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Exam Preview:

1. According to the reference material, lower-grade ores are mined commercially if they are a byproduct of mining for another material, such as gold or phosphate.
    a. True
    b. False
2. In depleted uranium, the beta radiation from the decay of ^{234}Th and ^{234m}Pa amounts to nearly ___ the alpha radiation from ^{238}U and ^{234}U.
    a. 5X
    b. 4X
    c. 3X
    d. 2X
3. Radiological control personnel should perform trend analyses on all permanent radiological areas. At a minimum, two complete survey records should be evaluated and included in the trend analysis program for each survey required to be performed by the facility routine control program.
    a. True
    b. False
4. 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 _____.
    a. 3 months
    b. 8 months
    c. 6 months
    d. 1 year

5. According to the reference material, in facilities with significant quantities of ^{99}Tc, radiation monitoring techniques must be able to detect the high-energy beta radiation from this isotope.
    a. True
    b. False
6. Several DOE facilities have adopted specifications on recycled uranium that limit the amount of transuranic alpha activity to ____% of the total uranium alpha activity, thus limiting the potential inhalation dose from transuranics to a small fraction of the total.
    a. 0.1
    b. 0.5
    c. 1.0
    d. 1.5
7. According to the ANSI N317 Specification, the portable monitoring instrument should have a minimum battery lifetime of ___ hours of continuous operation.
    a. 100
    b. 200
    c. 300
    d. 400
8. According to the reference material, a heavy metal, uranium is chemically toxic to kidneys and exposure to soluble (transportable) compounds can result in renal injury.
    a. True
    b. False
9. Using Table 2-3. Uranium Specific Activities, which of the following type of uranium has a Wt. % ^{235}U of 2?
    a. Natural
    b. Depleted
    c. Enriched
    d. Highly Enriched
10. Using Table 2-6. Health Effects from Acute Intake of Soluble Uranium, which of the uranium concentrations corresponds to a threshold for permanent renal damage in a 70 kg person?
    a. 4 mg
    b. 8 mg
    c. 2.1 mg
    d. 21 mg

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FOREWORD

This Technical Standard 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 Technical Standard provides guides to good practice, updates existing reference material, and discusses 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 Technical Standard provides information to assist uranium facilities in complying with Title 10 of the Code of Federal Regulations, Part 835 (10 CFR Part 835), *Occupational Radiation Protection* and various DOE Orders. This technical standard supplements DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835*, *Occupational Radiation Protection* (DOE, 2008a) and DOE-STD-1098-2008, *Radiological Control* (DOE, 2009c).

This Technical Standard has been updated to include provisions in the 2007 amendment to 10 CFR Part 835. This amendment updated the dosimetric terms and models for assessing radiation doses, both internal and external. Of particular interest for this Standard, the biological transportability of material is now classified in terms of absorption types: F (fast), M (medium) and S (slow). Previously this was classified in terms of material class: D (days), W (weeks) and Y (years). Throughout this Standard, discussions of previous studies describing the biological transportation of material in the body will continue to use D, W and Y, as appropriate. Discussions of other requirements which have not amended their dosimetric terms and models continue to use the older terminology.

This Technical Standard 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.

1 INTRODUCTION

1.1 Purpose and Applicability

This Technical Standard provides operational guidance, practical lessons learned and experience gained, guides to good practice, and reference information on the safe handling of uranium. The Technical Standard 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 Part 835). This Technical Standard supplements the DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835* (2008a); DOE Orders; and DOE-STD-1098-2008, *Radiological Control* (2009c), and its sole purpose is the protection of workers, the public, and the environment from the hazards that are inherent in uranium storage, processing, and handling.

This Technical Standard replaces DOE-STD-1136-2009, *Guide of Good Practices for Occupational Radiological Protection in Uranium Facilities* (DOE, 2009b), 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 Technical Standard 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 Technical Standard, in concert with applicable regulatory documents, will help in building a comprehensive and technically-defensible radiological control program.

1.2 Definitions

A glossary is provided in Appendix A. In all cases, the definitions provided in this Technical Standard are consistent with those provided in 10 CFR Part 835, its Guides, and DOE-STD-1098-2008 (2009c).

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 Dosimetry (Chapter 6), Nuclear Criticality Safety (Chapter 7), Waste Management (Chapter 8), Emergency Management (Chapter 9), and Decommissioning (Chapter 10).

2 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, to produce radioisotopes, to provide steam for propulsion, and radiation for research. 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 power 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). Heavy-water moderated 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 (g of Isotope per 100g of Material)

Isotope	Natural	Typical Commercial Feed Enrichment	Depleted	Specific Activity (Ci/g)	Neutron Capture Cross Section (barns)
^{238}U	99.28	97.01	99.8	3.3 E-7	2.7
^{235}U	0.72	2.96	0.2	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 from the increased ^{234}U than from the 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)]

			Energies (MeV) and Abundances of Major Radiations (Low yield radiation is not included)		
	Nuclide	**Half-Life**	**Alpha**	**Beta**	**Gamma**
Primary Uranium Isotopes	^{238}U	4.51×10^{9} y	4.15 (21%) 4.20 (79%)		
	^{235}U	7.1×10^{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	2.47×10^{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	2.12×10^{5} y		0.292 (100%)	
	^{237}Np	2.14×10^{6} y	4.78 (75%) 4.65 (12%)		
	^{238}Pu	86.4 y	5.50 (72%) 5.46 (28%)		
	^{239}Pu	2.44×10^{4} y	5.16 (88%) 5.11 (11%)		
	^{240}Pu	6.6×10^{3} y	5.17 (76%) 5.12 (24%)		
	^{241}Pu	13.2 y		0.021 (100%)	
	^{232}U	72 y	5.26 (31%) 5.32 (69%)		
	^{236}U	2.34×10^{7} y	4.47 (24%)		

(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. Figures 2-1 and 2-2 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.

Figure 2-1. Uranium Series Decay Chain

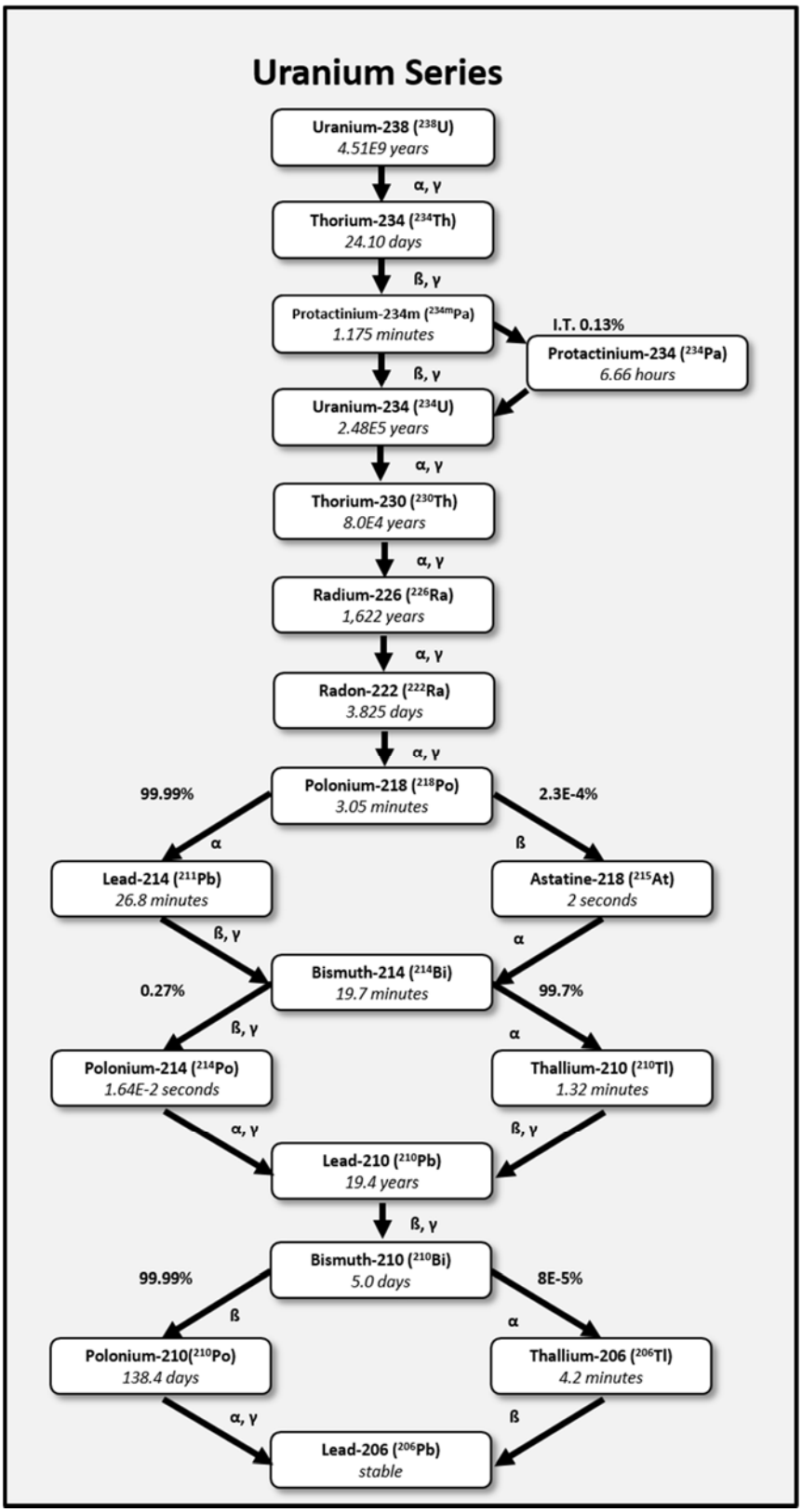
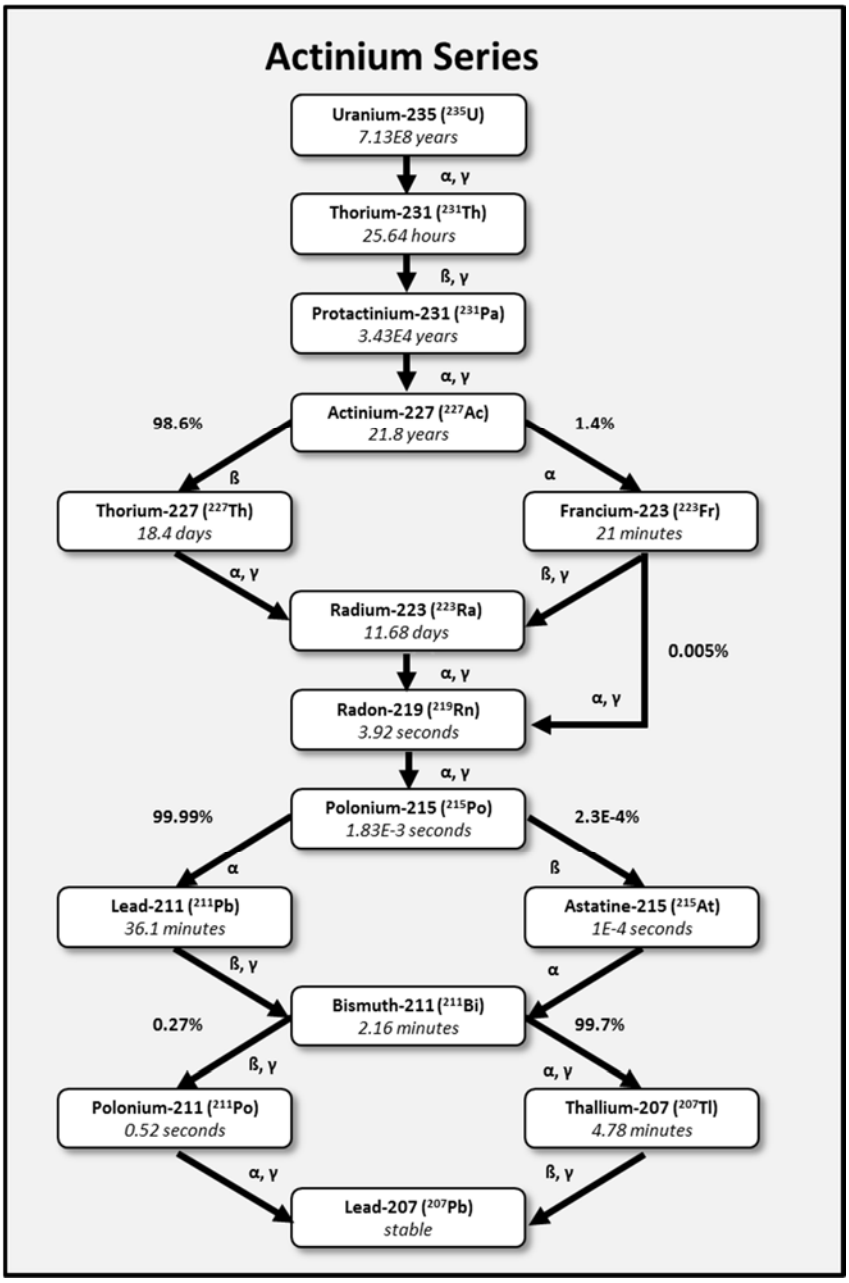


Figure 2-2. Actinium Series Decay Chain



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 2-3.

Table 2-3. Uranium Specific Activities

Type	Wt. % ^{235}U	Specific Activity of Mixture (Ci/g)
Natural	0.71	7×10^{-7}
Depleted	0.2	4×10^{-7}
Enriched	2	1×10^{-6}
Highly Enriched	20	9×10^{-6}

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

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

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.

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.

$$Specific\ Activity\ (SA) = [0.4 + 0.38E + 0.0034E^2]\ x\ 10^{-6}\ Ci/g$$

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

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

The specific activity of the resulting mixture is

$$SA_{mix} = \left[\frac{9.36 + 1.17}{2}\right] x\ 10^{-6}\ Ci/g = 5\ x\ 10^{-6}\ Ci/g$$

As a historical note, some of the earlier documentation refers to the "special curie" of natural uranium, which was defined as 3.7 x 10^{10} d/s of ^{234}U, 3.7 x 10^{10} d/s of ^{238}U, and 1.7 x 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 Derived Air Concentration (DAC) values for several radionuclides are shown in Table 2-4. These include the three uranium isotopes and selected contaminants in recycled uranium.

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 annual limit on intakes (ALI) 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 2-4. DACs for Uranium and Selected Contaminants in Recycled Uranium

	Inhalation DAC, From 10 CFR Part 835, Appendix A	
Nuclide	**Type F µCi/mL (Bq/m³)[(a)]**	**Type M µCi/mL (Bq/m³) [(a)]**
^{238}U	5×10^{-10} (2×10^{1})	3×10^{-10} (1×10^{1})
^{235}U	5×10^{-10} (1×10^{1})	3×10^{-10} (1×10^{1})
^{234}U	5×10^{-10} (1×10^{1})	2×10^{-10} (9×10^{0})
^{234}Th	Not Listed	1×10^{-7} (3×10^{3})
^{231}Th	Not Listed	1×10^{-6} (5×10^{4})
^{99}Tc	1×10^{-6} (5×10^{4})	1×10^{-7} (6×10^{3})
^{237}Np	Not Listed	8×10^{-12} (3×10^{-1})
^{238}Pu	Not Listed	6×10^{-12} (2×10^{-1})
^{239}Pu	Not Listed	5×10^{-12} (2×10^{-1})
^{240}Pu	Not Listed	5×10^{-12} (2×10^{-1})
^{241}Pu	Not Listed	2×10^{-10} (1×10^{1})
^{236}U	5×10^{-10} (1×10^{1})	2×10^{-10} (1×10^{1})

(a) See last paragraph of Section 2.5 for discussion of Type F, M, and S.

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 (M) = 3×10^{-10} µCi/mL (use DAC for ^{238}U)

^{239}Pu derived air concentration (M) = 5×10^{-12} µCi/mL

$$\frac{1}{DAC(U-Nat)} = \frac{1}{3\ x\ 10^{-10}} = 3\ x\ 10^{9}$$

$$\frac{0.001}{DAC\left(^{239}Pu\right)} = \frac{0.001}{5\ x\ 10^{-12}} = 2\ x\ 10^{8}$$

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

$$Fraction\ of\ total\ hazard = \frac{2x10^{8}}{(2x10^{8}) + (3x10^{9})} = 0.06$$

Therefore, ^{239}Pu at 0.1% of the U-Nat activity represents 6% of the potential inhalation dose.

The activity of 1 gram of U-Nat = 2.5 x 10^4 dps

Therefore, the ^{239}Pu activity in the 1 gram of U-Nat.= 0.001 x 2.5 x 10^4 = 25 dps

The specific activity of ^{239}Pu is 2.27 dps/nanogram:

$$25\ dps/g\ U\ x\ \frac{1\ nanogram\ Pu}{1\ dps/nanogram} = 11\ n\ anograms\ Pu/g\ U$$

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 is 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, heavy neoprene gloves, and safety glasses). 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 doses to the skin from 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 health 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 alloy-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 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 chips and turnings are pyrophoric and tend to catch fire.

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 Types M and S material (See last paragraph of Section 2.5 for discussion of Type F, M and S), 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 and lens of the eye dose resulting from external radiation exposures, due mainly to the 2.29 MeV (E_{max}) beta from ^{234m}Pa.The surface dose rates shown in Table 2-5 result primarily from beta radiation from decay products. The dose rates decrease quickly with distance because of the attenuation of the beta radiation and the small yield of the gamma radiation.

Table 2-5. Beta Surface Dose Rates from Equilibrium Thickness of Uranium Metal and Compounds[(a)]

Source	Beta Surface Dose Rate (mrad/hr)
U-Nat metal slab[(b)]	233
UO_2	207
UF_4	179
$UO_2(NO_3)_26H_20$	111
UO_3	204
U_3O_8	203
UO_2F_2	176
$Na_2U_2O_7$	167

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

(b) freshly polished, no oxide

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. 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.

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 International Commission on Radiological Protection (ICRP) Publication 66 (1994a) 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 exists for accidental ingestion of uranium. Particles removed from the respiratory tract by ciliary action are transferred to the gastrointestinal tract. 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 (1966). This model calculates material transferred from the GI tract to the blood based on solubility classes (ICRP, 1979; IAEA, 1994) or based on a single value for all compounds, as described in ICRP Publication 69 (1995a).

Distribution of uranium transferred into the bloodstream is calculated using a once-through metabolic model. ICRP Publication 30 (1979) also provides values for this distribution and excretion to calculate committed doses and long-term tissue retention. Recent models (Wrenn et al., 1994; ICRP, 1995a) 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. Epidemiological and toxicological animal studies of oral exposure to relatively soluble naturally occurring isotopic mixtures appear demonstrate that the health effects are chemical toxicity to tissues; while those from inhalation exposure may include a slight radiological component, especially if the exposure is chronic (ATSDR, 2013). For inhalation of relatively insoluble uranium compounds, the radiological hazard can be the most limiting due to an increased residence time in the lungs and low fractional absorption to blood (Leggett, et. al., 2012).

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 (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.

Data on human exposures and the effect of various intakes of uranium are summarized in Table 5-7. 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 2-6. 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 N13.22-2013); 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 Publication 68 (1994b).
(d) See discussion in (Just and Elmer, 1984)

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 DOE regulated facilities, the more conservative of the two standards (OSHA or ACGIH) must be used unless enrichment and solubility dictate more stringent controls based on radiological concerns. Table 2-6 lists airborne concentration limits for transportable uranium that have been published by various organizations.

Table 2-7. 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, 1992b)
ACGIH[(a)]	0.2	Threshold Limit Values and Biological Exposure Indices for 2005, American Conference of Governmental Industrial Hygienists (ACGIH, 2005)
OSHA[(b)]	0.05 (soluble) 0.25 (insoluble)	29 CFR § 1910.1000
NIOSH	0.05	National Institute for Occupational Safety and Health (NIOSH)

(a) ACGIH also has a short term exposure limit of 0.6 mg/m^3. This is based on a 15 minute time weighted average exposure.

(b) Preferred/recommended limit.

Past limits for single acute inhalation intakes have been set by the ICRP in its Publication 6, (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 µg/m^3. Lawrence (1984) derived acute inhalation intake limits of 15 and 80 mg for Type F and Type M 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 Part 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 68 (1994b) metabolic model for Type F 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 Part 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

a. Lack of data on the effects of combined exposures to UO_2F_2 and HF.

b. Lack of detailed information on effects of short-term exposures to soluble/transportable

uranium in the range from 100-1000 mg/m^3.

c. 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 kidney burdens (in µg U/g Kidney) listed in Table 2-7.

Table 2-8. Uranium Levels for Various Effects

Effect	Kidney Burden, µg U/g Kidney	Total Kidney Burden, mg U	Intake, mg
No effect	1.1	0.337	6.5
Maximal Nonlethal	2.2	0.71	13
LD_{50}	54.8	16.79	322

The kidney burden values can be used to derive an intake based on the ICRP Publication 78 (1997), model for uranium metabolism (62% of inhaled Type F uranium is taken up into the bloodstream (82% deposited – 62% systemic and 20% to GI tract) and 8.5% of that goes to the kidneys). For example, the "no effect" value in Table 2.7 corresponds to an intake of (1.1µg U/g)(310 g/kidney)(1000 µg/mg)/((0.62 fraction systemic)(.085 fraction systemic going to kidney) = 6.5 mg.

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.

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 (0.62)
f_k = bloodstream fraction entering kidneys (0.085) and

$R = \lambda K_b$ where

where

R = kidney removal rate (mg/day)
λ= 0.099 (day^{-1}) (ICRP, 1997)
K_b = amount in the kidney (mg)

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:

$R = (0.337) \times 0.099 = 0.033$ mg/day

$K = B_r\ (m^3/day) \times C_a (mg/m^3) \times (0.62) \times (0.085)$

$K = R = 0.033$ mg/day

$B_r \times C_a \times (0.62) \times (0.085) = 0.033$ mg/day

$B_r \times C_a = 0.63$ mg/day

Standard man breathes 9.6 m^3 of air in an 8-hour day, so the resulting concentration limit is 0.63/9.6 = 0.066 mg/m^3. This is 30% 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., Type F) 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 F) 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, 1992a).

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. ICRP Publication 88 (2001) gives dose coefficients for the embryo, fetus and newborn of females following intake of selected radionuclides. It also provides a review of biokinetic and dosimetric models for calculating doses to the offspring of mothers following intakes by the mother before or during pregnancy.

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 2-4 provides 10 CFR Part 835 derived air concentration values for selected radionuclides. Table 2-8 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.

Table 2-9. 2005 ACGIH Threshold Limit Values (TLVs) for Selected Metals

	TLV	
Metal	**TLV-TWA, mg/m^3**	**TLV-STEL, mg/m^3**
Uranium	0.2	0.6
Beryllium	0.002	0.01
Lead	0.05	--
Mercury, elemental and inorganic forms	0.025	--
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. The 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 60 (1991a) classifies all materials into three material types - F, M, and S. Type F is most transportable (pulmonary removal half-time of days), Type S the least transportable (removal half-time of years), and Type M 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 2-9 lists several common uranium compounds and their assigned material types.

Table 2-10. Inhalation Classification for Some Uranium Compounds

Uranium Compound	Chemical Name	Material Type
Uranium hexafluoride	UF_6	Type "F"
Uranyl fluoride	UO_2F_2	Type "F"
Uranyl nitrate	$UO_2(NO_3)_2$	Type "F"
Uranyl acetate	$UO_2(C_2H_3O_2)_2$	Type "F"
Uranyl chloride	UO_2Cl_2	Type "F"
Uranyl sulfate	UO_2SO_4	Type "F"
Uranium trioxide	UO_3	Type "M"
Uranium tetrafluoride	UF_4	Type "M"
Uranium oxide	U_3O_8	Type "S"(b)
Uranium dioxide	UO_2	Type "S"(b)
Ammonium diuranate	$(NH_4)_2 + U_2O_7$	Type "M" (a)
Uranium aluminide	UAl_x	Type "S"
Uranium carbide	UC_2	Type "S"
Uranium-zirconium alloy	UZr	Type "S"
High-fired uranium dioxide	UO_2	Type "S"(b)

(a) Ammonium diuranate is known to contain uranium as UO3, and should not be assigned to a single inhalation class.

(b) The solubility of uranium oxides is very dependent on heat treatment. The rate of oxidation may also affect the solubility. 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 chemical 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 receive 100 millirem committed effective dose (CED) from all internal and external occupational doses received in a years. Therefore, it is prudent to calculate organ doses and CED 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 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). For comparison, the more conservative of the two chemical toxicity standards (OSHA or ACGIH) is used to compare the chemical hazard with the radiological hazard. The ICRP Publication 68 (1994b) dose coefficients and ICRP 78 (1997) models are used in the following examples to determine the relative hazards for acute and chronic exposure situations. Table 2-10 summarizes the dose coefficients used in the following examples.

Table 2-11. Dose Coefficients for Determining Enrichments above which Radiological Hazards Become Limiting

	Effective Dose Coeff, Sv/Bq	Effective Dose Coeff, Sv/Bq	Effective Dose Coeff, Sv/Bq	Organ Dose Coeff, Sv/Bq	Organ	Organ Dose Coeff, Sv/Bq	Organ	Organ Dose Coeff, Sv/Bq	Organ
	F	M	S	F		M		S	
U-234	6.40E-07	2.10E-06	6.80E-06	1.10E-05	BS	1.60E-05	LG	7.50E-05	ET
U-235	6.00E-07	1.80E-06	6.10E-06	1.10E-05	BS	1.40E-05	LG	6.90E-05	ET
U-238	5.80E-07	1.60E-06	5.70E-06	1.00E-05	BS	1.30E-05	LG	6.50E-05	ET
ET – Extrathoracic airways				LG – Lung			BS – Bone Surface		

To determine which hazard is limiting for a chronic exposure, the chemical toxicity air concentration limit (0.2 mg/m^3 for material type “S” and 0.05 mg/m^3 for material types “F” and “M”) and a calculated derived air concentration (DAC), based on percent enrichment, are used. For material type “F” the DAC is based on 50 rems to the bone surface, for material type “M” the DAC is based on 5 rems effective dose, and for material type “S” the DAC is based on 50 rems to extrathoracic portion of the respiratory tract. The dose coefficients from Table 2-10 are used to calculate a DAC for varying enrichments. The enrichment at which the DAC is equal to the chemical toxicity limit forms the "dividing line" between chemical and radiological limits as the limiting hazard. Generally, exposures to higher enrichments are limited by radiological limits; exposures to lower enrichments by chemical limits. Example 3a provides the methodology for determining the "dividing line" enrichment for the chronic exposure scenario. The same calculation is done using the 100 millirem monitoring threshold.

For an acute exposure scenario, the amount of an intake which would result in exceeding the fifteen minute time weighted chemical toxicity short term exposure limit for material types “F”, “M”, and “S” uranium (i.e., 0.6 mg/m^3 x 0.3 m^3 air intake per fifteen minute) is

determined. This amount is compared to the amount of an intake which would result in exceeding the radiological limit. Again, for material type "F" the radiological limit is based on 50 rems to the bone surface, for material type "M" the radiological limit is based on 5 rems to the whole body, and for material type "S" the radiological limit is based on 50 rems to extrathoracic portion of the respiratory tract. Example 3b provides the methodology used for an acute exposure scenario. Table 2-11 summarizes the "dividing line" enrichments, above which radiological limits are controlling. The same calculation is done using the 100 millirem monitoring threshold.

Example 3a - General Solution, Chronic Exposure

Step 1. For increasing ^{235}U enrichments, and consequential ^{234}U enrichment, determine the activity fraction (AF) from ^{234}U, ^{235}U, and ^{238}U in the enriched uranium.

$$AF_{234} = \frac{Enrichment_{234} * SA_{234}}{Enrichment_{234} * SA_{234} + Enrichment_{235} * SA_{235} + Enrichment_{238} * SA_{238}}$$

$$AF_{235} = \frac{Enrichment_{235} * SA_{235}}{Enrichment_{234} * SA_{234} + Enrichment_{235} * SA_{235} + Enrichment_{238} * SA_{238}}$$

$$AF_{238} = \frac{Enrichment_{238} * SA_{238}}{Enrichment_{234} * SA_{234} + Enrichment_{235} * SA_{235} + Enrichment_{238} * SA_{238}}$$

Where:
Enrichment is the percent by weight.
SA_{234} = specific activity of ^{234}U = 2.30E+08 Bq/g
SA_{235} = specific activity of ^{235}U = 79312 Bq/g
SA_{238} = specific activity of ^{238}U = 12329 Bq/g

$$Enrichment_{234} \cong \frac{(0.0055\ natural\ fraction\ ^{234}U)}{(0.72\ natural\ fraction\ ^{235}U} * {}^{235}U\ enrichment * 1.2^{(a)}$$

(a) Gaseous diffusion enrichment of ^{235}U results a proportionally greater enrichment of ^{234}U. The 1.2 factor takes into account the gaseous diffusion process causing a greater increase in ^{234}U than in ^{235}U (See Section 2.1.3).

$$Enrichment_{238} = 100 - Enrichment_{234} - Enrichment_{235}$$

Step 2. Using the dose coefficients from Table 2-10 and the activity fractions, for increasing enrichments, determine activity enriched uranium organ and effective dose coefficients for material types "F", "M" and "S".

$$\%\ Enrichment\ Dose\ Coeff = AF_{234} * dose\ coeff_{234} + AF_{235} * dose\ coeff_{235} + AF_{238} * dose\ coeff_{238}$$

For example, for 5% enriched uranium (i.e., 5%, by weight, of the uranium is ^{235}U), material type "S" dose to the extrathoracic portion of the respiratory tract:

Using the equations from Step 1, it is determined that, for 5% enriched uranium, 87% of the activity is from ^{234}U, 3% from ^{235}U, and 10% from ^{238}U.

From this we can determine a dose coefficient for 5% enriched uranium using the above equation:

$$5\%\ Enrichment\ Dose\ Coeff = 0.87 * 7.5E-5\ Sv/Bq + 0.03 * 6.9E-05\ Sv/Bq + 0.1 * 6.5E-5\ Sv/Bq = 7.38E-5\ Sv/Bq$$

Step 3. The ALI is determined, for different enrichments using the % enrichment dose coefficient determined in Step 2. For material type "F" the % enrichment ALI is based on 50 rems to the bone surface, for material type "M" it is based on 5 rems effective dose, and for material type "S" on 50 rems to extrathoracic portion of the respiratory tract.

$$\%\ Enrichment\ ALI = \frac{Dose\ Limit}{\%\ Enrichment\ dose\ coeff}$$

The % enrichment ALI, which is units of Bq, is converted to a mass using the SA % enrichment formula and is divided by the volume of air a worker breathes in a work year (2000 hours), 2400 m^3, to give a mass airborne concentration value.

$$SA\ \%\ Enrichment = [(0.4 + 0.38(enrichment) + 0.0034(enrichment)^2] * 1.0E-6;$$

Where enrichment is the fraction by weight of ^{235}U, expressed as a percentage.

Note that the value of *SA_% Enriched* as calculated by the formula is in units of µCi/g (Equation from 10 CFR Part 20 Appendix B, footnote 3)

$$Air\ Concentration\ (g/m^3) = \frac{(\%\ Enrichment\ ALI)}{SA\ \%Enrichment * Air\ Volume\ Intake}$$

The mass airborne concentration value is used for comparison with the chemical toxicity limit.

Continuing the above example: the ALI for 5% enriched uranium, material type "S" dose to the extrathoracic portion of the respiratory tract:

$$\%\ Enrichment\ ALI = \frac{0.5\ Sv}{7.38E-5\ Sv/Bq} * \frac{1\ \mu Ci}{37000\ Bq} = 0.18\ \mu Ci$$

$$SA\ 5\%\ Enrichment = (0.4 + 0.38 * 5 + 0.0034 * 5^2) * 1.0E-6 = 2.38E-6\ \mu Ci/g$$

$$Air\ Concentration\ = \frac{0.18\ \mu Ci}{2.38E-6\ \mu Ci/g * 2400\ m^3} * \frac{1000\ mg}{1\ g} = 0.032\ mg/m^3$$

Step 4. This value is compared to the chronic chemical toxicity air concentration limit (0.2 mg/m^3 (ACGIH TLV) for material type "S" and 0.05 mg/m^3 (OSHA soluble PEL) for material types "F" and "M". As the enrichment increases the air concentration values determined for radiological control decrease. The enrichment where they fall below the chemical toxicity limits becomes the "dividing line" where radiological limits become more restrictive.

In the above example the air concentration value, 0.032 mg/m^3, is lower than the chemical toxicity value and is therefore limiting (for type "S" material, chronic exposure, radiological considerations are limiting for all enrichments).

Step 5. The above process is repeated using the 100 millirem monitoring threshold in lieu of the annual dose limit. For these calculations the effective dose coefficients are used for all isotopes.

Example 3b - General Solution, Acute Exposure

Steps 1 and 2: Same as in example 3a.

Step 3. This is similar to Step 3 in example 3a, with the exception that % enrichment ALI is not divided by the volume of air a worker breathes in a work year, 2400 m^3.

The annual limit on intake (ALI) is determined, for different enrichments using the % enrichment dose coefficient determined in Step 2. For material type "F" the % enrichment ALI is based on 50 rems to the bone surface, for material type "M", it is based on 5 rems to the whole body, and for material type "S", on 50 rems to extrathoracic portion of the respiratory tract. The resultant value, in Bq, is converted to a mass using the SA % enriched formula. The result is a mass intake limit.

$$\%\ Enrichment\ ALI = \frac{Dose\ Limit}{\%\ Enrichment\ dose\ coeff}$$

$$Mass\ Intake\ Limit = \frac{\%\ Enrichment\ ALI}{SA\ \%\ Enrichment}$$

From example 3a: the ALI for 5% enriched uranium, material type "S" dose to the extrathoracic portion of the respiratory tract:

$$\%\ Enrichment\ ALI = \frac{0.5\ Sv}{7.38E-5\ Sv/Bq} * \frac{1\ \mu Ci}{37000\ Bq} = 0.18\ \mu Ci$$

$$Mass\ Intake\ Limit = \frac{0.18\ \mu Ci}{2.38E-6\ \mu Ci/g} * \frac{1000\ mg}{1\ g} = 75.8\ mg$$

Step 4. This value is compared to the product of the volume breathed in fifteen minutes times the short term exposure limit for chemical toxicity (0.6 mg/m^3 (OSHA STEL) * 0.3 m^3

breathed per fifteen minutes = 0.16 mg for material types “F”, “M”, and “S”. As the enrichment increases the mass acute intake limit determined for radiological control decreases. The enrichment where they fall below the chemical toxicity limits becomes the “dividing line” where, for an acute exposure, radiological limits become more restrictive.

In the above example the mass acute intake limit, 75.8 mg, is higher than the chemical toxicity value and therefore the chemical toxicity limit is controlling for this enrichment (for acute exposures, compared to the radiological limit, chemical toxicity considerations are always limiting).

Step 5. The above process is repeated using the 100 millirem monitoring threshold in lieu of the annual dose limit. For these calculations the effective dose coefficients are used for all isotopes.

Table 2-12. Impact of Monitoring at 100 Millirem Enrichments above which Radiological Limits Predominate (Calculations Not Shown)

	Acute		Chronic	
Material Type	Using 100% of Radiological Limit	Using 2% of Radiological Limit	Using 100% of Radiological Limit	Using 2% of Radiological Limit
F	(a)	(a)	21.8%	(b)
M	(a)	99.6%	12.0%	(b)
S	(a)	41.8%	(b)	(b)

(a) Chemical toxicity limits are limiting at all enrichments.
(b) Radiological limits are limiting at all enrichments.

As shown in Table 2-11, for chronic exposures, the 100 millirem monitoring threshold (i.e., 2% of radiological limit) results in radiological conditions, requiring evaluation of the need to monitor radiological intakes, as a controlling factor. For chronic exposures the radiological dose limits are controlling for type “S” material for all enrichments, and for lower enrichments for types “M” and “F” material. For acute exposure situations, the radiological monitoring threshold is a controlling factor for higher enrichments for types “M” and “S” material. For all other acute exposure situations, the chemical toxicity limits, which are based on a fifteen minute or less exposure, are more controlling than the radiological limits, which are based on an annual exposure.

2.6 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 5-8.

Table 2-13. 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 Publication 23 (1975)		

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.

2.7 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, e.g., back injuries from lifting, dropping heavy parts on feet. 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.

2.7.1 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.

2.7.2 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 3 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).

2.7.3 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) 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).

2.7.4 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-1081-2014, *Primer on Spontaneous Heating and Pyrophoricity* (2014b), contains additional guidance.

3 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 Part 835. DOE has provided supporting and clarifying guidance in the DOE G 441.1-1C, DOE-STD-1098-2008 (2009c), and DOE Radiological Control Technical Positions. Other related source documents include publications of the EPA, NRC, 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 441.1-1C (2008a) suggests the following, based on the content of 10 CFR Part 835:

a. Organization and Administration
b. ALARA Program
c. External Dosimetry Program
d. Internal Dosimetry Program
e. Area Monitoring and Control
f. Radiological Controls
g. Emergency Exposure Situations
h. Nuclear Accident Dosimetry
i. Records
j. Reports to Individuals
k. Radiation Safety Training
l. Limits for the Embryo/Fetus

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 G441.1-1C (2008a) and DOE-STD-1098-2008 (2009c) 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 Part 835 and shall be approved by DOE (10 CFR § 835.101). Any changes that decrease the effectiveness of the RPP shall be approved by DOE before implementation (10 CFR § 835.101).

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 than 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 Part 830, *Nuclear Safety Management* (2011f), DOE O 414.1D, *Quality Assurance* (2013e), and their associated guides. Specific guidance applicable to RPPs is provided in DOE G 441.1-1C (2008a).

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, which 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 formal plans

and measures for applying the as low as reasonably achievable (ALARA) process to occupational exposure (10 CFR § 835.101(c)).

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 Part 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 Part 835. Written procedures should be developed and employed under the following circumstances:

a. When worker health and safety are directly affected

b. When the expected outcome for the process or operation requires that a specific method be followed

c. When the process or operation is infrequently used and competence training cannot assure adequate implementation

d. 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. DOE G 441.1-1C provides 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 N13.6-2010, *Practice for Occupational Radiation Exposure Records Systems* (2010) 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.

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 DOE-STD-1098-2008 (2009c)). 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 DOE-STD-1098-2008 (2009c) and DOE-HDBK-1122-2009, *Radiological Control Technician Training Program* (2011c).

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 DOE-STD-

1098-2008 (2009c) 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 426.2, *Personnel Selection, Training, Qualification, and Certification Requirements for DOE Nuclear Facilities* (2013b). DOE-STD-1098-2008 (2009c) 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

a. DOE-HDBK-1130-2008, Radiological Worker (RW) Training

b. DOE-HDBK-1131-2007, General Employee Radiological Training

c. DOE-STD-1107-97, CN1, Knowledge Skills and Abilities for Key Radiation Protection Positions at DOE Facilities

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 Part 835 and DOE O 458.1 establish the policy of maintaining doses ALARA for workers and the public, respectively, resulting from radiation from DOE operations. 10 CFR § 835.101(c) requires that plans and programs implementing the ALARA process be prepared, and 10 CFR § 835.704(b) requires that records must be must be maintained to demonstrate the implementation of ALARA. DOE G 441.1-1C (2008a) and DOE-STD-1098-2008 (2009c) 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 HEPA exhaust fans, ductwork and filters, and special fixtures to hold highly radioactive materials requiring detailed inspection, repair, modification, or fabrication. Such devices can reduce radiation exposures by controlling contamination, 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 441.1-1C (2008a).

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 441.1-1C (2008a).

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 441.1-1C (2008a).

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 441.1-1C (2008a).

3.2.5.1 Radiological Surveys and Data Trending

Sections 835.401 - 835.403 of 10 CFR Part 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

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 as follows:

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

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, the

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)

d. Dose rates

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

f. Date and time the survey was performed

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 10 CFR § 835.701(b) which requires retention of records until final disposition is authorized by DOE.

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

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

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.

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 N317-1980, *Performance Criteria for Instrumentation Used for In-Plant Plutonium Monitoring* (1980). Performance specifications are also given in IEEE/ANSI N323AB-2013 (2014a), *Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments*; IEEE/ANSI N42.17A-2003, *Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions* (2004a); IEEE/ANSI N42.17C-1989, *Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Extreme Environmental Conditions* for portable radiological control instrumentation (1990b); and IEC Publication 60325:2002, *Radiation Protection Instrumentation – Alpha, beta and alpha/beta (beta energy >60 keV) contamination meters and monitors* (2002b), for alpha and beta contamination meters and monitors. Criteria for air monitoring instrumentation are provided in ANSI/HPS N13.1-2011, *Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities*

(2011a); IEC Publication 60761, *Equipment for Continuously Monitoring Radioactivity in Gaseous Effluents* (2002a)*;* and ANSI N42.17B-1989, *Performance Specifications for Health Physics Instrumentation - Occupational Airborne Radioactivity Monitoring Instrumentation* (1990a). Criticality alarm systems are discussed in ANSI/ANS 8.3-1997, *Criticality Accident Alarm System (1997)*. The criteria discussed in the following sections are specified in these standards as referenced.

Portable Monitoring Instruments

ANSI N317 (1980) 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 (2004a) 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 (1980) 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 criteria 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 criteria stated in ANSI N317 (1980)

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 (2004a) has a broader scope than ANSI N317 (1980), 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 60325 (2002b) 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 (1980). 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. (1988) provides guidance on each of these functions.

ANSI N42.17B (1990a) provides additional performance criteria for air monitors used to detect uranium. This standard provides specifications for general criteria (e.g., sampler design, units of readout, alarm threshold), 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 (1980); 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 (2011a) 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 (1980) 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 N323AB (2014a) 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 Part 835.1304 and DOE Order 420.1C, *Facility Safety* (2015a).

Effluent Monitors. Facilities should evaluate potential emissions in accordance with ANSI/HPS N13.1 (2011a) 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 1000 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* (1979) 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 shall 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 441.1-1C (2008a) and ANSI N323AB (2014a) 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 6980-1, *Nuclear Energy – Reference Beta Particle Radiation* (2006), which 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/ANS, 1997). 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 Part 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), 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. DOE-STD-1098-2008 (2009c) 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 Part 835 exist (10 CFR §835.601- § 835.606). DOE Guide G-441.1-1C (2008a) and DOE-STD-1098-2008 (2009c) 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 controlled in accordance with 10 CFR § 835.501. The degree of control shall be commensurate with existing and potential radiological hazards within the area.

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 Part 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 Technical Standard.

3.2.8 Nuclear Accident Dosimetry

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

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 Part 835 establishes radiation protection program records requirements. Section 835.701(b) states that, unless otherwise specified, records shall be retained until final

disposition is authorized by DOE. An acceptable program for generating and administering occupational radiation protection program records and reports to maintain compliance with the provisions of 10 CFR §835.702, § 835.703, and § 835.704 are described in chapter 13 of DOE G 441.1-1C (2008a).

Most of the required radiological records have established retention periods. The retention periods are discussed in DOE O 243.1B, *Records Management Program*. Individual records may be covered by the Privacy Act; DOE has codified its Privacy Act regulations in 10 CFR Part 1008, *Records Maintained on Individuals (Privacy Act)*.

3.2.10 Radiation Safety Training

A thorough radiation safety training program shall 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 shall be documented and records maintained to demonstrate compliance with 10 CFR § 901. DOE G 441.1-1C (2008a) and DOE-STD-1098-2008 (2009c) provide guidance on information to be presented during the training programs.

DOE requires biennial radiation safety training for general employees and radiological workers and when there is a significant change to radiation protection procedures. 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 or either Rad Worker I or Rad Worker II Training.

The level of radiation worker training should be determined in accordance with the Table 3-1 of DOE-STD-1098-2008. All training shall be consistent with 10 CFR § 835.901 and should be consistent with the guidance provided in DOE G 441.1-1C (2008a). All training dispositions and records shall 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. DOE has developed General Employee Radiological Training for this purpose.

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

d. Radioactive Material Areas

Guidance for training of members of the public is provided in DOE G 441.1-1C (2008a) and DOE-STD-1098-2008 (2009c). 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. In accordance with 10 CFR § 835.208, the total effective dose limit for members of the public exposed to radiation and/or radioactive material during access to a controlled area is 0.1 rem in a years.

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:

a. Evidence of completing required training, as applicable

b. Visitor radiation exposure disclosure

c. 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.

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 460.1C, *Packaging and Transportation Safety* (2010).

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

i. DOE field office, if the package is to be used for Highway Route Controlled Quantity inter-area shipments

3.3.2 Approval for Editorial Changes

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

3.3.2.1 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.

3.3.3 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 422.1, *Conduct of Operations* (2014a).

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 must have a written policy on radiation protection, including an ALARA policy. All radiation protection procedures and controls must have recognizable or formal technical bases for limits, methods, and personnel protection standards. Procedures must 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, which is authorized by DOE in accordance with 10 CFR § 835.701(b). ANSI N13.6-2010 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.

3.3.3.1 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 422.1 (2014a) apply. Procedures are a key factor affecting radiation protection performance. 10 CFR Part 835.104 requires written procedures. 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.

3.3.3.2 Posting and Labeling

The requirements for area posting and radioactive material labeling are established in 10 CFR Part 835, Subpart G. Guidance on implementing the regulatory requirements can be found in DOE G 441.1-1C (2008a) and DOE-STD-1098-2008 (2009c). 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 422.1 (2014a).

3.3.3.3 Instrument Calibration

The status of installed and portable radiological instruments should be well known and appropriate to the use. All instruments and equipment used for monitoring shall meet the requirements of 10 CFR § 835.401.

"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 including source checks are routinely made to verify operability, 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.

3.3.3.4 Audits

Conduct of operations does not, in itself, address requirements for auditing. 10 CFR Part 835.102 requires an internal audit of the radiation protection program no less frequently than 36 months. Guidance states 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 422.1 (2014a) 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.

3.3.3.5 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.

3.3.4 Integrated Safety Management

The radiological control program must be developed and implemented in a manner that is consistent with the DOE approved Radiation Protection Program required by 10 CFR § 835.101

and should be consistent with the provisions of DOE Policy P 450.4A, *Integrated Safety Management Policy* (2011b), 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 DOE-STD-1098-2008 (2009c) and this Technical Standard as a guide to integrating radiological control measures into work planning and execution.

11 APPENDIX A - REFERENCES

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12 APPENDIX B - GLOSSARY

Terms used consistent with their regulatory definitions.

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.

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.

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.

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 Chapter of DOE G 441.1-1C)

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 Chapter of DOE G 441.1-1C)

decontamination: The process of removing radioactive contamination and materials from personnel, equipment, or areas.

Department of Energy operations: Those activities for which DOE has 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.

detector: A device or component designed to produce a quantifiable response to ionizing radiation, normally measured electronically. (Portable Monitoring Instrument Calibration Chapter of DOE G 441.1-1C)

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-1111-2013 (2013a). (External Dosimetry Program Chapter of DOE G 441.1-1C)

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 Chapter of DOE G 441.1-1C)

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. (DOE-STD-1098)

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-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. (DOE-STD-1098).

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 Chapter of DOE G 441.1-1C)

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 Chapter of DOE G 441.1-1C)

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 Chapter of DOE G 441.1-1C)

portable air sampler: An air sampler designed to be moved from area to area.

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 Chapter of DOE G 441.1-1C)

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 Part 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. (DOE-STD-1098)

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. (DOE-STD-1098)

radiological control organization: An organization responsible for radiation protection. (Sealed Radioactive Source Accountability and Control Chapter of DOE G 441.1-1C)

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. (DOE-STD-1098)

removable contamination: Radioactive material that can be removed from surfaces by nondestructive means, such as casual contact, wiping, brushing, or washing. (DOE-STD-1098)

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. (DOE-STD-1098)

workplace monitoring: The measurement of radioactive material and/or direct radiation levels in areas that could be routinely occupied by workers.