, hazardous, and radioactive mixed waste generated, to the extent economically practicable,
- preparing and updating waste minimization plans for their waste-generating facilities or activities (small waste generators in a larger facility may be grouped with others in a facility or activity plan),
- implementing the facility-specific or activity-specific waste minimization plan,
- providing input to the organization responsible for waste characterization and minimization, to support the waste minimization program,
- communicating waste minimization plans to their employees, and ensuring that employees receive appropriate training,

- ensuring that existing system/equipment replacement or modification is designed and installed to minimize generation of waste,

- developing new waste-minimization strategies and identifying cognizant staff for waste minimization communications between facility personnel, and

- identifying new waste-generating facilities or activities and significant process changes to existing facilities or activities to the waste characterization and waste-minimization organization.

Waste volume control, or waste minimization, involves limiting the amount of material that becomes contaminated, segregating clean and contaminated material, and prolonging the useful life of equipment and material to minimize replacement. Sometimes, materials can be completely cleaned so that disposal as sanitary waste (or refurbishment in clean areas) is an option.

Program design decisions can affect uranium waste-generation. For example, the quantity of protective clothing may be a significant factor. If an incinerator is available, combustible protective clothing may be selected to have a low ash content and generate a minimum of harmful effluents, such as oxides of nitrogen or halogenated compounds. In other facilities, water-washable, reusable protective clothing may minimize waste disposal.

In many nuclear facilities, contamination of packaging materials is a problem. For example, if a tool or material (e.g., a pump or some ion exchange resin) is to be used in a contaminated area, as much of the packaging material must be removed as possible before the material enters the radiological area.

Another opportunity for waste minimization occurs when materials are used as a contingency protection against contamination. For example, strippable coatings may be applied to an area that is not expected to become contaminated or may receive only minor contamination so that it can be easily cleaned. Another example involves disposable surgeons' gloves, which are routinely worn inside glove-box gloves. Unless there are serious contamination control problems in the facility, these can be surveyed and disposed of as sanitary waste rather than LLW or TRU waste.

If a piece of equipment is to have more than a single use in a contaminated environment, every possible measure should be taken to ensure its continued reliability rather than relying on frequent replacements. Tools should be of the highest quality and maximum flexibility consistent with the situation. For example, if a wrench is needed to maintain a piece of equipment in a glove-box, consideration should be given to future needs and storage provisions. A socket set with interchangeable sockets may ultimately create less waste than a box-end wrench of each size that is needed.

Likewise, all tools and equipment to be placed in a contaminated environment should be tested for reliability and preferably used on a clean mock-up to ensure their serviceability before they become contaminated in order to avoid unnecessary waste volume.

9.0 EMERGENCY MANAGEMENT

It is DOE policy that all DOE facilities and activities be prepared to deal with operational emergencies in a way that minimizes consequences to workers, the public, and the environment. Formal emergency management programs are the final element of DOE's defense-in-depth against adverse consequences resulting from its operations.

9.1 EMERGENCY MANAGEMENT IN DOE

DOE Order O 151.1B (DOE 2003c) requires all DOE elements and contractors to plan and prepare for the management of emergencies. DOE Order O 151.1 cancels the 5500-series, which previously formed the basis for DOE's Emergency Management System (EMS). However, where a contractual obligation to comply with the 5500-series Orders exists, they will remain in effect until the contract is modified to delete the references to the requirements in the canceled Orders. There is little difference between the facility emergency planning/preparedness requirements of the 5500-series Orders and DOE O 151.1B. Except as noted, the following discussion of emergency management principles, requirements, and guidance is applicable to all DOE facilities, regardless of which Orders pertain.

9.1.1 KEY EMERGENCY MANAGEMENT PRINCIPLES

DOE emergency management policy and direction is based on four key principles: planning and preparedness commensurate with hazards; integrated planning for health, safety, and environmental emergencies; classification of and graded response to emergencies; and multiple levels (tiers) of emergency management responsibility.

> **Note on Terminology:** Within the EMS, "planning" includes the development of emergency plans and procedures and the identification of personnel and resources necessary to provide an effective response. "Preparedness" is the procurement and maintenance of resources, training of personnel, and exercising of plans and procedures. "Response" is the implementation of the plans during an emergency to mitigate consequences and recover.

9.1.1.1 Planning and Preparedness Commensurate with Hazards

Because of the wide range of activities and operations under DOE's authority, standards and criteria suited to one type of facility or hazard may be inappropriate for another. To deal with this diversity in circumstances while ensuring an adequate overall state of preparedness, DOE Orders specify standards for the structure and features of emergency management plans and require the details of each feature be tailored to the hazards of the facility. This approach provides a more complete and quantitative understanding of the hazards while providing for focused and cost-effective emergency planning and preparedness.

9.1.1.2 Integrated Planning for Health, Safety, and Environmental Emergencies

A wide variety of operational emergencies can occur at DOE operations. Some may involve loss of control over radioactive or other hazardous materials unique to DOE operations, while others may involve security, the impact of natural phenomena, environmental damage, or worker safety and health. Planning, preparedness, and response requirements for some types of emergency conditions are specified by other agencies having authority over DOE facilities and activities. For example, Federal regulations on occupational safety, environmental protection, and hazardous waste operations all have certain

emergency planning requirements. Rather than meet these requirements piecemeal through separate programs, DOE has combined, under the EMS, all planning and preparedness activities for emergency events having health, safety, or environmental significance.

9.1.1.3 Classification of Emergencies and Graded Response

Operational emergencies involving hazardous materials are grouped into one of three classes according to magnitude or severity. Classification of events is intended to promote more timely and effective response by triggering planned response actions appropriate to all events of a given class. This principle, termed "graded response," is embodied in DOE Order requirements and is important to the management of response resources.

9.1.1.4 Tiers of Emergency Management Responsibility

Within the EMS, responsibility for emergency management extends from the individual facility level to the cognizant DOE field element and culminates at the cognizant Headquarters Program Office. The responsibilities vested at each level of the hierarchy are specified in the applicable Orders. The responsibility and authority for recognizing, classifying, and mitigating emergencies always rests with the facility staff. The head of the cognizant field element oversees the response of contractors and supports the response with communications, notifications, logistics, and coordination with other DOE elements. The DOE Headquarters (HQ) Emergency Operations Center (EOC) receives, coordinates, and disseminates emergency information to HQ elements, the cognizant Program Office, Congressional offices, the White House, and other Federal Agencies.

9.1.2 Requirements Pertaining to All DOE Operations

DOE Orders identify 13 standard program elements that compose each DOE facility emergency management program. The 13 elements form a standard framework, with the details of each program element varying according to the nature and magnitude of the facility hazards and other factors. The Orders require that a facility-specific hazards assessment be conducted and the results used as the technical basis for the program element content. Using the results of an objective, quantitative, and rigorous hazards assessment as its basis, each program is configured to the specific hazards and response needs of the facility.

Detailed guidance on the implementation of the Order requirements has been published by the DOE Office of Emergency Management. The Emergency Management Guides (EMGs) specify acceptable methods for meeting the Order requirements. Individual guides have been published for the hazards assessment process and for each of the 13 program elements.

9.2 SPECIFIC GUIDANCE ON EMERGENCY MANAGEMENT FOR URANIUM FACILITIES

This section provides technical guidance that is specifically applicable to the development and implementation of emergency management programs for uranium facilities. It is intended to supplement, not replace, the more general recommendations provided in the EMGs.

9.2.1 Hazards Assessment

The emergency management hazards assessment for a facility that stores or processes uranium or its compounds should follow the basic assessment process outlined in the Hazards Assessment EMG (DOE 1992d). Unique properties and characteristics of uranium and its compounds should be considered at certain steps in the hazards assessment process.

9.2.1.1 Description of Facility and Operations

The properties of the hazardous material do not significantly affect the manner in which this step of the hazards assessment is performed.

9.2.1.2 Identifying and Screening the Hazards

The objective of this step is to identify hazards that are significant enough to warrant consideration in a facility's emergency management program. It is recommended that screening thresholds (or quantities) be selected for each hazardous material. This screening threshold value is then compared to the inventories of the material at risk of being released from a single event. If a particular inventory of material is less than the screening threshold value, the consequences of its release are presumed to be minimal. The potential release of that inventory need not be analyzed further.

Several possible sources of screening threshold values are suggested by the EMG; the primary source is the Threshold Planning Quantities (TPQs) published in U.S. Environmental Protection Agency (EPA) regulations, 40 CFR 355, Appendix A. Neither uranium nor any of its compounds are listed in 40 CFR 355, Appendix A. A second recommended source of screening threshold values is the 40 CFR 302.4 List of Hazardous Substances and Reportable Quantities (RQs) (EPA 1995). Uranyl acetate and uranyl nitrate are listed there, both with RQs of 100 pounds. Appendix B to 40 CFR 302.4 gives RQs for radionuclides and specifies 0.1 Ci as the RQ for ^{234}U, ^{235}U and ^{238}U.

Lacking any widely accepted screening value (such as a TPQ or RQ), it is appropriate to establish facility and site-specific screening thresholds based on the properties of the material. A screening threshold can be determined by modeling a unit release of the material to the atmosphere at ground level and determining the consequences at some reference distance under conservative dispersion conditions. The screening threshold value is typically a quantity that, if released, would produce consequences on the order of one-tenth the threshold for protective action at the facility boundary.

Facility- and site-specific screening quantities for most materials are based on one hazardous property of the material that dominates all other considerations. However, for uranium and its compounds, three very different hazardous aspects need to be considered. When establishing a screening threshold, the chemical toxicity of the element or compound and its radiological toxicity need to be compared. The property for which the smallest release leads to an exposure or dose criterion being exceeded should serve as the basis for the screening value. In general, for natural or low-enriched uranium, chemical toxicity will be the dominant concern for soluble materials while radiological toxicity (radiation dose) will be limiting for insoluble materials. For enrichment above a few percent (as ^{235}U), the limiting concern may be either chemical or radiological, depending on the chemical form, solubility, and particle size of the material. For highly enriched material, radiological toxicity will nearly always be limiting.

If there is potential for accidental nuclear criticality, the consequences of that event should be analyzed as discussed later in this section. However, the quantity and enrichment of uranium needed to achieve criticality are sufficiently large that for practical purposes, radiological or chemical toxicity will always serve as the basis for determining whether a given inventory does or does not need to be analyzed.

9.2.1.3 Characterizing the Hazards

The objective of this step is to describe the hazardous materials in sufficient detail to allow accurate modeling of releases and calculation of consequences. The following properties of uranium and its compounds strongly influence the release potential and consequences.

- **Chemical form.** The chemical toxicity and reactive properties of any uranium compound must be weighed against the inherent toxicity of the compound or the uranium alone. For example, gaseous uranium hexafluoride (UF_6) reacts with atmospheric moisture and undergoes hydrolysis, producing uranyl fluoride (UO_2F_2) and hydrogen fluoride (HF), a highly corrosive and toxic gas. Depending on the temperature, humidity, and uranium enrichment, the HF may be a more serious health and safety concern than either the UF_6 or the contained uranium. Some uranium compounds ignite violently on contact with air, water, or hydrocarbons.

- **Physical form.** Physical form influences the release potential and toxicity of uranium and its compounds in numerous ways. Large, monolithic pieces of uranium metal may be relatively benign; however, they can develop a pyrophoric surface due to effects of air and moisture. Finely divided metallic uranium can react violently with numerous other materials or self-ignite in air, yielding respirable particles of uranium compounds. UF_6 is a solid at ambient temperature but goes directly to a gaseous state above ~270 °F at atmospheric pressure.

- **Solubility.** For air exposure, permissible exposure levels for soluble uranium compounds are based on the chemical toxicity (particularly to the kidney), while for insoluble compounds, radiotoxicity (radiation dose to the lung) is limiting.

- **Particle size.** Particle size and the range of sizes have a large effect on the radiotoxicity of inhaled materials. Larger particles will be cleared rapidly from the upper respiratory regions, delivering little radiation dose to the lung tissues. Small particles are deposited deeper in the lung and are cleared very slowly, producing a much larger dose per unit activity inhaled.

- **Enrichment.** Enrichment, or specific activity of the isotope mixture, often determines the relative importance of radiological and chemical toxicity for more soluble materials.

9.2.1.4 Developing Event Scenarios

Properties of the hazardous material do not significantly affect the manner in which this step of the hazards assessment is performed.

9.2.1.5 Estimating Potential Event Consequences

For the scenarios developed in the previous step, this step determines the area potentially affected, the need for protective actions, and the time available to take those actions. The way these consequences are determined will depend on properties of the hazardous material. For uranium and its compounds, the following possibilities should be considered.

- **Model types.** Depending on the relative significance of radiological and chemical toxicity, the analyst may need to calculate either radiation dose, air concentration, or both for the postulated releases. For a specific scenario, different models may be needed to analyze different consequences to determine which effect is limiting (for example, radiation dose, soluble uranium intake, or HF concentration).

- **Model features.** For reactive species, the ability to model the transformation and depletion of material during transport is important to a sound analysis. Because the hydrated uranyl fluoride formed by hydrolysis of UF_6 is a solid, some will be lost due to gravitational settling as a plume moves away from the release point. When analyzing consequences of a postulated accidental criticality, correcting for the decay during transport of the short-lived fission product gases will produce a more accurate assessment of consequences.

9.2.2 Program Elements

Properties and characteristics of uranium and its compounds will also need to be considered in formulating the emergency management program elements. Following are specific program element considerations related to the hazardous properties of uranium.

9.2.2.1 Emergency Response Organization

The primary influence of uranium's hazardous properties on the Emergency Response Organization (ERO) is in the staffing of the consequence assessment component. As will be discussed below in Consequence Assessment, staff assigned to the ERO should be knowledgeable of, and able to quantitatively evaluate, both the health physics (radiological) and industrial hygiene (non-radiological) aspects of the hazard.

9.2.2.2 Offsite Response Interfaces

The specific properties of the hazardous material do not significantly affect the content of this program element.

9.2.2.3 Operational Emergency Event Classes

As with all hazardous materials, classification of emergencies for uranium facilities should be based on the predicted consequences at specific receptor locations, as compared with numerical criteria for taking protective action (dose, exposure, air concentration). If a material has two or more recognized modes of effect and associated protective action criteria, classification decisions should be based on the more limiting one.

Example: The postulated release of a quantity of a uranium compound will produce a radiological consequence corresponding to the classification criterion for Alert. The chemical toxicity of the uranium compound is such that the non-radiological consequence exceeds the criterion for Site Area Emergency. The postulated release should be classified as a Site Area Emergency.

The appropriate classification for the postulated event or condition should be determined during the hazards assessment process and the observable features and indications identified as Emergency Action Levels (EALs) for that event/condition.

9.2.2.4 Notification

The specific properties of the hazardous material do not significantly affect the content of this program element.

9.2.2.5 Consequence Assessment

As discussed above, models and calculational methods used for consequence assessment should be appropriate physical, chemical, and radiological properties of the hazards. Models used to calculate and project the radiological and non-radiological consequences of a release of uranium and its compounds should be the same ones used in the hazards assessment process. If the same models are not used, the differences between outputs should be characterized and documented to avoid the potential for confusion and indecision during response to an actual emergency.

Environmental monitoring capability for assessing consequences of a uranium release should conform to several general principles.

- Procedures for measurement of airborne uranium should provide for timely analysis and reporting of results in units that correspond to decision criteria. Decision points based on initial screening measurements with field instruments should account for the expected levels of radon progeny collected on the air sample media. Portable survey instruments capable of performing alpha spectroscopy measurements can be used to provide rapid isotopic analysis of uranium collected on sample media.

- Measurement of uranium deposition should be planned and proceduralized to yield results that correspond to those produced by the predictive models used for emergency response. The correlation between direct or indirect radioactivity measurements (in units of activity) and measurement methods that give mass or concentration of uranium in a sample should be established for the expected enrichment values of material that might be released.

- If the potential exists for release of uranium compounds with high chemical toxicity (such as UF_6), it is not practical to plan to use survey teams to quantify concentrations in a plume. The high risk to survey personnel, the protective equipment necessary to minimize that risk, the time needed to prepare and position a team for such a survey, and the limited value of the information that could be gained all weigh against this approach to assessing the consequence of a highly toxic release.

- Environmental air concentrations are commonly measured continuously around the perimeter of some uranium facilities. Consequence assessment procedures should provide for the rapid retrieval and analysis of sample media from fixed samplers that may be operating in an area affected by a uranium release.

9.2.2.6 Protective Actions

Because the health consequences of a given intake of uranium, or its compounds, are highly dependent on properties such as enrichment, particle size, and solubility, facility and site-specific protective action criteria stated in terms of observable quantities and features of the release should be developed. In order for protective action criteria stated in terms of calculated dose or concentration to be valid, the calculational models should account for the properties of the material.

9.2.2.7 Medical Support

If the potential exists for significant uranium intakes, the emergency management program should include specific planning for the quantification of exposure, diagnosis of health effects, and treatment. Medical facilities providing emergency medical support should be provided with references relating to uranium toxicity and treatment protocols. Criteria for implementing treatments such as surgical excision of contaminated tissue or use of chelating agents should be discussed with the medical staff and sources of real-time advice and assistance be identified.

9.2.2.8 Recovery and Reentry

The specific properties of the hazardous material do not significantly affect the content of this program element.

9.2.2.9 Public Information

The specific properties of the hazardous material do not significantly affect the content of this program element.

9.2.2.10 Emergency Facilities and Equipment

Except for instruments and analysis methods used in consequence assessment, specialized facilities and equipment will not be required to meet the emergency management program needs of uranium facilities. Equipment and analytical techniques for detection and measurement of uranium in environmental sample media should have sufficient sensitivity to measure levels at or below those corresponding to decision criteria. Whereas larger sample sizes or longer counting times may be used to reduce the limit of detection for routine environmental surveillance, time constraints may dictate that more sensitive techniques be used for emergency response. Kinetic phosporimetry, a fast, sensitive, and accurate method for direct determination of uranium, permits analysis of many sample media directly or with limited sample preparation.

9.2.2.11 Training, Drills and Exercises, Program Administration

The specific properties of the hazardous material do not significantly affect the content of these program elements.

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10.0 DECONTAMINATION AND DECOMMISSIONING

At the end of the useful life of a facility, activities are undertaken to restore the facility to non-contaminated status and permit its unrestricted use. These activities are typically termed decontamination and decommissioning (D&D).

Although uranium facilities are no longer useful and operational activities are no longer conducted, measures must be continued to control the residual radioactivity. The decision may be made to undertake a D&D program to minimize or eliminate long-term institutional control. This may be done in a variety of ways, most of which may be termed D&D. The exception is converting the facility to some other nuclear use. With the elimination of the DOE weapons production mission, more uranium-contaminated facilities will require D&D in the near future.

This section provides guidance on establishing and implementing an effective D&D program. Major topic areas include regulations and standards, design features, D&D program, D&D techniques, and D&D experience. The following subsections concentrate on the radiation protection aspects of D&D at uranium-contaminated DOE facilities.

10.1 REGULATIONS AND STANDARDS

The standards that apply to the decommissioning of a uranium-contaminated facility include virtually all of those that were applicable during facility operations, plus some additional ones. The occupational safety and radiation protection regulations, radioactive and hazardous chemical disposal regulations, and transportation requirements are unaffected by the activity to which they apply.

No single DOE regulation covers all D&D requirements due to the wide variety of issues encompassed by D&D. These issues include project management, environmental surveillance, health and safety of workers and the public, engineering design, characterization survey techniques, D&D techniques, waste management, and waste transport. The primary DOE Orders pertaining to D&D activities are DOE O 435.1,Ch.1, Radioactive Waste Management (DOE 2001b); DOE O 231.1A Environment Safety and Health Reporting (DOE 2003d); DOE O 420.1A Facility Safety (DOE 2002). The DOE operations offices may have implementation procedures corresponding to these Orders with which contractors will also need to comply.

Section 5 of DOE O 435.1,Ch.1, Radioactive Waste Management, is the primary DOE Order dealing with decommissioning of radioactively contaminated facilities. It requires that DOE organizations develop and document their programs to provide for the surveillance, maintenance, and decommissioning of contaminated facilities. Requirements are divided into the following categories: general, facility design, post-operational activities, decommissioning project activities, and quality assurance. These categories are discussed below in Section 10.3.

Section V of DOE Order 5400.5,Ch.2, Radiation Protection of the Public and Environment, provides radiological protection requirements and guidelines for cleanup of residual radioactive material and management of the resulting wastes and residues and release of property. This DOE Order establishes a basic public dose limit of 100-mrem (1-mSv) effective dose equivalent in a year for exposure to residual radioactive material (in addition to naturally occurring "background" exposures).

10.1.1 Other Regulations

D&D of most uranium-contaminated facilities will involve cleanup of a combination of radioactive wastes, hazardous wastes, and mixed wastes. Federal regulations that are applicable to the cleanup and disposal of these wastes are summarized in this section, along with the DOE guidance on implementation of the following regulations:

10.1.1.1 National Environmental Policy Act (NEPA) (USC, 1970) and 40 CFR 1500 (CEQ, 1992)

This act established a national policy to ensure that environmental factors are considered in any Federal agency's planning and decision-making. DOE O 451.1B, Ch.1, National Environmental Policy Act Compliance Program (DOE 2001c), defines DOE responsibilities and procedures to implement NEPA. The decommissioning of a DOE uranium facility will require a determination of whether or not the action is a "major or significant government action adversely affecting the environment" in accordance with NEPA. If it qualifies as such an action, an environmental assessment (EA) or environmental impact statement (EIS) will be required. The EA or EIS must discuss the amount of material that will remain onsite and its effect, in addition to addressing the alternatives. The alternatives will include retaining radioactive material onsite under DOE control, cleaning the site to a level that would be acceptable for unrestricted release, and the null or no-action alternative of "walking away" from the site. If the action does not require an EA or EIS, either because the possible adverse effects are insignificant or because decommissioning was adequately addressed in a pre-operational or other EA or EIS, then the decommissioning can proceed in accordance with the information contained in other applicable regulations.

10.1.1.3 Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) (USC 1980) and 40 CFR 300 (EPA 1992a)

This act requires the identification and cleanup of inactive hazardous waste sites by responsible parties, and imposes certain response and reporting requirements for releases of hazardous substances.

10.1.1.4 Superfund Amendments and Reauthorization Act (SARA) and 40 CFR 300.

Interagency agreements can also exist between DOE, EPA, state, and local agencies (Daugherty 1993). Any special arrangement agreed to as part of an interagency agreement will need to be honored during the D&D activities.

Additional guidance relating to regulations and standards for D&D activities at uranium facilities may be found in:

DOE-HDBK-1113-98, Radiological Safety Training for Uranium Facilities (DOE 1998c)
DOE-HDBK-1132-99, Design Considerations (DOE 1999o)
DOE-STD-3007-93 (including Change Notice 1, September 1998), Guidelines for Preparing Criticality Safety Evaluations at DOE Non-Reactor Nuclear Facilities (DOE 1993b) DOE-STD-1120-98, Integration of Environment, Safety, and Health into Facility Disposition Activities (DOE 1998f)

10.1.2 Residual Radioactivity Levels

A primary concern in the D&D of any nuclear facility is the level of residual radioactivity that may be permitted for unrestricted use. The U.S. Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors (AEC 1974) provides definitive values for acceptable surface contamination levels for termination of operating licenses for nuclear reactors and for materials, equipment, and facilities. This document is based on the methodology of ICRP Publication 2, ICRP Publication 26 and ICRP Publication 30. More current guidance material includes: ANSI/HPS N13.12 (ANSI 1999); NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM), December 1997; November 17, 1995 memo from RF Pelletier entitled *Application of DOE requirements for release and control of property containing residual radioactive material ;* NUREG-1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions*, December 1997.

Congress has mandated that the EPA develop guidelines that will be applicable to all nuclear facilities as well as to the release of formerly contaminated or controlled radioactive facilities for unrestricted release. Such guidelines will likely be based on the radiation dose to the maximum exposed member of the general population. The maximum allowable annual dose has not yet been determined, but values of 50, 10, 1, and 0.1 mrem/y are being considered by the EPA as the "de facto de minimis" levels for the disposal of contaminated material.

Section 4 of DOE Order 5400.5,Ch.2, Radiation Protection of the Public and Environment, provides the following DOE guidelines for cleanup of residual radioactive material, management of the resulting wastes, and release of property. The basic public dose limit for exposure to residual radioactive material in addition to natural background exposures is 100-mrem (1-mSv) effective dose equivalent in a year. The effective dose equivalent in a year is the sum of the effective dose equivalent from exposures to radiation sources external to the body during the year plus the cumulative effective dose equivalent

(CEDE) from radionuclides taken into the body during the year. See DOE/CH-8901, A Manual for Implementing Residual Radioactive Material Guidelines (DOE 1989b), for procedures for deriving specific property guidelines for allowable levels of residual radioactive material, based on the dose limit of 100 mrem (1 mSv).

DOE Order 5400.5,Ch.2, also provides the following guidelines for residual concentrations of radionuclides in soil, concentrations of airborne radon decay products, external gamma radiation, surface contamination, and radionuclide concentrations in air or water.

10.1.2.1 Residual Radionuclides in Soil

Generic guidelines for thorium and radium(^{226}Ra, ^{228}Ra, ^{230}Th, and ^{232}Th) are 5 pCi/g averaged over the first 15 cm of soil below the surface and 15 pCi/g averaged over 15-cm-thick layers of soil more than 15 cm below the surface. For other radionuclides in soil (e.g., uranium), specific guidelines shall be derived from the basic dose limit by means of an environmental pathway analysis using specific property data where available. Residual concentrations of radioactive material in soil are defined as those in excess of background concentrations averaged over an area of 100m^2.

10.1.2.2 Airborne Radon Decay Products

Applicable generic guidelines are found in 40 CFR 192 (EPA 1992b). In any occupied or habitable building, the objective of remedial action shall be, and a reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) not to exceed 0.02 WL. Remedial actions are not required to comply with this guideline when there is reasonable assurance that residual radioactive material is not the source of the radon concentration.

10.1.2.3 External Gamma Radiation

The average level of gamma radiation inside a building or habitable structure on a site to be released without restrictions shall not exceed the background level by more than 20 uR/h.

10.1.2.4 Surface Contamination

DOE guidelines on surface contamination levels are similar to those in Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, except that no guidance is provided for surface contamination levels of transuranics. DOE is holding the TRU limit in reserve pending development of standards more clearly applicable to DOE facilities. For TRU waste, the limits in Regulatory Guide 1.86 are as follows:

Regulatory Guide 1.86 Limits

Removable Contamination	20 dpm/100 cm^2
Fixed Contamination	100 dpm/100 cm^2
Maximum Contamination	300 dpm/100 cm^2

10.1.2.5 Residual Radionuclides in Air and Water

Residual concentrations of radionuclides in air shall not cause members of the public to receive an effective dose equivalent greater than 10 mrem (0.1 mSv) in 1 year (DOE Order 5400.5, Ch.2). In 40 CFR 141, National Interim Primary Drinking Water Regulations (EPA 1992c), EPA provides a limit of 4 mrem/y annual dose equivalent to the whole body or any internal organ of any member of the public from manmade radionuclides in drinking water.

NRC is updating their decommissioning regulations and criteria. The new NRC methodology is presented in NUREG/CR-5512 (NRC 1992c) and is consistent with the recommendations in ICRP Publications 26 and 30. NRC will establish a three-layered hierarchy for developing decommissioning criteria to be used to evaluate the release of property after D&D. The first and second layers use conservative models and parameters, and the third layer uses site-specific models and data to provide a more accurate approximation of actual conditions. Four criteria will be calculated upon which to make D&D decisions: 1) a surface contamination level for buildings and building materials (in dpm/100 cm^2), 2) volume contamination criteria for volume sources in buildings (in pCi/g), 3) soil contamination criteria (in pCi/g), and 4) a total site inventory (in Ci).

These criteria require calculation of dose to members of the general population. The exposure scenarios will have to include all exposure pathways that are credible under the proposed disposition. If the site is part of a closely guarded government reservation, certain pathways may be eliminated, such as the use of well water directly from the site and ingestion of significant quantities of fruits and vegetables grown on the site. However, if the site will be released for unrestricted use, such scenarios should be considered. The computer codes used for calculation of dose to the public from decommissioned facilities will include the currently accepted exposure models and site-specific or maximum credible parameters for exposure pathways.

10.2 DESIGN FEATURES FOR NEW FACILITIES

Design of the facility should allow easy D&D of equipment and materials. Details on designing facilities for ease of decommissioning are discussed in the following sections.

10.2.1 Building Materials

In general, the design features that aid in contamination control during operation also facilitate decommissioning. The inclusion of all the building materials suggested in this section may be cost-prohibitive, but they should be considered if the budget allows. Maintenance procedures that are used during operation are also important in controlling the spread of contamination to clean areas and, therefore, facilitate decommissioning, too.

Less permeable building materials are more easily decontaminated. Any concrete with uncoated surfaces that comes in contact with uranium solutions or uranium-contaminated air will require surface removal and disposal as radioactive waste at the end of its life. If there are cracks through which contaminated solutions have penetrated, the entire structure may need to be disposed of as radioactive waste.

Metal surfaces may also require decontamination. In general, the more highly polished the surface, the easier it will be to decontaminate. If feasible, all stainless steel that will come into contact with

uranium should be electro-polished before being placed into service. If high-efficiency particulate air (HEPA) filtration has failed at any time during facility operation, roofs may require decontamination. Metal roofs are easiest to decontaminate, but even these may contribute to the volume of radioactive waste unless unusual measures are taken to clean them. Built-up and composition roofs will be difficult to clean to unrestricted release levels.

Interior surfaces are most easily cleaned if they were completely primed and painted before the introduction of radioactive materials into the facility. If interior surfaces are repainted during operation, their disposal as clean waste is likely to require removal of the paint. However, if the paint has deteriorated, cleaning for unrestricted use may be as difficult as if the material had never been painted. Wood will almost certainly become contaminated, as will plasterboard and other such materials.

Floor surfaces are likely to be a problem. Concrete should be well sealed and covered with a protective surface. Single sheet, vinyl flooring with heat-sealed seams is preferable to asphalt or vinyl tile because it is more easily cleaned. If the floor needs resurfacing, it is preferable to overlay new flooring material rather than remove the old material and expose the underlying floor.

Carpets are not recommended because they are difficult to clean and survey and bulky to dispose of and they do not adequately protect the underlying surface. In some areas, such as control rooms, their use may be justified by noise control requirements; however, their contamination control limitations should be considered. If used, carpets should be surveyed frequently and disposed of as radioactive waste when they become contaminated.

10.2.2 Ventilation Systems

In addition to decommissioning considerations, the design of the ventilation system will depend on the operations conducted in the facility. Adequate air flow for all operations and good design practices will help keep the facility clean during operations and will facilitate decommissioning. Fiberglass duct work may present a fire hazard and may be more difficult to decontaminate than stainless steel, especially stainless steel that has been electro-polished. Welded joints are less likely to collect contamination than bolted ones; however, bolted joints are easier to remove and the most contaminated areas are readily accessible for cleaning.

Filters should be positioned in ventilation systems to minimize contamination of ductwork (e.g., filtration of glove-box exhaust air before it enters a duct leading to a plenum).

10.2.3 Piping Systems

Potentially contaminated piping systems imbedded in concrete are a common and relatively expensive decommissioning problem. Most often, they must be sealed and removed last, after all other radioactive material has been removed and the building is being demolished by conventional methods. Often, they provide the major impetus for demolishing a building rather than converting it to some non-nuclear use. For this reason, it is best to run pipes in chases or tunnels that have been lined (usually with stainless steel) to prevent contamination from penetrating building surfaces. To minimize hand jackhammer work required during decommissioning, floor drains should not be enclosed in concrete.

10.2.4 Soil-Contamination Considerations

Depending on the activity levels found, locations where contaminated effluents have penetrated the ground may require excavation during decommissioning. The facility design should minimize such areas. Particular attention should be paid to storm runoff from roofs, storage areas, contaminated equipment storage, and liquid waste treatment impoundments (including sanitary sewage systems if they may receive some small amount of contamination during the life of the facility).

10.2.5 Other Features

Installed decontamination and materials-handling equipment that facilitates operation and maintenance also generally facilitates decommissioning in two ways. First, it can be used for its intended purposes of cleaning and moving equipment during the decommissioning phase. Even more important, it usually contributes to a cleaner, better-maintained facility, where nonfunctional equipment is moved out when it is no longer needed and work surfaces are kept free of spreadable contamination.

Other features include the following:

a. minimizing service piping, conduits, and ductwork,

b. caulking or sealing all cracks, crevices, and joints,

c. using modular, separable confinements for radioactive or other hazardous materials to preclude contamination of fixed portions of the structure,

d. using localized liquid transfer systems that avoid long runs of buried contaminated piping,

e. using equipment that precludes the accumulation of radioactive or other hazardous materials in relatively inaccessible areas, including curves and turns in piping and ductwork,

f. using designs that ease cut-up, dismantling, removal, and packaging of contaminated equipment from the facility,

g. using modular radiation shielding, in lieu of or in addition to monolithic shielding walls,

h. using lifting lugs on large tanks and equipment, and

i. using fully drainable piping systems that carry contaminated or potentially contaminated liquids.

10.3 DECONTAMINATION AND DECOMMISSIONING PROGRAM REQUIREMENTS

The basic requirements for a D&D program at a DOE facility are found in Section 5 of DOE O 435.1, Ch.1, Radioactive Waste Management. Planning for facility decommissioning shall be initiated during the design phase for new facilities and before termination of operations for existing operational facilities. DOE O 435.1, Ch.1, divides the discussion of requirements into several time periods: pre-operational and operational activities, post-operational activities, D&D activities, and post-decommissioning activities. Requirements for each of these time periods and for quality assurance are presented in the following sections.

10.3.1 Pre-Operational and Operational Activities

Determination of the natural background levels of radiation and of the background and fallout radionuclides is a critical step in decommissioning. These levels are best determined before the facility becomes operational. These levels need to be determined so the incremental dose occurring from material left onsite at the termination of operations can be assessed.

The contamination control practices and records maintained during facility operation will also be important. If paint is used in contamination fixation (seldom an optimum, but sometimes a necessary, practice), it should be of a distinctive color and the location should be permanently recorded. Other records are also helpful in planning and executing final decontamination for dismantling. Spills, pipe and tank leaks, ventilation failures, burial of low-level radioactive or potentially radioactive materials onsite, or other actions that might affect decommissioning shall become part of the permanent record of the facility and be considered in decommissioning planning. Insights from workers who worked at the facility during the operational phase can also provide useful information on past incidents.

10.3.2 Post-Operational Activities

DOE program organizations shall identify contaminated facilities under their jurisdiction, document the potential for reuse and recovery of materials and equipment, and develop decommissioning schedules. Before decommissioning activities begin, adequate surveillance and maintenance should be performed for inactive facilities that allow them to: 1) meet applicable radiation protection, hazardous chemical, and safety standards, 2) maintain physical safety and security standards, and 3) reduce potential public and environmental hazards. Deactivation operations, such as removing all high-level waste and stored hazardous materials, should be performed by the facility operator as part of the last operational activities before entering into the decommissioning phase.

10.3.3 Decontamination and Decommissioning Activities

The following discussion of D&D activities is divided into four phases: site characterization, environmental review process, general decommissioning planning, decommissioning project plan, and decommissioning operations.

10.3.3.1 Site Characterization

Characterization data shall be collected to support a thorough physical, chemical, and radiological characterization to fulfill the requirements of NEPA reviews, and the RCRA, CERCLA, and SARA preliminary assessment/site investigations and detailed engineering. The facility characterization shall include the following:

a. drawings, photographs, or other records reflecting the as-built and as-modified condition of the facility and grounds,

b. the condition of all structures, existing protective barriers, and systems installed to ensure public, occupational, and environmental safety,

c. the type, form, quantity, and location of hazardous chemical and radioactive material from past operations at the site, and

d. information on factors that could influence the selection of decommissioning alternatives (safe storage, entombment, dismantling), such as potential future use, long-range site plans, facility condition, and potential health, safety, and environmental hazards.

One portion of the site-characterization process is a composite of several different types of surveys: background, scoping (or preliminary site characterization), and detailed characterization, as defined by Berger (1992). Guidance for conducting site characterizations as part of the remedial investigations/feasibility studies (RI/FS) under CERCLA can be found in EPA/540/G-89/004, Guidance for Conducting Remedial Investigations and Feasibility Studies Under CERCLA (EPA 1988b).

The background survey information (i.e., direct radiation levels and concentrations of potential radionuclide contaminants in construction materials and soils) may be performed as part of the environmental baseline studies during pre-operational activities. Otherwise, background levels should be determined at onsite or immediately offsite locations that are unaffected by operations.

The scoping or preliminary site characterization study should be performed to identify the potential radionuclide contaminants at the site, the relative ratios of these nuclides, and the general extent of contamination. The survey provides the basis for initial estimates of the required decommissioning effort and a framework for planning the more detailed characterization study. A limited number of measurements will be made at locations that are most likely to have contamination. Scoping or preliminary site characterization surveys may be conducted during the post-operational phase.

The detailed characterization survey will more precisely define the extent and magnitude of contamination. The resulting data will be used to assist in planning for the decontamination effort, including decontamination techniques and health and safety considerations during decommissioning.

10.3.3.2 Environmental Review Process

Candidate decommissioning alternatives should be identified, assessed, and evaluated, and preferred decommissioning alternatives should be selected based on the results of the environmental review. The review should be performed according to the requirements of NEPA, RCRA, CERCLA, and SARA. Depending on the operation, the environmental review may consist of an EA or an EIS (see Section 10.2.1).

10.3.3.3 Decommissioning Planning

The first step in decommissioning planning is the development of a series of absolute criteria. These will necessarily include such items as compliance with DOE Orders, EPA regulations, interagency agreements, and other statutes. They may also include commitments to states, landowners, or others, or provisions of the original EIS.

As these criteria are developed, other high-value criteria may also be established. These are likely to include such considerations as maximizing the aesthetic and recreational value of the site, performing decommissioning within allocated funds, lowest worker dose, lowest population dose, lowest cost, lowest future surveillance commitment, and least effect in case of probable accidents. Depending on the viability of the decommissioning action, the decision-making process that has been established, and the level of public concern, notice of a scoping meeting may be published in the Federal Register and

scoping meetings may be held. Similar actions may be taken to determine the applicable decommissioning criteria and the alternatives to be considered.

Whether or not a formal scoping meeting and EIS are used, it will be necessary to define the D&D options to be considered. Most of the analysis effort should be expended on those options that fulfill the absolute criteria so they can be ranked relative to the other high-value criteria. General options would typically include the following, which are taken from NUREG-0586, Final Generic Environmental Impact Statement (GEIS) on Decommissioning of Nuclear Facilities (NRC 1988):

a. **Decontamination (DECON) -** Decontamination is the alternative in which contaminated equipment, structures, and portions of a facility are physically removed from the site or their radioactive contaminants are removed by chemical or abrasive means. This alternative is the preferred approach to decommissioning uranium-contaminated facilities.

b. **Safe storage (SAFSTOR) -** SAFSTOR is sometimes referred to as "deferred decommissioning," the alternative in which nuclear facilities are placed and maintained in such a condition that the structure and contents can be safely stored and eventually decommissioned. In preparing a facility for SAFSTOR, the structure is left intact, but all nuclear fuels, radioactive fluids, and wastes are removed from the site. This alternative is generally considered when the following conditions occur:

1. Low-level waste disposal capacity is inadequate to implement DECON.

2. An adjacent operating nuclear facility would be adversely affected if the DECON alternative were implemented.

3. A positive benefit would be derived through a limited period of radioactive decay. A cost-benefit analysis should be performed, comparing total cost and radiation exposure resulting from DECON versus SAFSTOR. Then, a decision should be made whether any additional costs incurred for the SAFSTOR alternative are justified by the dose savings. Due to the long half-lives of uranium isotopes, radioactive decay is not a viable reason for using the SAFSTOR D&D option.

c. **Entombment (ENTOMB) -** The entombment alternative involves removing all nuclear fuels, radioactive fluids, and wastes from the site and encasing all structural and mechanical materials and components not decontaminated to acceptable levels in a structurally long-lived material, such as concrete. The entombed structure is maintained under appropriate continued surveillance until the radioactivity decays to a level permitting unrestricted release of the facility. The maximum allowable time in entombment should be less than 100 years. Due to the long half-lives of uranium isotopes, entombment is not a viable option for decommissioning of uranium-contaminated facilities.

d. **The no-action alternative, as required by NEPA -** In decommissioning, this is normally considered the "walk away" option.

Conversion of a facility for alternate nuclear or other controlled use has sometimes been considered a decommissioning mode; however, it is not truly decommissioned unless conversion involves removal of all radioactive material. Final disposition, when it occurs at the end of the new use, should consider the residual radioactivity onsite.

10.3.3.4 Decommissioning Project Plan

A decommissioning project plan should be prepared and should include the following:

a. physical, chemical, and radiological characterization data or references to such data,

b. a summary evaluation of decommissioning alternatives for the facility, including the preferred alternatives,

c. plans for meeting requirements from the environmental review process (NEPA, RCRA, CERCLA, and SARA),

d. radiological criteria to be used,

e. development of a health and safety plan for decommissioning,

f. projections of occupational exposure,

g. estimated quantities of radioactive waste to be generated, and

h. detailed administrative, cost-schedule, and management information.

If a contractor will be used to perform the D&D operations, the plan should include detailed technical specifications for selecting a contractor.

The site characterization survey should provide the necessary information on the type of facility or land area to be decommissioned and the type and amount of residual radioactive material that must be cleaned up. Other information to be considered in deciding the appropriate decommissioning alternative includes the following:

a. the availability of a final disposal facility for the radioactive waste, hazardous waste, or mixed waste,

b. the intended use of the site and components (e.g., Will the site be released for unrestricted or restricted use?),

c. the site characteristics (e.g., demography, accessibility, meteorology),

d. the CBA results, and

e. the resource considerations.

10.3.3.5 Decommissioning Operations

Decommissioning operations shall be conducted according to the approved decommissioning project plan. Significant deviations from the decommissioning project plan should be approved by the responsible field organization in consultation with the appropriate program office.

During decommissioning operations, remediation control surveys (Berger 1992) should be conducted to guide the cleanup in the real-time mode. This will ensure that the decommissioning workers, the public, and the environment are all adequately protected against exposures to radiation and radioactive materials arising from the decommissioning activities.

The volume of waste and the associated cost of decommissioning the waste will be greatly reduced if equipment can be cleaned up and disposed of as either non-radioactive waste or as non-TRU waste. Numerous techniques have been developed for decontamination of equipment and materials. Established techniques and the latest technology should be considered in minimizing the quantity of contaminated equipment that requires disposal and the waste generated from the decontamination processes. These techniques are described in Section 10.4.2.

In establishing a radiological control program for decommissioning operations, the scope of the decommissioning effort should be identified. Factors to be considered in program development include:

a. the type of facility or land area to be cleaned up,

b. the type and amount of radioactive contaminated material, hazardous waste, and mixed waste,

c. the radiological and hazardous material cleanup levels, and

d. the decommissioning methods being used.

The extent of the radiological control program will depend on the selected decommissioning alternative. For the SAFSTOR alternative, the radiological control program would be minimal following deactivation (i.e., surveillance activities) until the decontamination phase is initiated, at which time a full radiological control program would be necessary. For the DECON alternative, a fully staffed radiological control program would be needed from the start of decontamination. Typically, this program would be similar to the program conducted during normal operations. Entombment is not a viable alternative for decommissioning of uranium-contaminated facilities.

Also, the hazardous and radioactive contaminants present and the specific decontamination techniques (e.g., mechanical methods, high-pressure water, abrasive cleaning, vibratory finishing, ultrasonics, electro-polishing, decontamination foams, strippable decontamination coatings, and dry ice

blasting) used by each alternative will affect the extent of the radiological control program. For example, if an abrasive mechanical technique for decontaminating equipment (where airborne concentrations may be a concern) is chosen over just scrapping the equipment as waste, obviously the radiological control program will need to be more sophisticated.

10.3.4 Post-Decommissioning Activities

A final radiological and chemical survey report (or an independent verification survey report) and a project final report should be prepared. The final report should include a description of the project, the final status of the property, and the lessons learned from the project.

As defined in Berger (Berger 1992), confirmatory surveys may be performed by the regulatory agency to confirm the adequacy of the contractor's final radiological and chemical survey report. A confirmatory survey typically addresses from 1% to 10% of the site.

The responsible program organization should ensure any necessary long-term maintenance and surveillance or other safety controls are provided for the decommissioned property. The decommissioned property may be released from DOE ownership if the responsible program organization, in consultation with the Office of the Assistant Secretary for Environment, Safety, and Health (EH-1), certifies that the property meets applicable release criteria for residual radioactivity and hazardous chemicals. If appropriate release criteria are not met, the property may be reused for other program activities that may or may not involve radioactive or hazardous materials, provided adequate safety controls are maintained (see Section 5, 3.d.(5), of DOE Order 5820.2A).

10.3.5 Quality Assurance

Decommissioning activities shall be conducted according to the applicable requirements of the ANSI/ASME NQA-1, Quality Assurance Program Requirements for Nuclear Facilities (ANSI 1989) and other appropriate national consensus standards (e.g., EPA guidance documents in the EPA QA/R and EPA QA/G series should be used in the design of environmental monitoring programs). The quality assurance program for D&D activities should follow the guidelines in DOE Order 414.1B, Quality Assurance.

10.4 DECONTAMINATION AND DECOMMISSIONING EXPERIENCE

Discussions of D&D activities at several uranium facilities can be found in (Adkisson 1987) and (Wynveen et al. 1982). D&D activities took place in several types of uranium facilities including an enriched uranium fuel fabrication plant, a mixed oxide (Pu/U) fuel fabrication and development plant, a research and development laboratory, and a depleted uranium manufacturing plant. Equipment decontaminated, dismantled, or removed included glove boxes, fume hoods, laboratory equipment, piping, ventilation ducts, uranium and thorium sediments from a settling lagoon, and soil from a small shallow burial area. Decontamination techniques included wiping with a damp cloth, strippable paint, acid wash, and removal of soil and sediments. Some lessons learned from these D&D operations included the following:

a. Waste management planning should begin early in the D&D planning stages and account for the possibility there may be more stringent regulations for shipping hazardous or radioactive wastes

than disposing of it. Any waste package designs need to be reviewed to ensure compliance with all applicable waste management requirements.

b. Temporary contamination enclosures are effective in controlling contamination during size reduction of large equipment such as glove boxes. Any loose contamination on the equipment should be fixed prior to placing it in the enclosure.

1. Criticality safety issues should be considered regarding the geometry of any waste material containing fissile material.

2. D&D operations must be prepared for changes in regulatory criteria and implementation of these new criteria.

3. During decommissioning operations, personnel need to recognize the possibility of encountering elevated levels of contamination in unexpected locations such as the excavations for concrete structures or under existing roofing or flooring (Bernhardt et al., 1989).

4. Bernhardt et al. (1989) stressed the importance of establishing and documenting criteria for implementing regulations. For example, in meeting surface contamination guidelines, it is important to establish the acceptable detection efficiency of the detector and areas for averaging measurements.

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APPENDIX A

GLOSSARY

abnormal situation: Unplanned event or condition that adversely affects, potentially affects, or indicates degradation in the safety, security, environmental or health protection performance or operation of a facility.

absorbed dose: The energy absorbed by matter from ionizing radiation per unit mass of irradiated material at the place of interest in that material. The absorbed dose is expressed in units of rad (or gray) (1 rad = 0.01 gray). **(10 CFR 835)**

activity median aerodynamic diameter (AMAD): Fifty percent of the activity (aerodynamically classified) of aerodynamic diameter (e.g., the diameter of a unit density sphere that has the same terminal settling velocity in air as that the particle of interest) greater than the AMAD. **(ICRP 1994)**

air sampling: A form of air monitoring in which an air sample is collected and analyzed at a later time, sometimes referred to as retrospective air monitoring.

air monitoring: Actions to detect and quantify airborne radiological conditions by the collection of an air sample and the subsequent analysis either in real-time or in off-line laboratory analysis of the amount and type of radioactive material present in the workplace atmosphere.

airborne radioactive material: Radioactive material in any chemical or physical form that is dissolved, mixed, suspended, or otherwise entrained in air.

airborne radioactivity area: Any area, accessible to individuals, where:

(1) The concentration of airborne radioactivity, above natural background, exceeds or is likely to exceed the derived air concentration (DAC) values listed in appendix A or appendix C of this part; or

(2) An individual present in the area without respiratory protection could receive an intake exceeding 12 DAC-hours in a week. **(10 CFR 835)**

alarm set point: The count rate at which a continuous air monitor will alarm, usually set to correspond to a specific airborne radioactive material concentration by calculating the sample medium buildup rate.

ambient air: The general air in the area of interest (e.g., the general room atmosphere) as distinct from a specific stream or volume of air that may have different properties.

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 the reference man (ICRP Publication 23) that would result in a committed effective dose equivalent of 5 rem (0.05 sievert) or a committed dose equivalent of 50 rem (0.5 sievert) to any individual organ or tissue. ALI values for intake by ingestion and inhalation of selected radionuclides are based on Table 1 of the U.S. Environmental Protection Agency's Federal Guidance Report No. 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, published September 1988. **(10 CFR 835)**

as low as reasonably achievable (ALARA): The approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonable, taking into account social, technical, economic, practical, and public policy considerations. ALARA is not a dose limit but a process which has the objective of attaining doses as far below the applicable limits of 10 CFR 835 as is reasonably achievable. **(10 CFR 835)**

bioassay: The determination of kinds, quantities, or concentrations, and, in some cases, locations of radioactive material in the human body, whether by direct measurement or by analysis, and evaluation of radioactive materials excreted or removed from the human body. **(10 CFR 835)**

breathing zone air monitoring: A form of air monitoring that is used to detect and quantify the radiological conditions of air from the general volume of air breathed by the individual, usually at a height of 1 to 2 meters. See *personal air monitoring.* **(Air Monitoring Guide)**

contamination area: Any area, accessible to individuals, where removable surface contamination levels exceed or are likely to exceed the removable surface contamination values specified in appendix D of this part, but do not exceed 100 times those values. **(10 CFR 835)**

continuous air monitor (CAM): An instrument that continuously samples and measures the levels of airborne radioactive materials on a "real-time" basis and has alarm capabilities at preset alarm set points. **(Air Monitoring Guide)**

contractor: Any entity under contract with the Department of Energy with the responsibility to perform activities at a DOE site or facility. **(10 CFR 835)**

decontamination: The process of removing radioactive contamination and materials from personnel, equipment, or areas.

Department of Energy operations: Those activities funded by DOE for which DOE has enforcement authority over environmental, safety, and health protection requirements.

Department of Energy site: Either a tract owned by DOE or a tract leased or otherwise made available to the Federal Government under terms that afford to the Department of Energy rights of access and control substantially equal to those that the Department of Energy would possess if it were the holder of the fee (or pertinent interest therein) as agent of and on behalf of the Government. One or more DOE operations/program activities are carried out within the boundaries of the described tract.

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). For the radionuclides listed in Appendix C of this part, the air immersion DACs were calculated for a continuous, nonshielded exposure via immersion in a semi-infinite atmospheric cloud. The value is based upon the DAC found in Table 1 of the U.S. Environmental Protection Agency's Federal Guidance Report No. 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, published September 1988. **(10 CFR 835)**

detector: A device or component designed to produce a quantifiable response to ionizing radiation, normally measured electronically. **(Portable Monitoring Instrument Calibration Guide)**

DOELAP: The Department of Energy Laboratory Accreditation Program defines a set of reference performance tests and provides a description of the minimum levels of acceptable performance for personnel dosimetry systems and radiobioassay programs under DOE-STD-1111-98 (DOE 1998b). **(External Dosimetry Program Guide)**

dose: The general term for absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or total effective dose equivalent as defined in 10 CFR 835. **(10 CFR 835)**

exposure: The general condition of being subjected to ionizing radiation, such as by exposure to ionizing radiation from external sources or to ionizing 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. **(Internal Dosimetry Guide)**

fixed contamination: Radioactive material that has been deposited onto a surface and cannot be readily removed by nondestructive means, such as casual contact, wiping, brushing, or laundering. Fixed contamination does not include radioactive material that is present in a matrix, such as soil or cement, or radioactive material that has been induced in a material through activation processes. **(RCS)**

fixed-location sampler: An air sampler located at a fixed location in the workplace.

grab sampling: A single sample removed from the workplace air over a short time interval, typically less than 1 hour.

high contamination area: Any area, accessible to individuals, where removable surface contamination levels exceed or are likely to exceed 100 times the removable surface contamination values specified in appendix D of 10 CFR 835. **(10 CFR 835)**

high-efficiency particulate air (HEPA) filter: Throwaway extended pleated medium dry-type filter with 1) a rigid casing enclosing the full depth of the pleats, 2) a minimum particle removal efficiency of 99.97% for thermally generated monodisperse di-octyl phlalate smoke particles with a diameter of 0.3 µm, and 3) a maximum pressure drop of 1.0-in. w.g. when clean and operated at its rated airflow capacity. **(RCS)**.

high radiation area: Any area, accessible to individuals, in which radiation levels could result in an individual receiving a deep dose equivalent in excess of 0.1 rem (0.001 sievert) in 1 hour at 30 cm from the radiation source or from any surface that the radiation penetrates. **(10 CFR 835)**

intake: The amount of radionuclide taken into the body by inhalation, absorption through intact skin, injection, ingestion, or through wounds. Depending on the radionuclide involved, intakes may be reported in units of mass (e.g., µg, mg), activity (e.g., µCi, Bq), or potential alpha energy (e.g., MeV, J) units. **(Internal Dosimetry Program Guide)**

minimum detectable amount/activity (MDA): The smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability, B, 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 (Type I error). The MDA is computed using the same value of α as used for the decision level (DL). The MDA depends on both α and B. Measurement results are compared to the DL, not the MDA; the MDA is used to determine whether a program has adequate detection capability. The MDA will be greater than or equal to the DL. **(Internal Dosimetry Program Guide)**

occupational exposure: An individual's exposure to ionizing radiation (external and internal) as a result of that individual's work assignment. Occupational exposure does not include planned special exposures, exposure received as a medical patient, background radiation, or voluntary participation in medical research programs. **(10 CFR 835)**

personal air monitoring: A form of breathing zone air monitoring that involves the sampling of air in the immediate vicinity (typically within one foot) of an individual's nose and mouth, usually by a portable sampling pump and collection tube (e.g., a lapel sampler) worn on the body. **(Air Monitoring Guide)**

portable air sampler: An air sampler designed to be moved from area to area.

radiation area: Any area, accessible to individuals, in which radiation levels could result in an individual receiving a deep dose equivalent in excess of 0.005 rem (0.05 millisievert) in 1 hour at 30 cm from the source or from any surface that the radiation penetrates. **(10 CFR 835)**

radiation-generating device (RDG): The collective term for devices which produce ionizing radiation, including certain sealed radioactive sources, small particle accelerators used for single purpose applications which produce ionizing radiation (e.g., radiography), and electron-generating devices that produce x-rays incidentally. **(Radiation-Generating Devices Guide)**

radioactive material: Any material that spontaneously emits ionizing radiation (e.g., X- or gamma rays, alpha or beta particles, neutrons). The term "radioactive material" also includes materials onto which radioactive material is deposited or into which it is incorporated. For purposes of practicality, both 10 CFR 835 and this Standard establish certain threshold levels below which specified actions, such as posting, labeling, or individual monitoring, are not required. These threshold levels are usually expressed in terms of total activity or concentration, contamination levels, individual doses, or exposure rates. **(RCS)**

radioactive material area: Any area within a controlled area, accessible to individuals, in which items or containers of radioactive material exist and the total activity of radioactive material exceeds the applicable values provided in appendix E of 10 CFR 835. **(10 CFR 835)**

radiological area: Any area within a controlled area which must be posted as a "radiation area," "high radiation area," "very high radiation area," "contamination area," "high contamination area," or "airborne radioactivity area" in accordance with 10 CFR 835. **(10 CFR 835)**

radiological worker: A general employee whose job assignment involves operation of radiation-producing devices or working with radioactive materials, or who is likely to be routinely occupationally exposed above 0.1 rem (0.001 sievert) per year total effective dose equivalent. **(10 CFR 835)**

radiological work permit (RWP): The permit that identifies radiological conditions, establishes worker protection and monitoring requirements, and contains specific approvals for radiological work activities. The Radiological Work Permit serves as an administrative process for planning and controlling radiological work and informing the worker of the radiological conditions. **(RCS)**

radiological control organization: An organization responsible for radiation protection. **(Sealed Radioactive Source Accountability and Control Guide)**

real-time air monitoring: Collection and real-time analysis of the workplace atmosphere using continuous air monitors (CAMs).

refresher training: The training scheduled on the alternate year when full retraining is not completed for Radiological Worker I and Radiological Worker II personnel. **(RCS)**

removable contamination: Radioactive material that can be removed from surfaces by nondestructive means, such as casual contact, wiping, brushing, or washing. **(RCS)**

representative air sampling: The sampling of airborne radioactive material in a manner such that the sample collected closely approximates both the amount of activity and the physical and chemical properties (e.g., particle size and solubility) of the aerosol to which the workers may be exposed.

source-specific air sampling: Collection of an air sample near an actual or likely release point in a work area using fixed-location samplers or portable air samplers.

survey: An evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. When appropriate, such an evaluation includes a physical survey of the location of radioactive material and measurements or calculations of levels of radiation, or concentrations or quantities of radioactive material present. **(RCS)**

very high radiation area: Any area, accessible to individuals, in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rad (5 gray) in 1 hour at 1 meter from a radiation source or from any surface that the radiation penetrates. **(10 CFR 835)**

workplace monitoring: The measurement of radioactive material and/or direct radiation levels in areas that could be routinely occupied by workers.

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