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HEALTH PHYSICS MANUAL OF GOOD PRACTICES FOR URANIUM FACILITIES - (USED AS A REFERENCE IN OU 3 RI/FS/PP APPENDIX A)

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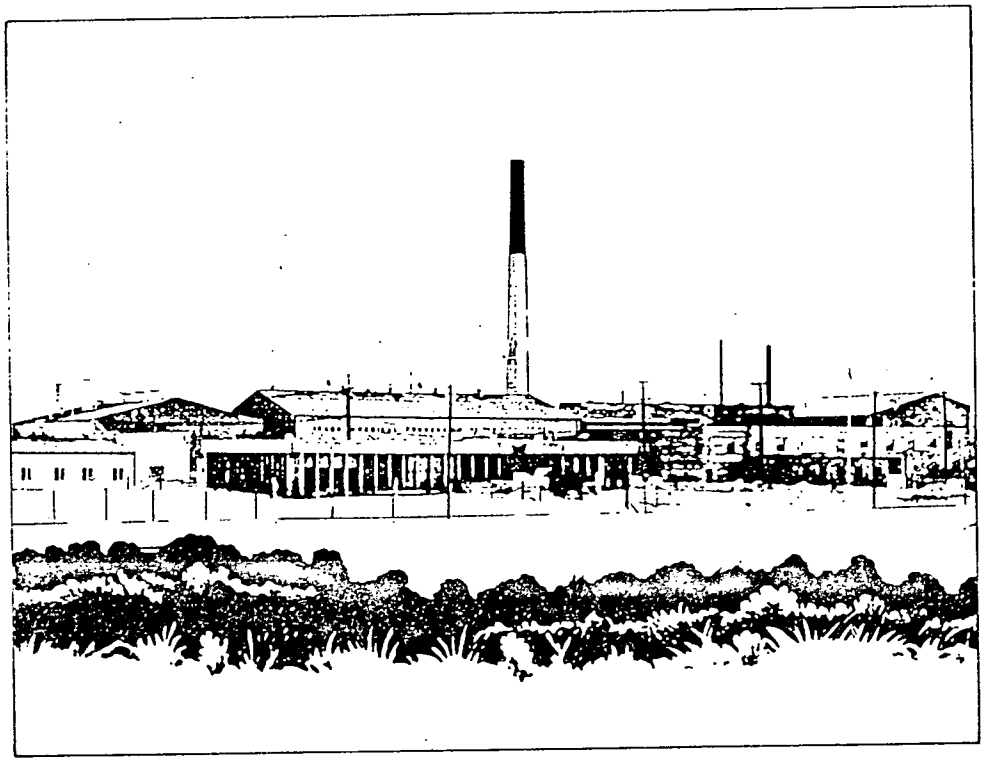
Health Physics Manual of Good Practices for Uranium Facilities



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DOE Project Manager

EG&G Idaho

June 1988

Prepared for the U.S. Department of Energy
Assistant Secretary for Environment, Safety, and Health
Under Contract DE-AC07-76IDO1570

Idaho National Engineering Laboratory

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FOREWORD

Historically from the health perspective, uranium has been considered more of a heavy metals than a radiological hazard. The phrase "its only uranium" was heard frequently throughout the nuclear industry. This early perspective resulted primarily from the relatively low specific activity of uranium and the fact that while kidney damage of a chemical origin was observed in laboratory animals administered uranium compounds, radiological toxicity at similar levels had not been demonstrated. However, the revisions to biokenetic models for the evaluation of radiation exposure as contained in ICRP Publication 26 and 30 have resulted in a decrease in the annual limit of intake for uranium. This, coupled with recent changes in our understanding of the biological behavior and health significance of uranium along with technology advances for controlling, monitoring, and evaluating potential exposures of workers make it apparent that greater attention can and must be placed on controlling uranium contamination in the workplace.

The "Health Physics Manual of Good Practices for Uranium Facilities" should prove to be extremely useful in providing information on design and implementation of radiation protection programs consistent with current standards and state-of-the-art technology. It is expected that this manual will serve as a guide in the evaluation of needed upgrade programs for older facilities as well as in the development of radiation protection programs for newer facilities.

The working group responsible for the development of this document was comprised of technical experts with extensive applied health physics experience. It was peer reviewed by DOE and DOE contractor personnel as well as experts in other sectors of the nuclear industry. We want to express our appreciation to both the working groups and the individuals who reviewed the document. This dedicated effort and the application of this Guide will indeed result in a significant contribution to the radiation protection of workers in the uranium industry.

Edward J. Vallario
Acting Director
Radiological Controls Division
Office of Nuclear Safety

ACRONYMS

ACGIH	American Conference of Governmental Industrial Hygienists
ALARA	as low as reasonably achievable
ALI	annual limit on intake
AMAD	activity median aerodynamic diameter
ANSI	American National Standards Institute
AVLIS	atomic vapor laser isotope separation
AWWA	American Water Works Association
BZ	breathing zone sampler
CAM	continuous air monitor
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
CPVC	chlorine polyvinyl chloride
DAC	derived air concentration
DBA	design-basis accident
DBE	design-basis earthquake
DBF	design-basis fire
DE	dose equivalent
DOE	U.S. Department of Energy
DOD	Department of Defense
DOT	Department of Transportation
DTPA	diethylenetriaminepenta-acetic acid
EA	environmental assessment
EDTA	ethylene diamine tetracetic acid
EIS	environmental impact statement
ECS	Emergency Control Station
EPA	U.S. Environmental Protection Agency
EPZ	emergency planning zone
ERDA	Energy, Research, and Development Administration
FMPC	feed material production center
GAS	general area sampler
GDP	Gaseous Diffusion Plant
GI	gastro intestinal

GM	Geiger-Mueller
GSD	geometric standard deviation
HEPA	high-efficiency particulate air
HPGE	hyperpure germanium
HQ	headquarters
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IEC	International Electrotechnical Commission
LLNL	Lawrence Livermore National Laboratory
MPC	maximum permissible concentration
MSHA	Mining Safety Health Administration
NAD	nuclear accident dosimeter/dosimetry
NAWAS	National Warning System
NCRP	National Council on Radiation Protection and Measurements
NCS	nuclear criticality safety
NEPA	National Environmental Policy Act
NFPA	National Fire Protection Association
NIOSH	National Institute of Occupational Safety and Health
NOAA	National Oceanic and Atmospheric Administration
NRC	U.S. Nuclear Regulatory Commission
NRRPT	National Registry of Radiation Protection Technologists
NUREG	Nuclear Regulatory Report
OBE	operating basis earthquake
OSHA	Occupational Safety and Health Administration
PAS	personal air sampler
PC	protective clothing
PMF	probable maximum flood
PSAR	preliminary safety analysis report
PRR	protective response recommendation
QA	quality assurance
RCG	radioactivity concentration guide
RCRA	Resource Conservation and Recovery Act
RU	recycled uranium
SDD	system design description
STEL	short term exposure limit

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TE	tissue equivalent
TL	thermoluminescent
TLD	thermoluminescent dosimeter
TLU	threshold limit values
UNH	uranyl nitrate
USGS	United States Geologic Service
VHE	very highly enriched
WBC	whole body count

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SECTION 1

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SECTION 1

INTRODUCTION

Uranium is a radioactive, heavy metal which is important to the Nuclear Industry because one of the isotopes (U-235) has a high fission cross section and is the fuel for nuclear reactors. Hence, uranium is the material at the "head end" of the nuclear fuel cycle.

As will be discussed in this manual, the half life of most of the naturally occurring isotopes of uranium are very long--so long in fact that chemical toxicity can predominate over the radiological hazard. It is well known that it requires significant mass quantities of uranium 238 and 235 to create a recognizable hazard. Personnel and radiation workers have historically become somewhat complacent in dealing with uranium: "It's only uranium" is a common problem attitude.

Periodically problems and/or exposures above the current acceptable levels direct attention to the practices and operations which deal with uranium. The current science and technology within the discipline of radiological safety have been reviewed and combined in a Health Physics Manual of Good Practice for Uranium Facilities in Department of Energy (DOE) Uranium facilities with the purpose of providing a balanced perspective of risk and control needs.

1.1 Purpose and Objectives

Several basic purposes and/or objectives guide the presentation of the material in this manual:

1. The overriding objective of this manual is to assist in assuring that all DOE facilities have the elements of a comprehensive and sound personnel radiological protection program.

2. The primary focus of this manual is the applied health physicist. The objective is to provide useful information and definitive guidance to the professional health physicist in the field.

3. Consistency within the DOE program from facility to facility is a major objective, even though the types of operations and applications may be significantly different.

4. It is expected that the formulation of design criteria for a new facility will be aided through the application of the information in this manual.

5. Several DOE facilities have operated for 30 years or more and were designed to less restrictive standards. It is expected that this manual will serve as a guide in the evaluation of needed upgrade programs in these older facilities, or as a guide in establishing control programs, which will provide protection equivalent to that provided in newer facilities.

6. Since it is the DOE policy and a "good practice" to require periodic independent evaluation of the radiological safety program being administered within each facility, this manual should serve as a valuable guide or check list for professionals appraising programs.

7. The data, information, and references contained herein should be of value as a reference source for programs of DOE facilities processing and/or handling uranium metal or any of its many compounds.

1.2 Scope

The scope of this manual is limited to applied problems in the work place. The following statements outline additional scope limitations.

1. This manual will not address environmental issues--including monitoring programs, limits, etc.

2. Only programs and principles associated with operations, applications, or facilities under the control of DOE will be addressed. For example, uranium mining and milling facilities and the associated problems will not be considered.

3. Personnel protection and the necessary information and data to make technically sound evaluations for this purpose will be the primary focus.

4. Waste disposal criteria and policy will not be discussed beyond those aspects of adequate monitoring, documentation, and control of effluents.

SECTION 2

PROPERTIES AND RELATIVE HAZARDS

Uranium is important to the nuclear industry primarily because of the naturally-occurring isotope U-235, which has a high thermal fission cross section. To increase the amount of U-235 fuel in a reactor core and to decrease the size of a reactor, natural uranium is "enriched" in U-235 by special processes such as gaseous diffusion, centrifuging, or laser separation. Uranium depleted in U-235 is also useful as shielding, counterweights, projectiles, target elements in DOE plutonium production reactors, etc.

This section presents basic radiological and chemical data and discusses the basis for current control limits. A variety of hazards are characteristic of the processes and materials inherent to these processes. The data and discussion are intended to provide a basis for understanding the changes in hazards as a function of such parameters as percentage U-235 enrichment, physical form, and chemical form.

2.1 Properties and Relative Hazards of Uranium

Natural uranium consists of three primary isotopes--U-238, U-235, and U-234. The natural abundances of these isotopes, as well as abundances in enriched (typical power reactor enrichment) and depleted uranium, are listed in Table 2-1. The decay products of uranium isotopes are also radioactive and form "decay chains." The decay chains of U-238 and U-235 (U-234 is a member of the U-238 decay chain), along with the half-lives and characteristic radiations of each nuclide, are listed in Tables 2-2 and 2-3.

Since DOE facilities do not routinely process uranium ore concentrates, the only non-uranium members of these decay chains that will be present in virgin feed materials are those that have grown in since the chemical extraction of the uranium. The nuclides that occur in sufficient

TABLE 2-1. TYPICAL ISOTOPIC ABUNDANCES (gm OF ISOTOPE PER 100gm OF NATURAL URANIUM)

Isotope	Natural	Typical Commercial Feed Enrichment	Depleted
U-238	99.2739 + 0.0007	97.01	99.75
U-235	0.7204 + 0.0007	2.96	0.25
U-234	0.0057 + 0.0002	0.03	0.0005

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TABLE 2-2. URANIUM SERIES (4n + 2)[a]

[a] This expression describes the mass number of any member in this series, where n is an integer. Example:

$$^{206}_{82}Pb \quad (4n + 2).....4(51) + 2 = 206$$

† Intensities refer to percentage of disintegrations of the nuclide itself, not to original parent of series.

• Complex energy peak which would be incompletely resolved by instruments of moderately low resolving power such as scintillators.

Nuclide	Historical name	Half-life	Major radiation energies (MeV) and intensities† α	β	γ
$^{238}_{92}U$	Uranium I	4.51×10^9 y	4.15 (25%) 4.20 (75%)	---	---
$^{234}_{90}Th$	Uranium X_1	24.1d	---	0.103 (21%) 0.193 (79%)	0.063c• (3.5%) 0.093c (4%)
$^{234}_{91}Pa^m$	Uranium X_2	1.17m	---	2.29 (98%)	0.765 (0.30%) 1.001 (0.60%)
99.87% 0.13% $^{234}_{91}Pa$	Uranium Z	6.75h	---	0.53 (64%) 1.13 (13%)	0.100 (50%) 0.70 (24%) 0.90 (70%)
$^{234}_{92}U$	Uranium II	2.47×10^5 y	4.72 (28%) 4.77 (72%)	---	0.053 (0.2%)
$^{230}_{90}Th$	Ionium	8.0×10^4 y	4.62 (24%) 4.68 (76%)	---	0.068 (0.6%) 0.142 (0.07%)
$^{226}_{88}Ra$	Radium	1602y	4.60 (6%) 4.78 (95%)	---	0.186 (4%)
$^{222}_{86}Rn$	Emanation Radon (Rn)	3.823d	5.49 (100%)	---	0.510 (0.07%)
$^{218}_{84}Po$	Radium A	3.05m	6.00 (~100%)	0.33 (~0.019%)	---
99.98% 0.02% $^{214}_{82}Pb$	Radium B	26.8m	---	0.65 (50%) 0.71 (40%) 0.98 (6%)	0.295 (19%) 0.352 (36%)
$^{218}_{85}At$	Astatine	~2s	6.65 (6%) 6.70 (94%)	† (~0.1%)	---
$^{214}_{83}Bi$	Radium C	19.7m	5.45 (0.012%) 5.51 (0.008%)	1.0 (23%) 1.51 (40%) 3.26 (19%)	0.609 (47%) 1.120 (17%) 1.764 (17%)
99.98% 0.02% $^{214}_{84}Po$	Radium C'	164μs	7.69 (100%)	---	0.799 (0.014%)
$^{210}_{81}Tl$	Radium C"	1.3m	---	1.3 (25%) 1.9 (56%) 2.3 (19%)	0.296 (80%) 0.795 (100%) 1.31 (21%)
$^{210}_{82}Pb$	Radium D	21y	3.72 (.000002%)	0.016 (85%) 0.061 (15%)	0.047 (4%)
$^{210}_{83}Bi$	Radium E	5.01d	4.65 (.00007%) 4.69 (.00005%)	1.161 (~100%)	---
~100% .00013% $^{210}_{84}Po$	Radium F	138.4d	5.305 (100%)	---	0.803 (0.0011%)
$^{206}_{81}Tl$	Radium E"	4.19m	---	1.571 (100%)	---
$^{206}_{82}Pb$	Radium G	Stable	---	---	---

TABLE 2-3. ACTINIUM SERIES (4n + 3)[a]

[a] This expression describes the mass number of any member in this series, where n is an integer. Example:

$$^{207}_{82}Pb \quad (4n + 3).....4(51) + 3 = 207$$

† Intensities refer to percentage of disintegrations of the nuclide itself, not to original parent of series.

* Complex energy peak which would be incompletely resolved by instruments of moderately low resolving power such as scintillators.
Data taken from Table of Isotopes and USNRDL-TR-802.

Nuclide	Historical name	Half-life	Major radiation energies (MeV) and intensities† α	β	γ
$^{238}_{92}U$	Actinouranium	7.1 x10^{8}y	4.37 (18%) 4.40 (57%) 4.58c* (8%)	---	0.143 (11%) 0.185 (54%) 0.204 (5%)
$^{234}_{90}Th$	Uranium Y	25.5h	---	0.140 (45%) 0.220 (15%) 0.305 (40%)	0.026 (2%) 0.084c (10%)
$^{234}_{91}Pa$	Protoactinium	3.25x10^{4}y	4.95 (22%) 5.01 (24%) 5.02 (23%)	---	0.027 (6%) 0.29c (6%)
$^{227}_{89}Ac$	Actinium	21.6y	4.86c (0.18%) 4.95c (1.2%)	0.043 (~99%)	0.070 (0.08%)
98.6% 1.4%					
$^{227}_{90}Th$	Radioactinium	18.2d	5.76 (21%) 5.98 (24%) 6.04 (23%)	---	0.050 (8%) 0.237c (15%) 0.31c (8%)
$^{223}_{87}Fr$	Actinium K	22m	5.44 (~0.005%)	1.15 (~100%)	0.050 (40%) 0.080 (13%) 0.234 (4%)
$^{223}_{88}Ra$	Actinium X	11.43d	5.61 (26%) 5.71 (54%) 5.75 (9%)	---	0.149c (10%) 0.270 (10%) 0.33c (6%)
$^{219}_{86}Rn$	Emanation Actinon (An)	4.0s	6.42 (8%) 6.55 (11%) 6.82 (81%)	---	0.272 (9%) 0.401 (5%)
$^{215}_{84}Po$	Actinium A	1.78ms	7.38 (~100%)	0.74 (~.00023%)	---
~100% .00023%					
$^{211}_{82}Pb$	Actinium B	36.1m	---	0.29 (1.4%) 0.56 (9.4%) 1.39 (87.5%)	0.405 (3.4%) 0.427 (1.8%) 0.832 (3.4%)
$^{215}_{85}At$	Astatine	~0.1ms	8.01 (~100%)	---	---
$^{211}_{83}Bi$	Actinium C	2.15m	6.28 (16%) 6.62 (84%)	0.60 (0.28%)	0.351 (14%)
0.28% 99.7%					
$^{211}_{84}Po$	Actinium C'	0.52s	7.45 (99%)	---	0.570 (0.5%) 0.90 (0.5%)
$^{207}_{81}Tl$	Actinium C"	4.79m	---	1.44 (99.8%)	0.897 (0.16%)
$^{207}_{82}Pb$	Actinium D	Stable	---	---	---

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abundance to have an impact on radiological controls are Th-234, Pa-234m, and Th-231. The long half-lived Th-230 in the U-238 chain and Pa-231 in the U-235 chain effectively prevent the accumulation of significant quantities of other decay products. Still, some Th-230 and Ra-226 may be found in the process waste water of some facilities, so it is prudent to include those nuclides in effluent/environmental monitoring programs. For workplace radiological controls, Th-234, Pa-234m, Th-231 and the uranium isotopes are those requiring primary consideration. In poorly ventilated areas where uranium is stored, elevated radon concentrations can occur (and have been demonstrated) from the small amounts of Ra-226 which both grow in and carry over as contaminant in the chemical separation processes.

Much of the uranium feed material that is currently handled at DOE facilities has been reclaimed, or recycled, from reprocessed, spent reactor fuel. The chemical processes by which recycled uranium is purified leave trace amounts of transuranic elements (neptunium and plutonium) and fission products (mainly technetium-99). Recycled uranium also contains trace amounts of uranium isotopes not found in nature, such as U-236. At the concentrations in uranium from fuel reprocessing facilities, the radiological impact of these impurities is negligible in many cases. However, there are many routine chemical processes which tend to concentrate these impurities either in the uranium product or in reaction by-products such that radiological controls and effluent/environmental monitoring programs must consider these impurities in some cases.

2.1.1 Radiological Properties

The primary isotopes of uranium are all long-lived alpha emitters. However, several other radionuclides can be radiologically significant at uranium facilities, depending upon the history of the uranium materials and the processing. Table 2-4 is a summary of radionuclides which can have radiological impacts at uranium handling facilities.

TABLE 2-4. RADIONUCLIDES AT URANIUM FACILITIES

Nuclide	Half-Life	Energies (Mev) and Abundances of Major Radiations		
		Alpha	Beta	Gamma
Primary Uranium Isotopes				
Uranium-238 (U-238)	4.51×10^9y	4.15 (25%) 4.20 (75%)	--	--
Uranium-235 (U-235)	7.1×10^8y	4.37 (18%) 4.40 (57%) 4.58 (8%)	--	0.144 (11%) 0.185 (54%) .204 (5%)
Uranium-234 (U-234)	2.47×10^5y	4.72 (28%) 4.77 (72%)	--	.053 (.2%)
Decay Products				
Thorium-234 (Th-234)	24.1d	--	0.103 (21%) 0.193 (79%)	0.063 (3.5%) 0.093 (4%)
Protactinium-234m (Pa-234m)	1.17m		2.29 (98%)	0.765 (0.30%) 1.001 (0.60%)
Thorium-231 (Th-231)	25.5h		0.140 (45%) 0.220 (15%) 0.305 (40%)	0.026 (2%) 0.084 (10%)
Impurities				
Technetium-99 (Tc-99)	2.12×10^5y	--	0.292	--
Neptunium-237 (Np-237)	2.14×10^6y	4.78 (75%) 4.65 (12%)		0.030 (14%) 0.086 (14%) 0.145 (1%)
Plutonium-238 (Pu-238)	86.4y	5.50 (72%) 5.46 (28%)		
Plutonium-239 (Pu-239)	24,390y	5.16 (88%) 5.11 (11%)		0.052 (0.02%)
Plutonium-240 (Pu-240)	6580y	5.17 (76%) 5.12 (24%)		
Plutonium-241 (Pu-241)	13.2 y		0.021	
Uranium-236 (U-236)	2.39×10^7y	4.47 (24%) 4.52 (76%)		

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Enrichment Effect

The specific activity[a] of uranium depends upon 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 values are given in Table 2-5 and Figures 2-1, 2-2, 2-3, and 2-4.

For conventionally enriched uranium, approximate alpha specific activity of a given uranium enrichment can be calculated from the following formula.

$$\text{Specific Activity} = (0.4 + 0.38E + 0.0034E^2) \times 10^{-6}\ \text{Ci/g}$$

where

E = % U-235 by weight.

Specific activity increases with enrichment, not because of the replacement of some U-238 ($T_{1/2} = 4.5 \times 10^9$ years) with U-235 ($T_{1/2} = 7.1 \times 10^8$ years), but primarily because of the increase in the amount of U-234 present ($T_{1/2} = 2.47 \times 10^5$ years). Gaseous diffusion, the existing enrichment technology, causes a greater increase in U-234 than in U-235. For example, when U-235 content is increased from 0.72% (natural) to 2.96%, (a factor of approximately four increase), U-234 content increases from 0.006% to 0.03%, (a five-fold increase).

---

a. The use of the "special curie" of natural uranium has caused (and still causes) considerable confusion. Readers are cautioned to be aware of the use of this special curie in the literature. Use of this unit in any application is strongly discouraged.

The "special curie" of natural uranium was defined as 3.7×10^{10} d/s of U-234, 3.7×10^{10} d/s of U-238, and 1.7×10^9 d/s of U-235. Thus 1 "curie" of natural uranium was actually slightly more than 2 curies of uranium alpha activity.

TABLE 2-5. URANIUM SPECIFIC ACTIVITIES

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

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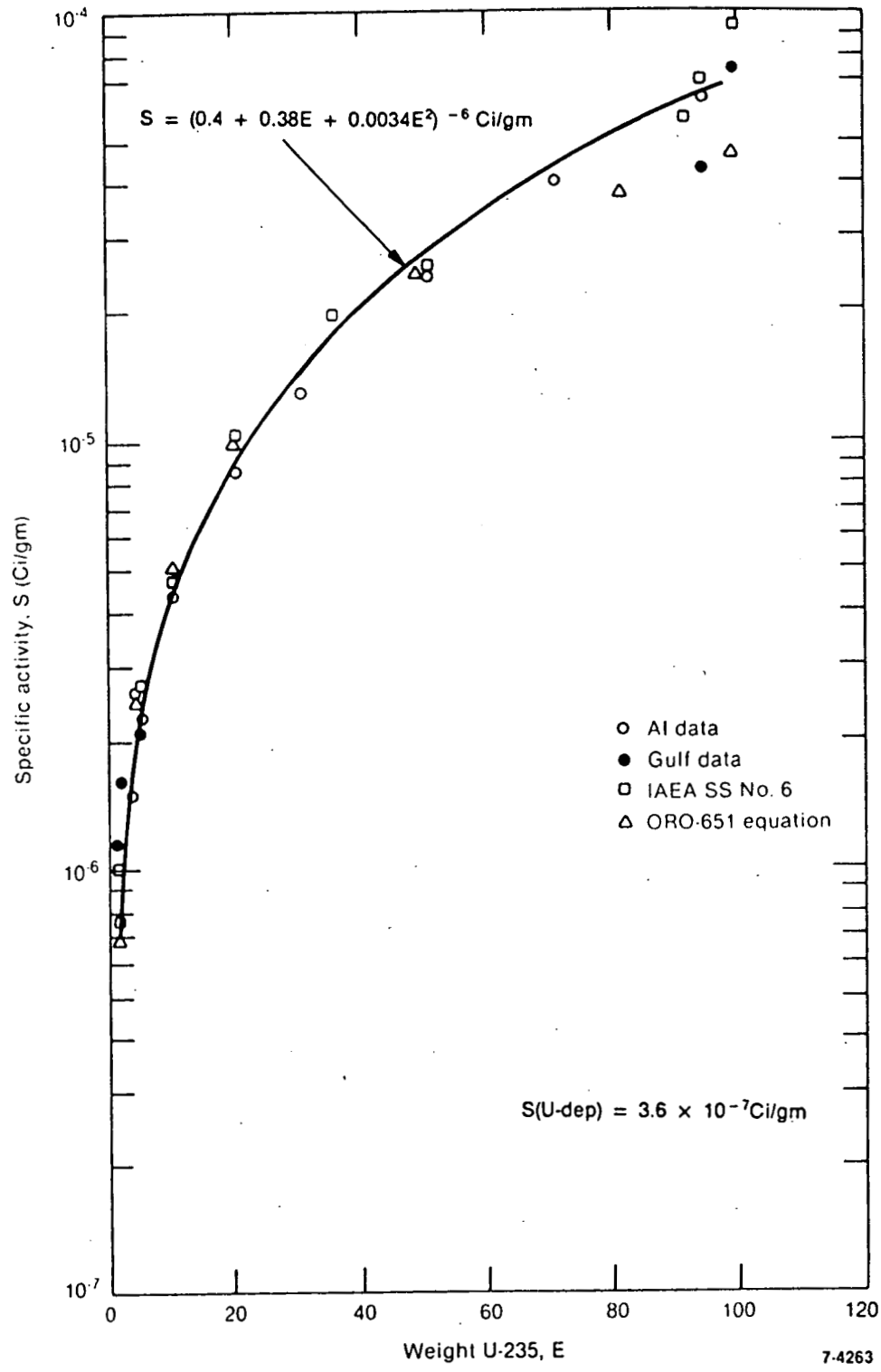

Figure 2-1. Specific activity for mixtures of U-238, U-234, and U-235.

000024

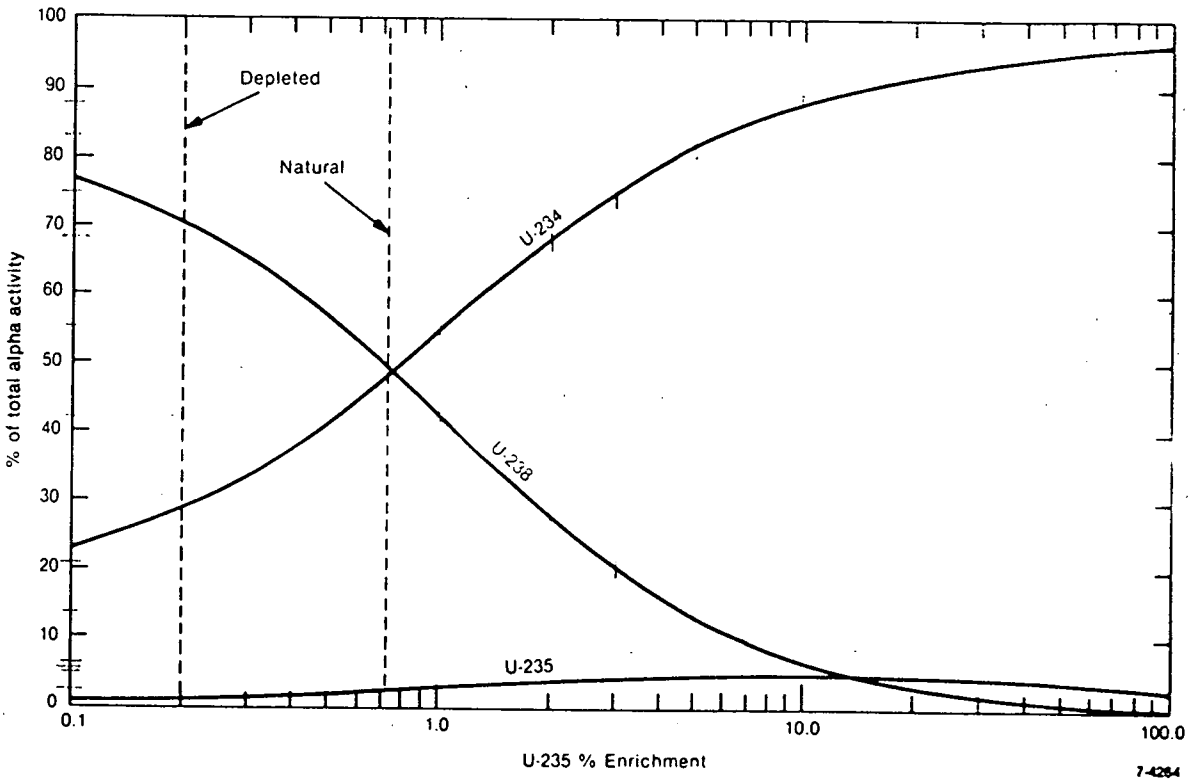

Figure 2-2. Percent of total radioactivity by isotope vs. % weight U-235 enrichment calculated from $S = (0.4 + 0.3\ E + 0.0034E^2)$ 10^{-6} Ci/g (gaseous diffusion process).

000025

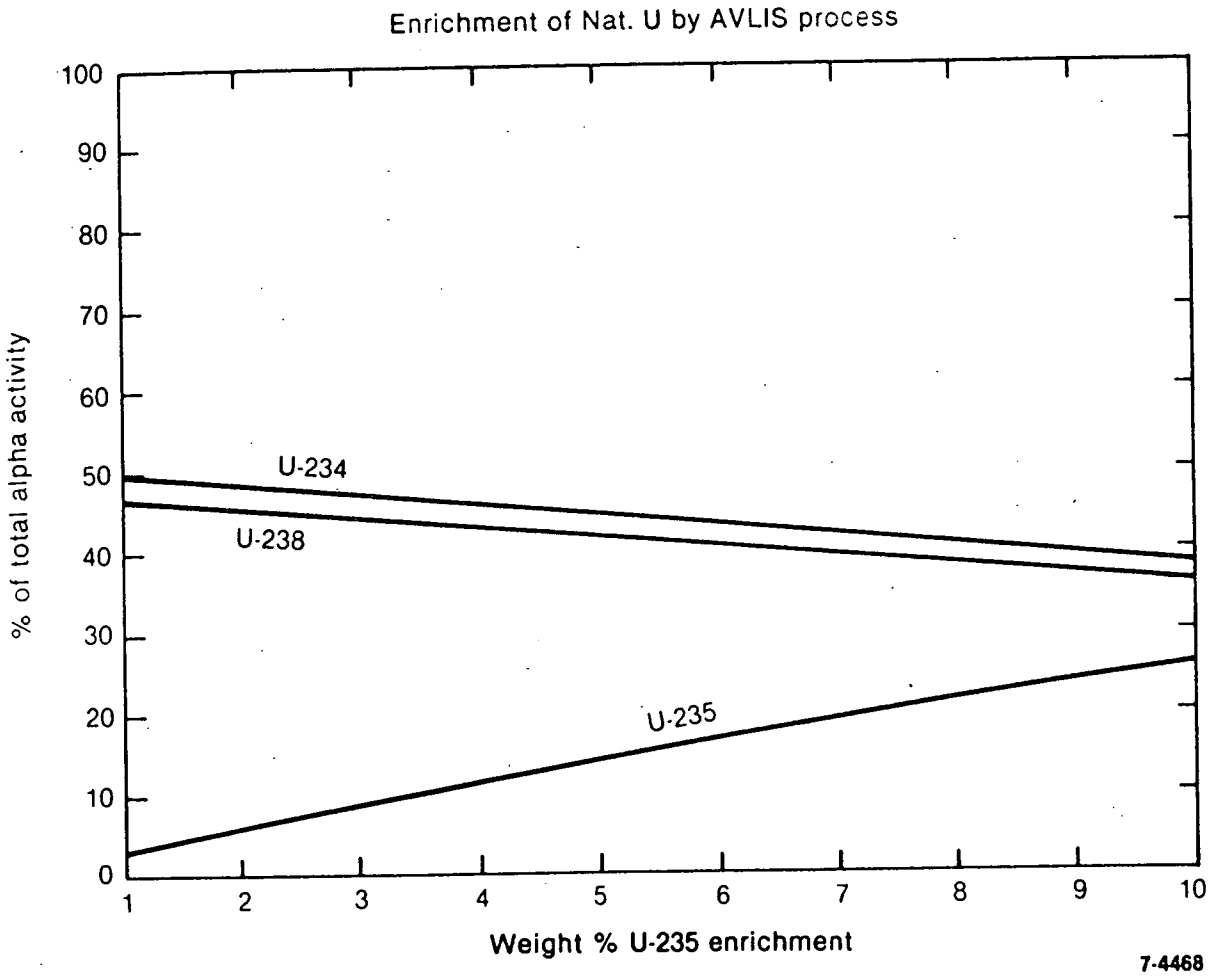

Figure 2-3. Approximate percent alpha activity contribution for ALVIS enriched natural uranium.

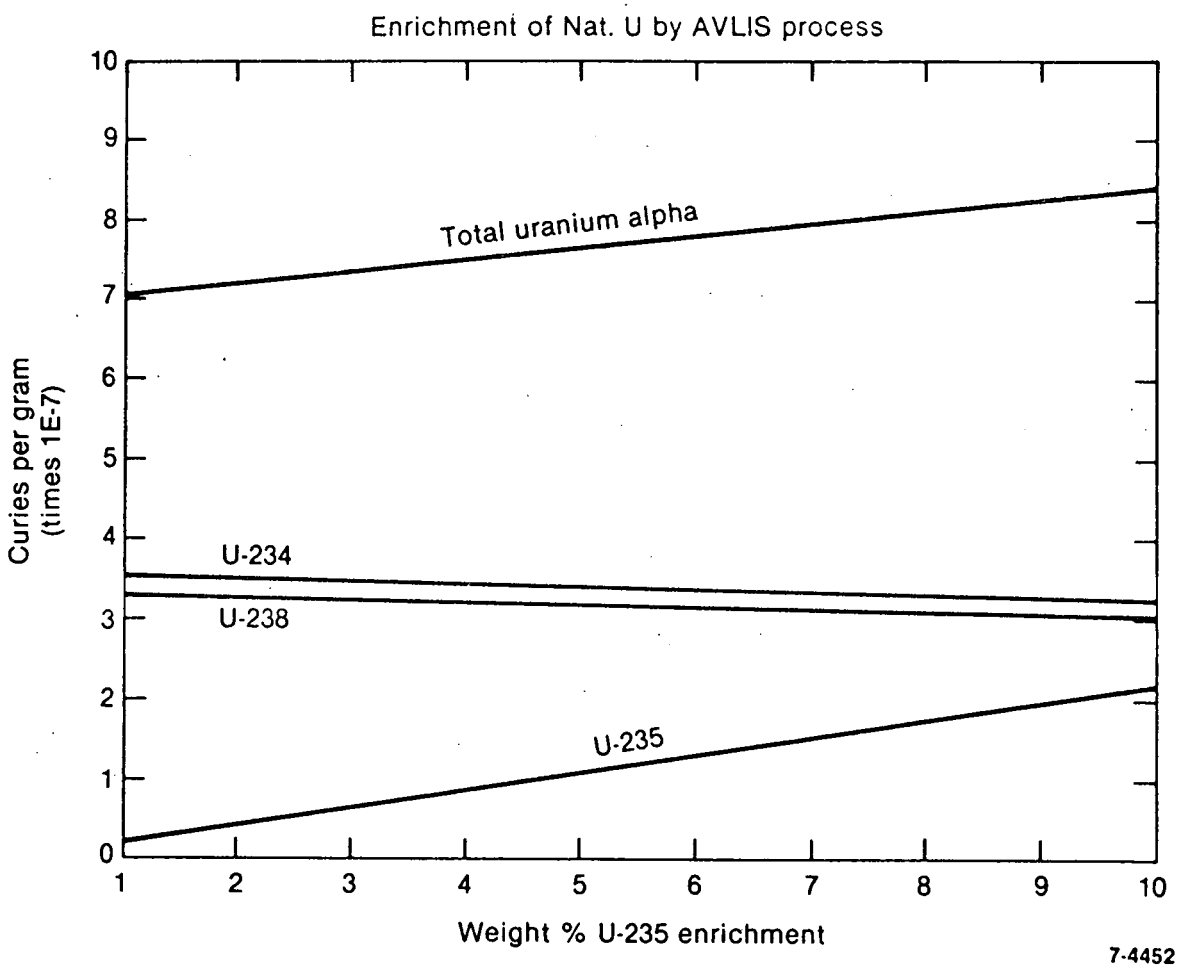

Figure 2-4. Estimated uranium specific activity for AVLIS enrichment (of natural uranium).

000027

Laser isotopic separation (the technology selected for future enrichment facilities) is expected to separate only U-235, leaving the U-234 with the "tails," or depleted uranium. Therefore, when laser-enriched uranium becomes available, the radiological characteristics of both enriched and depleted uranium will change when compared to conventional separation techniques. Figures 2-3 and 2-4 demonstrate this effect.

The half-lives of the three natural uranium isotopes are sufficiently long so that the specific activity of uranium will not change because of radioactive decay, regardless of enrichment. Specific activity does change with enrichment, but does not vary linearly. Blending different enrichments of uranium (a fairly common practice at some DOE facilities) results in specific activities that differ from those calculated from the specific activity equation shown on Figure 2-1. Example 1 illustrates this effect, which limits the effectiveness of the specific equation. The specific activity calculated for 11% enrichment is approximately 6% lower than the value predicted by combining the values calculated for the different enrichments (5.0×10^{-6} versus 5.3×10^{-6}).

The recycling of irradiated uranium also provides a means for obtaining material whose specific activity varies from the value calculated from the equation on Figure 2-1, because that equation is not applicable to recycle material. For these reasons, specific activities that are calculated from the above formula should be considered approximations only. If exact values of specific activity are required, those values should be determined analytically.

The Annual Limit on Intake (ALI) for several radionuclides are shown in Table 2-6 and in Figure 2-5. Since the ALI for the three primary uranium isotopes are expressed in activity units, enrichment has little impact on inhalation and ingestion ALIs. However, as was seen in Table 2-5, as enrichment increases from 2% to 20%, the specific activity increases eight-fold. Consequently, the mass of material that corresponds to one ALI decreases by a factor of 8. The degree of enrichment also

Example 1

[1 kg of 20% enriched U is blended with 1 kg of 2% enriched U.]

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

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

$$= 9.36 \times 10^{-6} \text{ Ci/g}$$

$$SA_2 = [0.4 + 0.38(2) + 0.0034(2)^2] \times 10^{-6} \text{ Ci/g}$$

$$1.17 \times 10^{-6} \text{ Ci/g}$$

The specific activity of the resulting mixture is

$$\frac{(9.36 + 1.17)}{2} \times 10^{-6} \text{ Ci/g} = 5.27 \times 10^{-6} \text{ Ci/g}$$

The enrichment of the final mixture is

$$\frac{(200 \text{ g} + 20 \text{ g}) \text{ U-235}}{2000\text{g total U}} = 11\% \text{ U-235}$$

Using the equation for specific activity,

$$SA = [0.4 + 0.38(11) + 0.0034(11)^2] \times 10^{-6} \text{ Ci/g}$$

$$= 4.99 \times 10^{-6} \text{ Ci/g}$$

000029

TABLE 2-6. ANNUAL LIMITS ON INTAKE AND DERIVED AIR CONCENTRATIONS FOR SELECTED RADIONUCLIDES (FROM ICRP-30)

Nuclide	Inhalation		
	Class D	Class W	Class Y
ANNUAL LIMITS ON INTAKE (First values are in units of μCi; values in parentheses are in Bq.)			
	μCi (Bq)	μCi (Bq)	μCi (Bq)
U-238	1 (5×10^4)	8×10^{-1} (3×10^4)	5×10^{-2} (2×10^3)
U-235	1 (5×10^4)	8×10^{-1} (3×10^4)	5×10^{-2} (1×10^3)
U-234	1 (5×10^4)	8×10^{-1} (3×10^4)	3×10^{-2} (1×10^3)
Th-234	NL	2×10^{-2} (7×10^6)	2×10^{-2} (6×10^6)
Pa-234m	NL	NL	NL
Th-231	NL	5×10^3 (2×10^8)	5×10^3 (2×10^8)
Tc-99	5×10^3 (2×10^8)	5×10^2 (2×10^7)	NL
NP-237	NL	5×10^{-3} ($\times 10^2$)	NL
Pu-238	NL	5×10^{-3} (2×10^2)	2×10^{-2} (6×10^2)
Pu-239	NL	5×10^{-3} (2×10^2)	1×10^{-2} (5×10^2)
Pu-240	NL	5×10^{-3} (2×10^2)	1×10^{-2} (5×10^2)
Pu-241	NL	3×10^{-1} (1×10^4)	5×10^{-1} (2×10^4)
U-236	1 (5×10^4)	8×10^{-1} (3×10^{-4})	3×10^{-2} (1×10^3)
INHALATION DAC (First values are in units of uCi/ml; values in parentheses are in units of Bq/m^3)			
	μCi/ml (Bq/m^3)	μCi/ml (Bq/m^3)	μCi/ml (Bq/m^3)
U-238	5×10^{-10} (2×10^1)	3×10^{-10} (1×10^1)	2×10^{-11} (7×10^{-1})
U-235	5×10^{-10} (2×10^1)	3×10^{-10} (1×10^1)	2×10^{-11} (6×10^{-1})
U-234	5×10^{-10} (2×10^1)	3×10^{-10} (1×10^1)	2×10^{-11} (6×10^{-1})
Th-234	NL	8×10^{-8} (3×10^3)	6×10^{-8} (2×10^3)
Pa-234m	NL	3×10^{-6} (1×10^5)	--
Th-231	NL	3×10^{-6} (1×10^5)	3×10^{-6} (1×10^{-5})
Tc-99	2×10^{-6} (8×10^4)	3×10^{-7} (1×10^4)	NL
NP-237	NL	2×10^{-12} (9×10^{-2})	NL
Pu-238	NL	3×10^{-12} (9×10^{-2})	8×10^{-12} (3×10^{-1})
Pu-239	NL	2×10^{-12} (8×10^{-2})	5×10^{-12} (2×10^{-1})
Pu-240	NL	2×10^{-12} (8×10^{-2})	5×10^{-12} (2×10^{-1})
Pu-241	NL	1×10^{-10} (4)	3×10^{-10} (1×10^1)
U-236	5×10^{-10} (2×10^1)	3×10^{-10} (1×10^1)	2×10^{-11} (6×10^{-1})

NL = Not listed.

NOTE: ALI and DAC values are only defined to one significant digit.

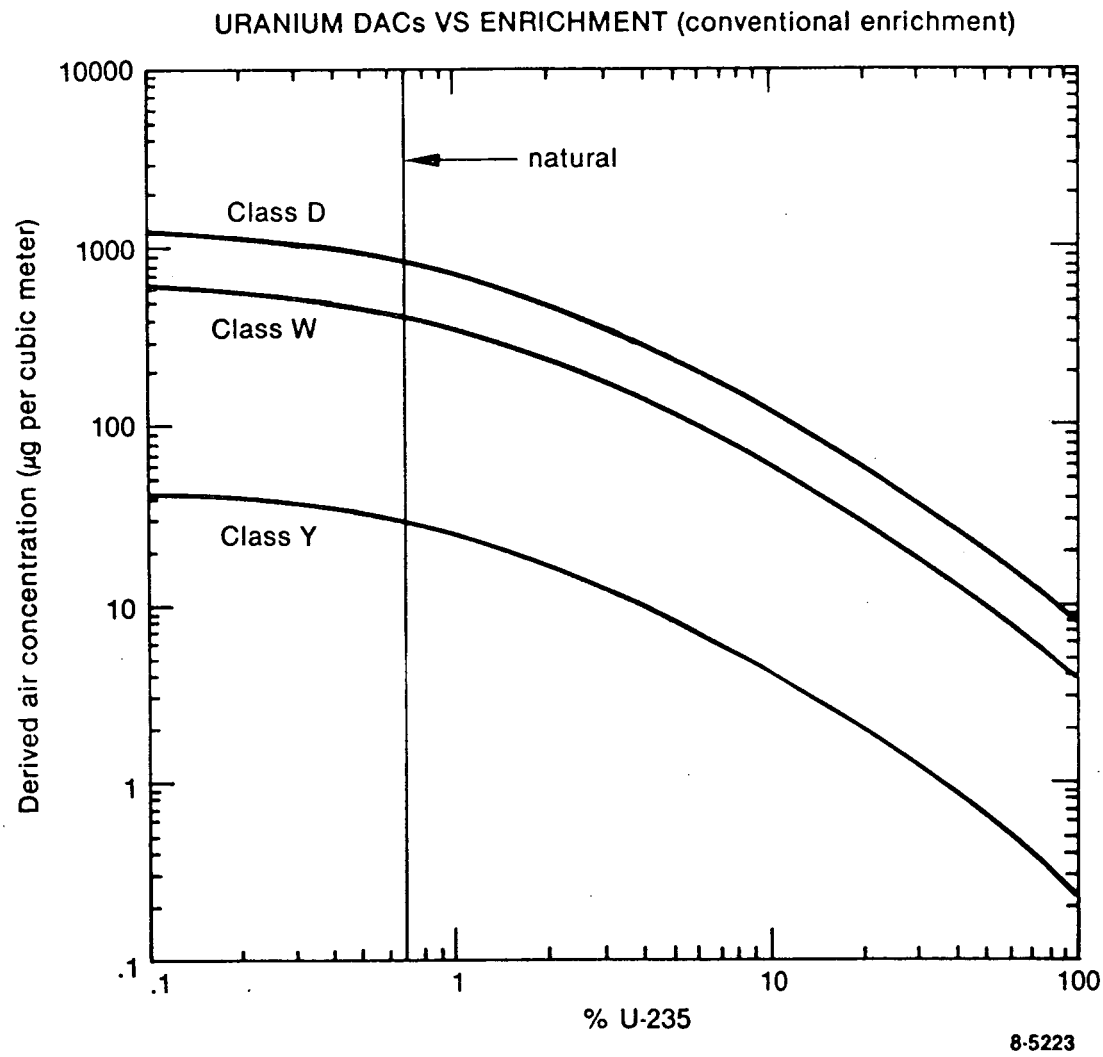

Figure 2-5. DACs vs. weight % U-235 enriched by gaseous diffusion process.

affects the controls that are required for external penetrating radiation exposure, because of the increase in the amount of gamma-emitting U-235 that is present.

Hazards From Decay Products

The uranium decay products, listed in Table 2-4, all decay by beta particle emission. Consequently, the inhalation hazards associated with these nuclides is usually overshadowed by that from the alpha-emitting uranium isotopes. The decay products do give rise to shallow or eye dose equivalent external radiation exposures, due mainly to the 2.29 MeV E max beta from Pa-234m. The dose rates shown on Table 2-7 result primarily from beta radiation from decay products.

The fact that some uranium decay products have short-lives (on the order of days) indicates that those decay products will usually be present with uranium during processing. An assumption of secular equilibrium should not be made, however, because many routine chemical processing steps separate uranium from its decay products. When this occurs 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 probably decrease in those areas due to the removal of the uranium.

Hazards From Recycled Uranium Contaminants

Much of the current enrichment feed is recycled materials. It is anticipated that the amount of recycled uranium (RU) will increase over the next several years. This is due to the plans of a number of nations to reprocess uranium from their power reactors. The environmental, safety and health challenges presented by the introduction of the increased quantities of RU into the DOE system for enrichment are briefly discussed below.

TABLE 2-7. BETA SURFACE DOSE RATES FROM EQUILIBRIUM THICKNESS OF URANIUM METAL AND COMPOUNDS.

Source	Surface Dose Rate* (mrad/hr)
Ntl. U metal slab	233
UO_2	207
UF_4	179
$UO_2(NO_3)_2 6H_2O$	111
UO_3	204
U_3O_8	203
UO_2F_2	176
$Na_2U_2O_7$	167

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

000033

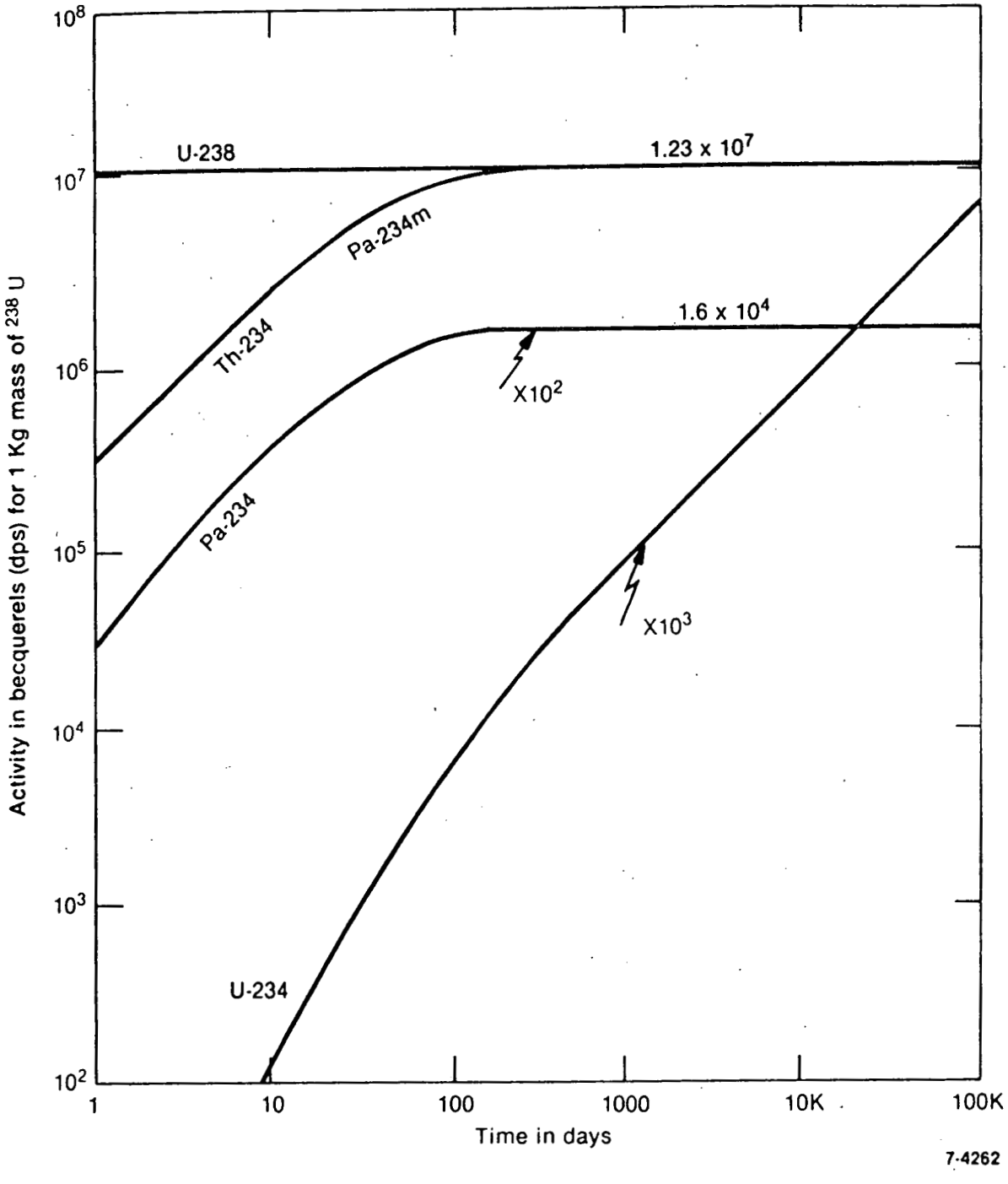

Figure 2-6. U-238 decay product ingrowth.

The isotopes of primary concern from RU are Tc-99, U-232, Np-237 and Pu-238 and 239. Technetium-99 tends to deposit within enrichment equipment and will "pocket" in the higher enrichment sections of the gaseous diffusion process. This requires special precautions in evacuating and purging equipment as well as special precautions prior to maintenance work. In equipment with accumulations of Tc-99, "soft" beta radiation fields of a few rad per hour may be encountered. This radiation is effectively attenuated by the protective clothing required for contamination control (one pair of industrial cloth coveralls, one pair of impermeable (Tyvek) coveralls and heavy neoprene gloves). While the Tc-99 should be effectively removed from the Gaseous Diffusion Plant (GDP) product, it will be present in uranium used in other DOE facilities. Since the ALI for Tc-99 is higher than that of uranium, inhalation is a concern only in situations where the technetium activity greatly exceeds that of the uranium that is present. As indicated, this condition can exist in certain locations near the top of a gaseous diffusion cascade because of the low atomic weight of technetium and its relative volatility. Technetium as pertechnetate is also difficult to remove from skin, which can result in skin contamination and significant skin doses.

The uranium isotopes (viewed as contaminants) that will increase due to the RU feed are U-232, U-234, and U-236. The U-236 will not pose much of a concern from a health and safety standpoint because its specific activity and radiation type are similar to those for the natural uranium isotopes. However, its presence will require higher enrichments for the same reactor applications. The effect of the U-234 is to increase the specific activity of any given enrichment of U-235. It is expected that the specific activity for a given enrichment would be about double that obtained from enrichment of normal uranium.

The isotope that would cause the major problems will be the U-232. The health hazards of U-232 are primarily due to the gamma activity of its decay products. The gamma emission of primary concern is due to the 1.9 year half-life decay product Th-228. The buildup of the gamma activity

is both time and process dependant. The U-232 decay products form non-volatile fluorides and will concentrate in cyclinders when UF_6 is vapor fed. The gamma activity in equipment processing gaseous UF_6 will be a function of the mass fraction of U-232 present in the gas phase. Estimates have been made which indicate that the level of gamma activity of enrichment cascade equipment will increase by a factor of 3. The activity on internal surfaces would increase from 10-20 mrad/hr to 30-60 mrad/hr; external surfaces would increase to about 3-4 mrad/hr. The major exposure increase from the U-232 will be in the handling of UF6 cylinder. Currently, the radiation field at the external surface of empty UF_6 cylinders is about 50-100 mrad/hr. Assuming a U-232 concentration of 0.5 ppm based on U-235 and a feed enrichment of 1%, a full 10-ton feed cylinder would have a surface radiation field of 80 mrad/hr and a reading of 500 mrad/hr at 30 cm from the cylinder surface when the cylinder is empty. These values are based on the U-232 being in equilibrium with its decay products; in reality, it would be unlikely that the decay products would reach much more that 50% of equilibrium values. Product cylinders would have higher gamma fields that feed cylinders. At 4% U-235 enrichment, a full 10-ton cylinder is expected to have a gamma field at the cylinder surface of 300 mrad/hr; at 30 cm from an empty cylinder, a gamma field of approximately 2 rad/hr will exist. While it would take 20 years without mitigating actions for this field to exist, approximately 50% of this level will be present in about 2 years. However, this problem can be significantly ameliorated by frequent cylinder cleaning. The U-232 will require some change in handling of cleaning solutions due to the higher gamma radiation present.

Transuranics

Transuranics (neptunium and plutonium isotopes) will exist in small quantities in feed materials. In most cases the radiological controls based on uranium hazards potential and concerns will be adequate to control the additional activity concerns presented. However, because of their higher specific activities and lower ALIs (compared to uranium isotopes),

transuranics can represent a significant internal dose concern even at very low mass concentrations. For example, for a Class W transportability mixture if plutonium-239 contamination contributes 0.1% of the total alpha activity in uranium, then it will contribute roughly 14% (see Example 2) of the total inhalation dose equivalent because of its lower ALI. Example 2 illustrates that it takes only 11 parts of Pu-239 per billion parts of natural uranium to attain an activity fraction of 0.1%.

Several DOE facilities have adopted specifications on recycled uranium that limit the amount of contained transuranic alpha activity to 0.1% of the total uranium alpha activity. In this way, the potential inhalation dose from transuranics is limited to a fraction of the total potential inhalation dose. 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 modify control limits and action levels as appropriate to reflect the transuranic content of those materials. As the transuranic-to-uranium ratio increases, radiological controls, based on uranium, become inappropriate because of the substantially lower Annual Limits for Intake for transuranics; more stringent controls are necessary. This is especially true when the analytical technique used for radiological control is gross alpha counting (such as for air sampling) or chemical analysis for uranium (such as photofluorometric urinalysis). Both techniques will underestimate the consequence of an analytical result, if consideration is not given to the accompanying transuranics. 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 contained uranium. When this is done, a portion of the impurities is recovered as well, and the radiological impurities in reaction by-products can become a perpetual 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 which are present.

000037

Example 2

One gram of U-Nat contains Pu-239 contamination to the extent that the Pu-239 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

Pu-239 relative activity = .001

U-Nat derived concentration (W) = 3×10^{-10} uCi/ml

Pu-239 derived air concentration (W) = 2×10^{-12} uCi/ml

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

$$\frac{0.001}{CG_a(\text{Pu-239})} = \frac{0.001}{2 \times 10^{-12}} = 5 \times 10^{8}$$

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

$$\text{Fraction of total hazard} = \frac{5 \times 10^{8}}{(5 \times 10^{8}) + (3 \times 10^{9})} = 0.14$$

Therefore, Pu-239 at 0.1% of the U-Nat activity represents 14% of the potential inhalation dose.

The Activity of 1 gram of U-Nat = 2.5×10^{4} dps

$0.001 \times 2.5 \times 10^{4} = 2.5 \times 10^{1}$ dps = the Pu-239 activity in the 1 gram of U-Nat

The specific activity of Pu-239 is 2.27 dps/nanogram

$$25 \text{ dps/g U} \cdot \frac{1 \text{ nanogram Pu}}{2.27 \text{ dps}} = \frac{11 \text{ nanogram Pu}}{\text{gram U}}$$

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

Technetium

In facilities with significant quantities of Tc-99, radiation measurement techniques must consider the low-energy beta radiation from Tc-99. Dosimeters that were developed to measure principally the 2.29 meV E_{max} beta from Pa-234m, may not be effective at measuring doses from Tc-99's 0.292 meV E_{max} beta. Similarly, contamination survey techniques and action levels that are based on Pa-234m beta particles may not be appropriate for the low energy Tc-99 betas. If a mixture of uranium and Tc-99 is present, survey technique must take into account the low-energy beta radiation from Tc-99 and should be based on Tc-99 or on the actual mixture, rather than on Pa-234m. Tc-99 levels have not been the controlling factor in many situations to date. The primary concern is to assure that instruments and survey techniques are adequate to detect Tc-99.

In uranium metal processing facilities it has been observed that residues in ventilation systems from high-temperature operations, such as uranium remelting/casting, or uranium chip burning, will tend to have higher Tc-to-U ratios than either feed or product material. The tendency of technetium to become airborne more readily than uranium can lead to Tc-99 contamination in areas where it is not expected, and environmental emissions even when the uranium is effectively confined in the work place. Technetium also tends to concentrate at the top of the gaseous diffusion cascade, where it becomes an inhalation and effluent problem when the cascade is opened for maintenance. Facilities that handle recycled uranium should analyze feeds, products, and by-products in order to determine the fate of Tc-99 within their processes, then modify monitoring equipment, control limits, and action levels as needed to ensure that doses from Tc-99 are properly evaluated and controlled.

Alpha-Neutron Hazard

Neutrons of approximately 2 MeV energy are generated by the interaction of alpha particles from uranium with the nuclei of fluoride and

other low-Z atoms. The magnitude of the neutron flux will vary 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/hr.

Very High Enrichment (97+%): 2-4 mrem/hr contact
1-2 mrem/hr 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 connected with empty containers while the maximum neutron exposures are connected with full containers.

2.1.2 Chemical Toxicity

Historically, the chemical toxicity of uranium has been a primary concern in establishing control limits and procedures. As a heavy metal, U is chemically toxic to kidneys and high exposure to soluble (transportable) compounds can result in renal injury. Although radiologically based controls are now receiving more attention than in the past, chemical toxicity is still an important consideration.

A concentration of 3 ug of uranium per gram of kidney tissue has been used as the guideline for controlling the chemical toxicity of uranium. Standard man has a kidney mass of 310 g, so this concentration translates to a total kidney burden of 1 mg.

Table 2-8 lists airborne concentration limits for transportable uranium that have been published by various organizations. Based on the 3 µg/gm of tissue values, 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). The Occupational Safety and Health Administration (OSHA) has adopted a limit of 0.050 mg/m^3.

Past limits for single acute inhalation intakes have been set by the International Commission on Radiological Protection (ICRP) (ICRP6) to 2.5 mg of soluble uranium inhaled in any one day. This value is based on 1 day's intake at the maximum permissible concentration (at the time) of 210 ug/m3. Lawrence (Lawrence 1984) has derived acute inhalation intake limits of 15 and 80 mg for Class D and Class W materials respectively. This derivation is based on not exceeding a kidney burden of 3 ug U per g kidney after a single acute inhalation. Human ingestion studies have indicated that an acute uptake of 0.1 mg uranium per kg body weight would not be considered harmful (Hursch and Spoor). For 70-kg standard man, this represents a 7 mg uptake, or a 15 mg intake of Class D material.

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

000041

TABLE 2-8. TOXICOLOGICAL LIMITS ON AIRBORNE CONCENTRATIONS OF TRANSPORTABLE (SOLUBLE) URANIUM

Agency	Chronic Exposure Occupational Limit (mg/m^3)	Reference
NRC	0.2	Footnote to Appendix B, 10 CFR 20
ACGIH	0.2	Threshold Limit Values and Biological Exposure Indices for 1986-1987, American Conference of Governmental Industrial Hygienists
OSHA[a]	0.05	29 CFR 1910.1000

a. Preferred/recommended limit--see Analyses that follow.

In the past few years, concerns have arisen about the adequacy of existing limits intended to prevent chemical damage to kidneys. In 1979 DOE contracted with researchers from the University of Rochester and the University of Utah to establish exposure conditions expected to cause varying degrees of injury to humans. The researchers expressed considerable reservations about:

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

c. Lack of data on thresholds for repairable injury.

Consequently, additional research was undertaken following which concensus was reached on 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}). Those uptake levels (in mg U/kg) are listed in Table 2-9 along with the corresponding total U in 70 kg standard man.

The values for "standard man" are based on the ICRP-30 model for uranium metabolism (47.6% of inhaled Class D uranium is taken up into the bloodstream, and 12% of that goes to the kidneys). Therefore, the "no effect" value in Table 2-9 corresponds to a kidney burden of (5.9) (.476) (.12) = 0.337 mg. The mass of kidney tissue in standard man is 310 g, so this kidney burden represents 1.1 ug U per gram of kidney tissue.

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

000043

TABLE 2-9. URANIUM LEVELS FOR VARIOUS EFFECTS.

Effect	Uranium Absorbed Into Bloodstream (mg U/kg of body weight)	Corresponding Class D Uranium Intake in Standard Man (mg)
No effect	0.04	5.9
Maximal Non-lethal	0.08	11.6
LD_{50}	2.0	294

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

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

where

K = Kidney Uptake Rate (mg/day)

B_r = Breathing Rate (m^3/day)

C_a = Air concentration (mg/m^3)

f_b = Inhaled fraction entering bloodstream (0.476)

f_k = Bloodstream fraction entering kidneys (0.12)

$$R = \lambda K_b$$

where

R = Kidney removal rate (mg/day)

λ = $\frac{0.693}{T_{1/2}}$ (day^{-1})

K_b = Amount in the kidney (mg)

$T_{1/2}$ = Biological half-life of U in kidney = 6 days

In order 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.

$$R = (0.337) \times \frac{0.693}{6} = 0.039 \text{ mg/day}$$

$$K = B_r \ (m^3/\text{day}) \times C_a (\text{mg}/m^3) \times (0.476) \times (0.12)$$

$$K = R = 0.039 \text{ mg/day}$$

$$B_r C_a \times (0.476) \times (0.12) = 0.039 \text{ mg/day}$$

$$B_r \times C_a = 0.68 \text{ mg/day}$$

Standard man breathes 9.6 m^3 of air in an 8-hour day, so the resulting concentration limit is 0.68/9.6 = 0.07 mg/m^3.

Therefore, an airborne contamination limit of 0.07 mg/m^3 for transportable uranium appears to be appropriate. This is approximately the same as the OSHA standard of 0.050 mg/m^3. Consequently, the OSHA limit is recommended for exposures to soluble/transportable (i.e., Class D) uranium unless enrichment dictates more stringent controls based an radiological concerns.

2.2 Human Response Indicators

Most data on human response to uranium exposure has come from accidental exposures (generally UF_6 releases). Accidental exposures to UF_6 have resulted in fatalities on at least three occasions. The fatalities were caused primarily by the HF that was formed by hydrolysis of UF_6 rather than the UF_6 itself. Several individuals have received high, non-fatal exposures. Those individuals who recovered experienced pulmonary edema, nausea, vomiting, abdominal cramps, and chemical burns on the skin due to HF formed by hydrolysis of UF_6. In addition urinary abnormalities, such as transient albuminuria and the presence of red cells and casts, were observed, as was retention of nitrogenous products such as urea and nonprotein 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, glucosuria, and polyuria (increased 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 gastro intestinal (GI) tract. Very little uranium is absorbed into the bloodstream from the GI tract.

In the event of an acute exposure to highly transportable (Class D) uranium compounds, urine samples should be collected 3-4 hours post-exposure and analyzed for uranium as soon as possible. If the uranium concentration is less than 2.0 mg/l, it is unlikely that any significant kidney damage has 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 is available. Lacking that information, it is best to check for albuminuria as an indicator of kidney damage. If kidney damage is indicated, 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 is indicative of a very serious exposure with potentially life threatening consequences and would indicate immediate hospitalization.

2.3 Comparative Hazards

Both the chemical and radiological hazards of uranium are moderate when compared to those of other industrial materials and radionuclides. Table 2-10 compares Threshold Limit Values (TLV) published by ACGIH for uranium and selected other metals, while the bottom half of Table 2-6 gives Derived Air Concentrations from ICRP-30 for selected radionuclides. 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 primary hazard associated with uranium depends upon its degree of enrichment, its chemical form, and its physical form. The degree of enrichment determines the gamma radiation intensity and the overall specific activity.

The effect that enrichment has on specific activity is illustrated in Figure 2-2. That figure (adapted from NRC Regulatory Guide 8.11) also gives 3.6×10^{-7} Ci/g as the specific activity of depleted uranium and lists the formula used in Section 2.1.1 for calculating specific activity of enriched uranium.

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

Chemical form determines solubility and consequent transportability in body fluids. All materials are classified by ICRP into three inhalation classes--D, W, and Y. Class D is most transportable (pulmonary removal half-time of days), Class Y is least transportable (removal half-time of years), and Class W is an intermediate category (removal half-time of

TABLE 2-10. ACGIH[a] THRESHOLD LIMIT VALUES (TLVs) FOR SELECTED METALS

Metal	Soluble and Insoluble TLV	
	TLV-TWA (mg/m^3)	TLV-STEL (mg/m^3)
Uranium	0.2	0.6
Beryllium	0.002	--
Lead	0.15	0.45
Mercury Vapor (all forms except alkyl)	0.05	--
Arsenic	0.2	--

a. American Conference of Governmental Industrial Hygienists

TLV - TWA = Threshold Limit Value - Time-Weighted Average

TLV STEL = Threshold Limit Value - Short Term Exposure Limit

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weeks). The transportability of an inhaled or ingested material determines its fate within the body, and therefore, the resulting dose or chemical effect. Table 2-11 lists several common uranium compounds and their assigned transportability classes.

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 nondispersible forms generally do not constitute an ingestion or inhalation hazard.

2.4 Radiological vs Toxic Limits

Since inhalation of uranium potentially poses both radiological and toxic hazards, determinations must be made about which is most limiting, and in what situations one hazard or the other can be neglected. When radiological hazards are limiting, chemical hazards can generally be neglected except in overexposure situations. When chemical hazards are limiting, radiological hazards (i.e., organ doses and effective dose equivalents) can be neglected only if radiation doses are below regulatory concern as defined by DOE Order 5480.11, Operations Offices implementing order, or other guidance from DOE or internal dosimetry. Radiological monitoring is required for individuals who might exceed 10% of an established quarterly or annual limit. For this reason it is prudent to calculate organ doses and effective dose equivalent for all significant intakes (see Section 6.8), since additional exposures in the same year may result in a total dose in excess of 10% of a dose limit. Even in low potential exposure level situations it is advisable to provide sufficient monitoring to demonstrate a comprehensive dosimetry/control program, which can prove invaluable in public reactions concerning possible future legal litigation in addition to providing basic worker protection.

TABLE 2-11. INHALATION CLASSIFICATION FOR SOME URANIUM COMPOUNDS

Uranium hexafluoride	UF_6	Class "D"[a]
Uranyl fluoride	UO_2F_2	Class "D"[a]
Uranyl nitrate	$UO_2(NO_3)_2$	Class "D"
Uranyl acetate	$UO_2(C_2H_3O_2)_2$	Class "D"
Uranyl chloride	UO_2Cl_2	Class "D"
Uranyl sulfate	UO_2SO_4	Class "D"
Uranium trioxide	UO_3	Class "D"
Uranium tetrafluoride	UF_4	Class "W"[a]
Uranium oxide	U_3O_8	Class "W"*
Uranium dioxide	UO_2	Class "W"*
Uranium tetroxide	UO_4	Class "W"
Ammonium diuranate	$(NH_4)_2 + U_2O_7$	Class "W"*
Uranium aluminide	UAl_x	Class "Y"[a]
Uranium carbide	UC2	Class "Y"
Uranium-zirconium alloy	UZr	Class "Y"
High-fired uranium dioxide	UO_2	Class "Y"*

a. "D", "W", and "Y" are inhalation solubility classes established by the International Commission on Radiological Protection. "D" class material is very soluble; lung retention time in days; "W" class material is moderately soluble; lung retention time in weeks; "Y" class material is relatively insoluble; lung retention time in years.

* Ammonium diuranate is known to contain uranium as UO_3, and should not be assigned to a single inhalation class (El-85), also, the solubility of uranium oxides is very dependent on heat treatment. Although references assign inhalation classes to various uranium compounds, it is recommended that solubility studies be performed in order to characterize the actual materials present.

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The determination of the limiting hazard (chemical or radiological) depends upon transportability (solubility in body fluids), enrichment, and duration of exposure (chronic or acute). As was shown in subsection 2.1.2, the "no effect" value of intake corresponds to a kidney burden of 0.337 mg and a chronic exposure airborne contamination level of 0.07 mg/m^3 for standard man. The 0.337 mg kidney burden, and ICRP-30 metabolic models are used in the following examples to determine the relative hazards for various exposure situations. Relative hazards are determined using both the OSHA standard of 0.05 mg/m^3 and the derived airborne contamination limit of 0.07 mg/m^3. The derivation is first provided in a general form, then variable values corresponding to specific exposure situations are provided, and a resulting table of relative hazards is generated.

To determine which hazard is limiting for a given exposure condition, the intake which corresponds to the "no effect" kidney burden is first calculated; the appropriate Annual Limit on Intake (ALI) for acute exposure or Derived Air Concentration (DAC) for chronic exposure is obtained; then the formula for specific activity is solved in order to determine the enrichment at which the ALI or DAC is equal to the "no effect" intake. This enrichment forms the "dividing line" between chemical and radiological effects as the limiting hazard. Exposures to higher enrichment are limited by radiological effects; exposures to lower enrichments by chemical effects.

The following variables are used in the general derivations in Examples 3a and 3b:

f_b = Fraction of inhaled uranium that promptly enters bloodstream

f_k = Fraction of uranium in bloodstream that enters kidneys

S = Specific activity in Ci/g obtained from $\frac{\text{ALI}}{\text{Intake}}$ or $\frac{\text{DAC}}{\text{Conc.}}$ in Step 2 of each example

B_r = Breathing rate for an 8-hour work day = 9.6 m^3

Example 3a

General Solution--Acute Exposure

STEP 1. Determine intake that results in kidney burden of 0.337 mg.

$$I = B_r \times C_a \times t$$

where

I = intake (mq)

B_r = Breathing Rate (m^3/day)

C_a = Air concentration (mg/m^3)

t = Exposure time (day)

$$I \times f_b \times f_k = 0.337$$

$$I = \frac{0.337 \text{ (mg)}}{f_b \cdot f_k}$$

STEP 2. Assume that the specific activity of the material is such that this intake is equivalent to one ALI. Then divide appropriate Annual Limit on Intake by intake determined in Step 1.

$$\frac{\text{ALI (Bq)}}{\text{Intake (mg)}} \cdot \frac{1000 \text{ mq}}{\text{g}} \cdot \frac{\mu\text{Ci}}{3.7 \times 10^4 \text{Bq}} = \frac{1}{37} \cdot \frac{\text{ALI}}{\text{Intake}} \text{ (}\mu\text{Ci/g)}$$

STEP 3. Use quadratic formula and equation for determining specific activity to calculate the enrichment which corresponds to the specific activity obtained in Step 2.

$$S = (0.4 + 0.38E + 0.0034E^2)\ \mu Ci/g$$

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

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

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

STEP 4. One solution will be less than zero. The other will be the enrichment which is the "dividing line" between chemical and radiological effects.

---

* Specific activity = SA x 10^{-6} Ci/g

Example 3b

General Solution--Chronic Exposure

STEP 1. Determine concentration at which chronic exposure results in an equilibrium kidney burden of 0.337 mg.

For equilibrium conditions,

$K = R$

where

K = Kidney uptake rate (mg/day)

R = Kidney removal rate (mg/day)

$$K = B_r \quad C_a \quad f_b \quad f_k = R = K_b \quad = .693\ K_b/T_B$$

$$(9.6\ MB)\ (C_a)\ (f_b)\ (f_k) = (0.337)\ \frac{(0.693)}{6}$$

$$C_a = \frac{(0.337)\ (0.693)}{(9.6)\ (f_b)\ (f_k)\ (6)}$$

STEP 2. Assume that the enrichment of the material is such that the concentration determined in Step 1 is equivalent to the Derived Air Concentration. Then divide appropriate Derived Air Concentration by the concentration determined in Step 1.

$$\frac{DAC\ (Bq/m^3)}{C_a\ (mg/m^3)} \times \frac{1000\ mg}{1g} \times \frac{1\mu Ci}{3.7 \times 10^4\ Bq} = \frac{1}{37} \times \frac{DAC}{C_a}\ (\mu Ci/g)$$

Example 3b (continued)

STEP 3. Use quadratic formula and equation for determining specific activity to calculate the enrichment which corresponds to the specific activity obtained in Step 2.

(See Step 3 in Example 4a.)

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

STEP 4. One solution will be less than zero. The other will be the enrichment which is the "dividing line" between chemical and radiological effects.

Table 2-12 show values used for f_b and f_k, intermediate results for intakes and concentrations, and resulting "dividing line" enrichments for acute and chronic exposures, respectively.

Several aspects of this derivation must be kept in mind when using this information. First, the derivation is based on standard metabolic models and, therefore, does not necessarily reflect the effects of a uranium uptake on a real person. Since individual metabolisms will not necessarily agree with the model, the enrichment at which chemical and radiological effects are equally limiting cannot really be precisely specified. Uncertainty in the relationship between enrichment and specific activity introduces additional imprecision. Consequently, exposures should be evaluated for both chemical and radiological impact for uptakes of uranium at enrichments close to the "dividing line" enrichment between chemical and radiological effects.

Next a derived chemical toxicity limit is compared to the ICRP recommended radiological limits. The derived chemical toxicity limit of 0.07 mg/m^3 is relatively close to the OSHA standard of 0.050 mg/m^3 for soluble uranium. It would be more appropriate to use the 0.050 mg/m^3 value in establishing monitoring programs to ensure regulatory compliance.

Finally, this derivation utilized radiological limits to determine limiting hazard, but proposed regulations require radiological monitoring at 10% of regulatory limits. Therefore, monitoring for radiological in addition to chemical control, must occur at lower enrichments than those previously identified.

The impact of using the OSHA standard of 0.050 mg/m^3 can be assessed by substituting that value for 0.07 mg/m^3 in the Class D line of Table 2-12, and by substituting 4.1 mg for 5.8 mg in the Class D line of Table 2-11. [The value 4.1 mg was calculated by reducing 5.8 mg by the same proportion that 0.071 mg/m^3 was reduced to get to 0.050 mg/m^3; 5.8 x (0.050/0.071) = 4.1.] When this is done, the enrichment at which

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TABLE 2-12. DETERMINATION OF "DIVIDING LINE" ENRICHMENTS AT WHICH RADIOLOGICAL EFFECTS BECOME LIMITING--CHRONIC EXPOSURE

Class	f_b	f_k	Concentration at Which Constant Exposure Results in 0.337 mg Steady State Burden	Derived Air Concentration (Bq/m^3)	Specific Activity of "Dividing Line" Enrichment	"Dividing Line" Enrichment
D	0.476	0.12	0.07 mg/m^3	20	7.6×10^{-6} Ci/g	15%
W	0.12	0.12	0.28 mg/m^3	10	9.7×10^{-7} Ci/g	1.3%
Y	0.05	0.12	0.68 mg/m^3	0.65[a]	b	--b

a. ICRP-30 lists Class Y DACs of 0.7 Bq/m^3 for U-238 and 0.6 Bq/m^3 for U-235 and U-234. This difference is the result of rounding to one significant figure. Non-rounded values for the three isotopes are all roughly 0.65 Bq.

b. The resulting specific activity is lower than that of depleted uranium. Consequently radiological effects are limiting for all chronic exposures to Class Y uranium.

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radiological effects from chronic exposures become limiting increases from 15% to 21%. There is no change for acute exposures; chemical toxicity is limiting for all Class D exposures. Table 2-13, Figures 2-7 and 2-8 illustrates the differences between the derived level of 0.07 mg/m^3 and the OSHA standard of 0.05 mg/m^3.

The impact of the requirement to perform radiological monitoring at 10% of regulatory limits can be assessed by reducing ALIs and DACs by a factor of 10, then repeating the calculations described in Examples 4a and 4b. The results of these calculations are also shown in Table 2-13.

The effects that enrichment, chemical form, and physical form have on the hazards associated with uranium are summarized in Table 2-14. The comparison of relative chemical and radiological hazards is based on a derived kidney burden resulting from an acute exposure at the "no effect" threshold. However, for transportability Class D, the effect of using the OSHA standard of 0.05 mg/m^3 is noted. The derivations used here can be applied to any limit on chemical toxicity, be it a regulatory limit or an internal exposure control limit. It should also be emphasized that radiological impact should be considered for all intakes, even for exposure situations where chemical toxicity is limiting.

2.5 Industrial Hazards

The principal industrial hazards associated with uranium are fires, hydrogen generation, generation of oxides of nitrogen, and associated mechanical hazards characteristic of heavy objects, i.e. back injuries from lifting, dropping heavy parts on feet, etc. Hydrogen fluoride (HF) and oxides of nitrogen (NOx) 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 catch fire spontaneously.

TABLE 2-13. IMPACT OF REQUIREMENT TO MONITOR AT 10% OF RADIOLOGICAL LIMIT ON THE ENRICHMENT AT WHICH RADIOLOGICAL CONCERN BECOMES LIMITING (I.E., "DIVIDING LINE" ENRICHMENT)

	Acute		Chronic	
Transportability Class	Using 100% of Radiological Limit	Using 10% of Radiological Limit	Using 100% of Radiological Limit	Using 10% of Radiological Limit
D, OSHA Std for Chemical Toxicity	(1)	51%	21%	1.6%
D, Derived Level Chemical Toxicity	(1)	38%	15%	0.8%
W	39%	3.6%	1.3%	(2)
Y	(2)	(2)	(2)	(2)

(1) Chemical toxicity concerns are limiting at all enrichments.

(2) Radiological effects are limiting at all enrichments.

Figures 2-7 and 2-8 represent the information presented in Table 2-13 for acute and chronic exposures, respectively.

TABLE 2-14. EFFECT OF MATERIAL CHARACTERISTICS ON RELATIVE HAZARDS

Characteristic	Material	Relative External Hazard	Relative Internal Hazard
I. Enrichment	A. Depleted, Natural, and Slightly Enriched Uranium	1. Little penetrating radiation 2. Moderate non-penetrating radiation from bare uranium	1. Inhalation results in potential chemical toxicity, and/or radiation doses to bone surfaces, lungs, and effective dose equivalents.
	B. Moderate to Highly Enriched Uranium	1. Significance of penetrating radiation increases with enrichment 2. Non-penetration radiation from bare uranium remains moderate	1. Same as above except that higher enrichment means higher doses per unit mass inhaled. For chronic exposures radiation doses become more significant and chemical toxicity less significant as enrichment increases.
II. Chemical Form	A. Transportability Class D (Soluble)	1. Chemical form does not significantly affect external hazard except that neutron radiation levels increase in homogeneous floride compounds	1. For acute exposure to any enrichment, chemical toxicity is more limiting than radiological Annual Limits on Intake. 2. For chronic exposures, chemical toxicity is more limiting up to 15% enrichment (21% of OSHA standard for chemical toxicity is used); then the nonstochastic dose limit for bone surfaces becomes limiting.
	B. Transportability Class W	1. Chemical form does not significantly affect external hazard except for floride compounds/mixtures	1. For acute exposures, chemical toxicity is more limiting up to 39% enrichment; above that level, the effective dose equivalent becomes limiting. 2. For chronic exposures, chemical toxicity is more limiting up to 1.3% enrichment; above that level, the effective dose equivalent becomes limiting.
	C. Transportability Class Y (Insoluble)	1. Chemical form does not significantly affect external hazard except for floride compounds/mixtures	1. Chronic and acute inhalation are limited by effective dose equivalent resulting primarily from lung dose.
III. Physical Form	A. Nondispersible	1. Non-penetrating dose rates are slightly higher from metal than from compounds. 2. Penetrating dose rates are not significantly affected by physical form.	1. Nondispersible forms generally don't pose an internal hazard.
	B. Dispersible	1. Non-penetrating dose rates are slightly higher from metal than from compounds 2. Penetrating dose rates are not significantly affected by physical form.	1. Dispersible forms pose internal hazards as described under "I. Enrichment" and "II. Chemical Form."

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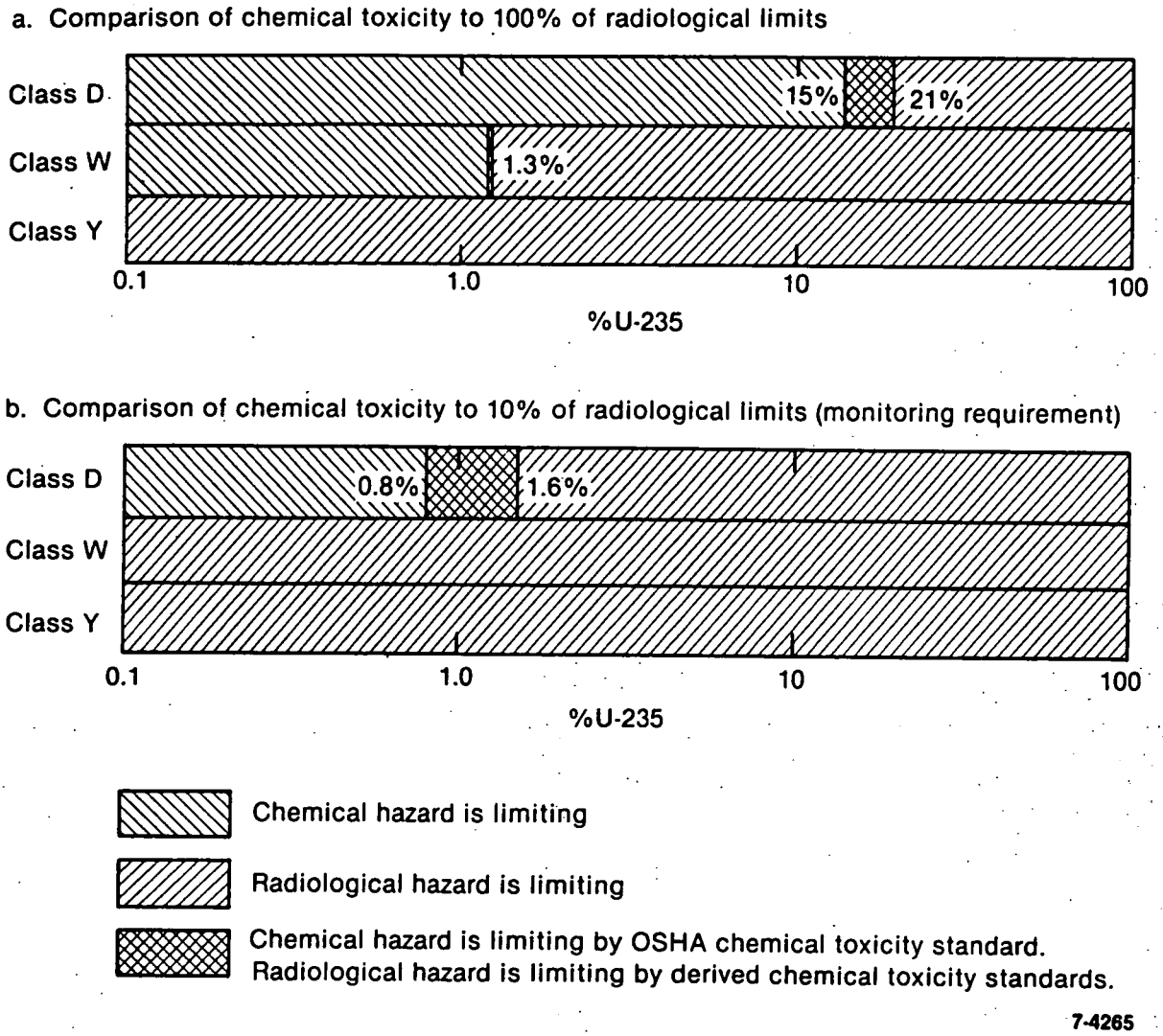

Figure 2-7. Effect of enrichment on limiting hazard--chronic exposure.

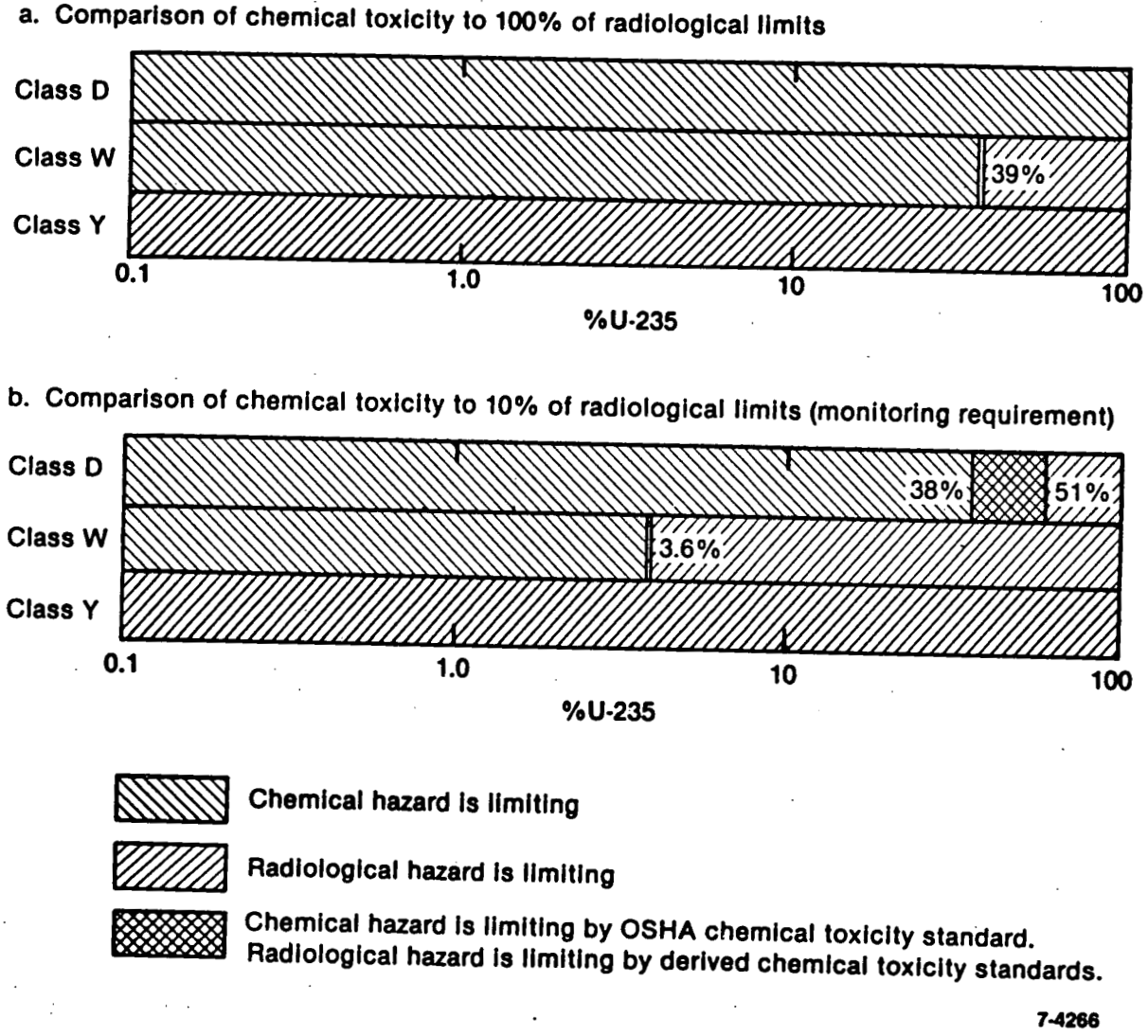

Figure 2-8. Effect of enrichment on limiting hazard--acute exposure.

2.5.1 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_2O 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. In general, individuals can smell HF at levels of .02-.2 mg/m3 versus the TLV of 2.5 mg/m3. The TLV was set based primarily on the irritation of eyes and mucous passages rather than on permanent damage. No person can tolerate an airborne concentration of 10 mg/m3; personnel will evacuate the area if they are able to do so. Exposure for as little as 15 min. to an airborne concentration of 20-30 mg/m3 may prove fatal (pulmonary edema).

2.5.2 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 which cause irritation to eyes, mucous membranes, and skin. Significant inhalation can result in pulmonary edema. The ACGIH TLV-TWA and TLV-STEL for nitric acid is 2 ppm and 4 ppm respectively.

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

2.5.3 Hydrogen Gas

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

2.5.4 Fire

Finely divided uranium metal is pyrophoric, capable of igniting spontaneously. This type of material should be handled and stored to minimize 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 extinguishes 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.

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SECTION 3

DOE FACILITIES, PROCESSES AND EXPERIENCES

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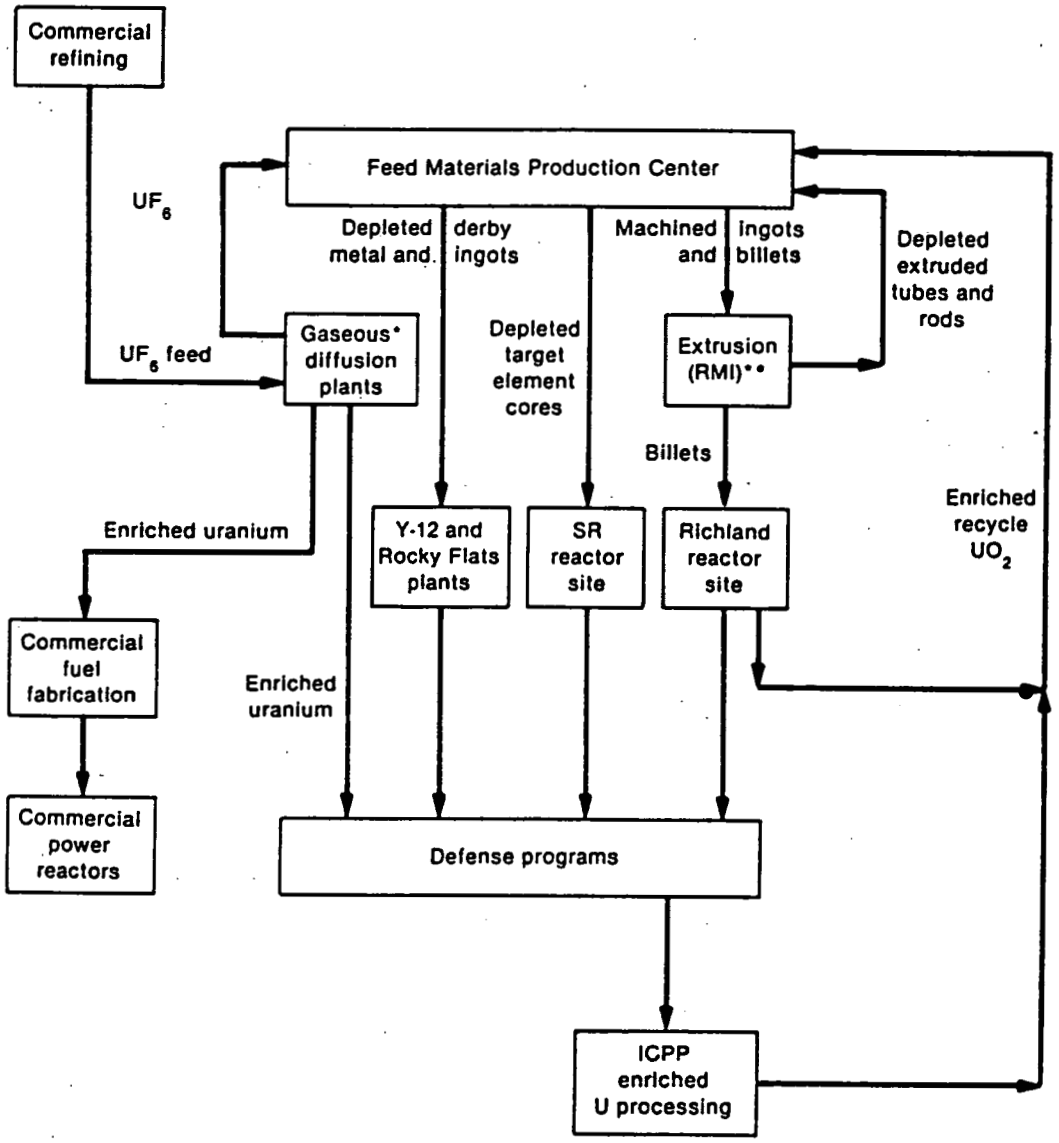
SECTION 3

DOE FACILITIES, PROCESSES AND EXPERIENCES

DOE facilities encompass a large variety of processes utilizing uranium in many chemical and physical forms. The attendant health hazards, as discussed in Section 2, vary with the uranium enrichment, presence of uranium decay products, recycled uranium (RU) contaminants (such as fission products), chemical composition, and physical characteristics (e.g., particle size).

This section provides a generic discussion of the major processes and uranium materials found in DOE facilities, as well as observations based on operating experience regarding potential hazards. Since many of the operations and processes are classified, the discussion will not refer to any specific facility. Figure 3-1 is a flow diagram outlining the role of various DOE facilities in programs administered by the Energy Assistant Secretary for Defense Programs. Figure 3-2 is a flow diagram of the nuclear fuel cycle within which DOE operates the uranium enrichment plants. Uranium enrichment is one step in the uranium fuel cycle involving the partial enrichment of the uranium-235 (U-235) isotope in naturally occurring uranium to obtain a product with increased fissionable uranium-235 content. Many steps in this cycle require chemical processes to change the characteristics of the uranium fuel. The enrichment step, however, involves only physical separation of isotopes, i.e., the uranium is in the chemical form uranium hexafluoride (UF_6) when it enters the separation equipment and remains UF_6 throughout processing.



* Oak Ridge, Tenn; Paducah, Ky; and Piketon, Ohio

** RMI Company, Ashtabula, Ohio

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Figure 3-1. Role of DOE Facilities in DOD administered programs.

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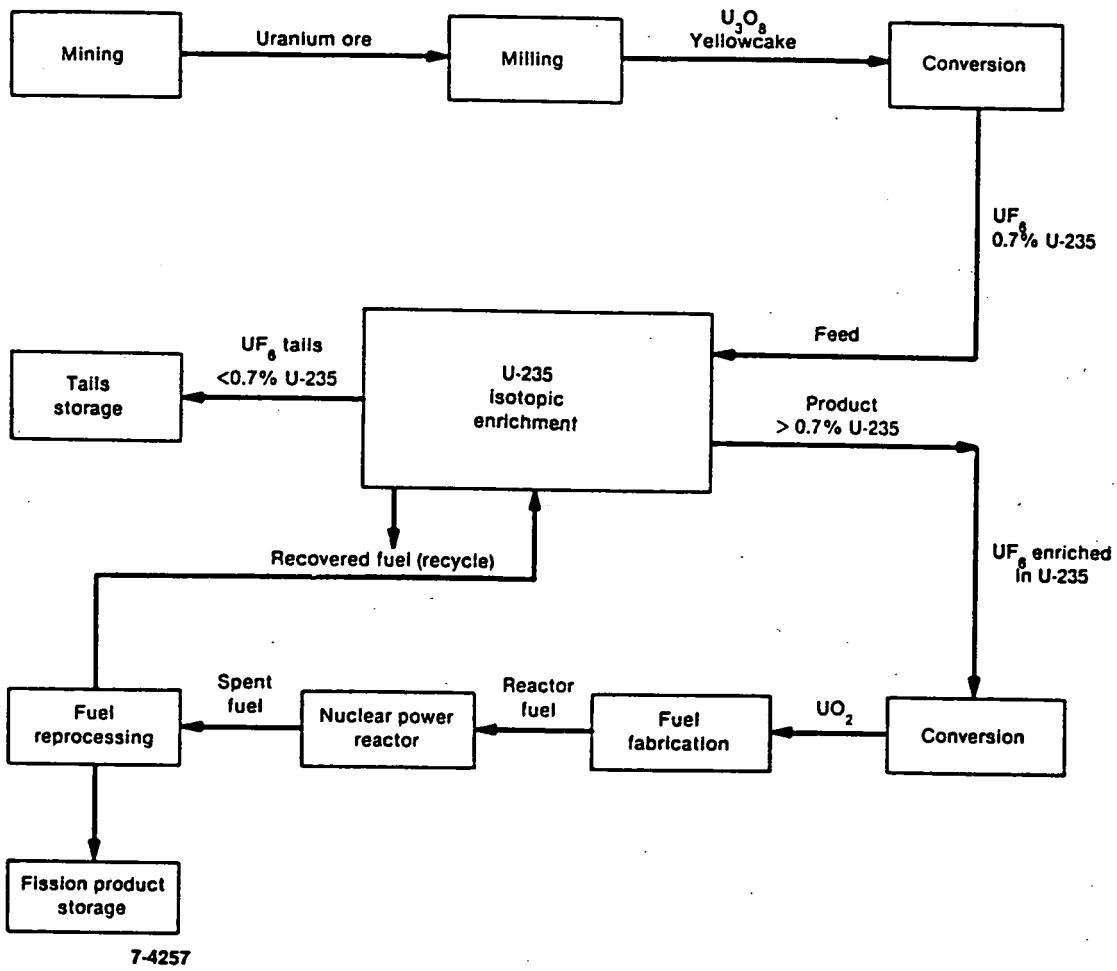

Figure 3-2. Processing scheme for nuclear power reactor fuel.

3.1 Facilities and Processes

3.1.1 Gaseous Diffusion

Process Description

DOE involvement in the nuclear fuel cycle generally begins with uranium enrichment operations and facilities. There are two possible feed streams to the uranium enrichment step (see Figure 3-2). One feed stream is "virgin" coming from uranium ore. The second feed stream has been through the enrichment-conversion-fuel-fabrication-nuclear reactor fuel reprocessing chain prior to being returned to the enrichment step. This second feed or "recycle" will contain trace amounts of fission products and transuranics that were formed in the nuclear reactor and not completely removed in the fuel reprocessing step. The uranium element appears in nature in three isotopes having atomic weights of 238, 235 and 234. The 235 and 234 isotopes are fissionable and capable of sustaining a critical reaction. Natural uranium contains 0.7 percent U-235 isotope. The percentage of U-235 in the uranium is increased by isotopic separation currently utilizing the gaseous diffusion process.

Three basic requirements must be met to apply the gaseous diffusion process. These are a stable process gas (UF_6), a porous membrane, and a driving force to cause selective diffusion of the molecules through the porous membrane.

The isotopic separation is accomplished by diffusing uranium, which has been combined with fluorine to form uranium hexafluoride gas (UF_6), through a porous membrane (barrier) and utilizing the different molecular velocities of the two isotopes to achieve separation. The uranium-235 enrichment through each stage is so minute that literally thousands of stages are required to increase the assay from 0.7 percent to the desired assay.

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Uranium hexafluoride (UF_6) has many advantages for use in the separation of uranium isotopes. It is one of a few stable uranium compounds with an appreciable vapor pressure at moderate temperature. Furthermore, the fluorine atoms in the molecule have only one natural isotopic weight and the difference in masses of the U-235 F_6 and U-238 F_6 molecules is due entirely to the uranium isotopes. Thus, the fractionation by molecular weight separates only uranium isotopes. Table 3-1 lists the characteristics of UF_6. Figure 3-3 shows the UF_6 phase diagram. Health hazards associated with UF_6 and other uranium compounds are described in later sections. The disadvantages of uranium hexafluoride are that UF_6 is very toxic, corrosive and reactive chemically, and its use necessitates special materials of construction and special operating techniques, and places limitations on the operating temperatures and pressures which are used. Although equipment is treated prior to installation, UF_6 reacts with the interior equipment surfaces and barrier. This reaction produces uranium compounds less soluble than UF_6 (e.g., UF_4 and uranium-metal complexes) and provides a mechanism for the ingrowth of uranium decay products.

The gaseous diffusion process uses porous tubes (barriers) to achieve separation. To ensure diffusive flow, a uniform pore size of less than two-millionths of an inch diameter must be maintained. The pore size for diffusive flow is so small that literally acres of barrier surface are required in a large production plant. The amount of barrier surface in each stage, or the number of porous tubes, depends on the required plant capacity. By cascading or connecting the basic separation stages in series, the desired level of enrichment can be achieved. Because the separative capability per stage is so small, the exact number (always large) of stages required is determined by the enrichment needed.

TABLE 3-1. CHARACTERISTICS OF URANIUM HEXAFLUORIDE (UF_6)

Characteristics	Value
Molecular Weight-- $^{235}UF_6$	349.03
$^{238}UF_6$	352.04
Sublimation Point at 14.7 psia	133.8°F (56.6°C)
Vapor Pressure at Melting Point (147.3°F, 64.1°C)	22.0 psia (152 x 10^3Pa)
Density of Solid at 68°F (20°C)	317.8 lb/ft^3 (5091 kg/m^3)
Density of Liquid at 235°F (112.8°C)	207.3 lb/ft^3 (3320 kg/m^3)
Density of Liquid at 250°F (121.1°C)	202.9 lb/ft^3 (3250 kg/m^3)
Critical Temperature	446.6°F (230.2°C)
Triple Point Data:	
Temperature	147.3°F (64.1°C)
Pressure	22 psia (152 x 10^3 Pa)
Density of Liquid	227.7 lb/ft^3 (3647 kg/m^3)
Heat of Sublimation at 147.3°F (64°C)	58.2 Btu/lb (135 x 10^3 J/kg)
Heat of Fusion at 147.3°F (64°C)	23.5 Btu/lb (54.7 x 10^3 J/kg)
Heat of Vaporization at 147.3°F (64°C)	35.1 Btu/lb (81.6 x 10^3 J/kg)
Thermal Conductivity:	
41°F, Solid (5°C)	3.70 x 10^{-3} Btu/hr•ft•°F (6.40 x 10^{-3} W/m•°K)
162°F, Liquid (72.2°C)	9.27 x 10^{-2} Btu/hr•ft•°F (16.04 x 10^{-2} W/m•°K)
Heat of Reaction for UF_6 and H_2O at 77°F (25°C)	137 Btu/lb of UF_6 (26.8 kcal/g mole)
Heat Capacity of Reaction Products at Room Temperature	
UO_2F_2	0.0821 Btu/lb•°F (343.8 J/kg•°K)
HF	0.348 Btu/lb•°F) (1460 J/kg•°K)

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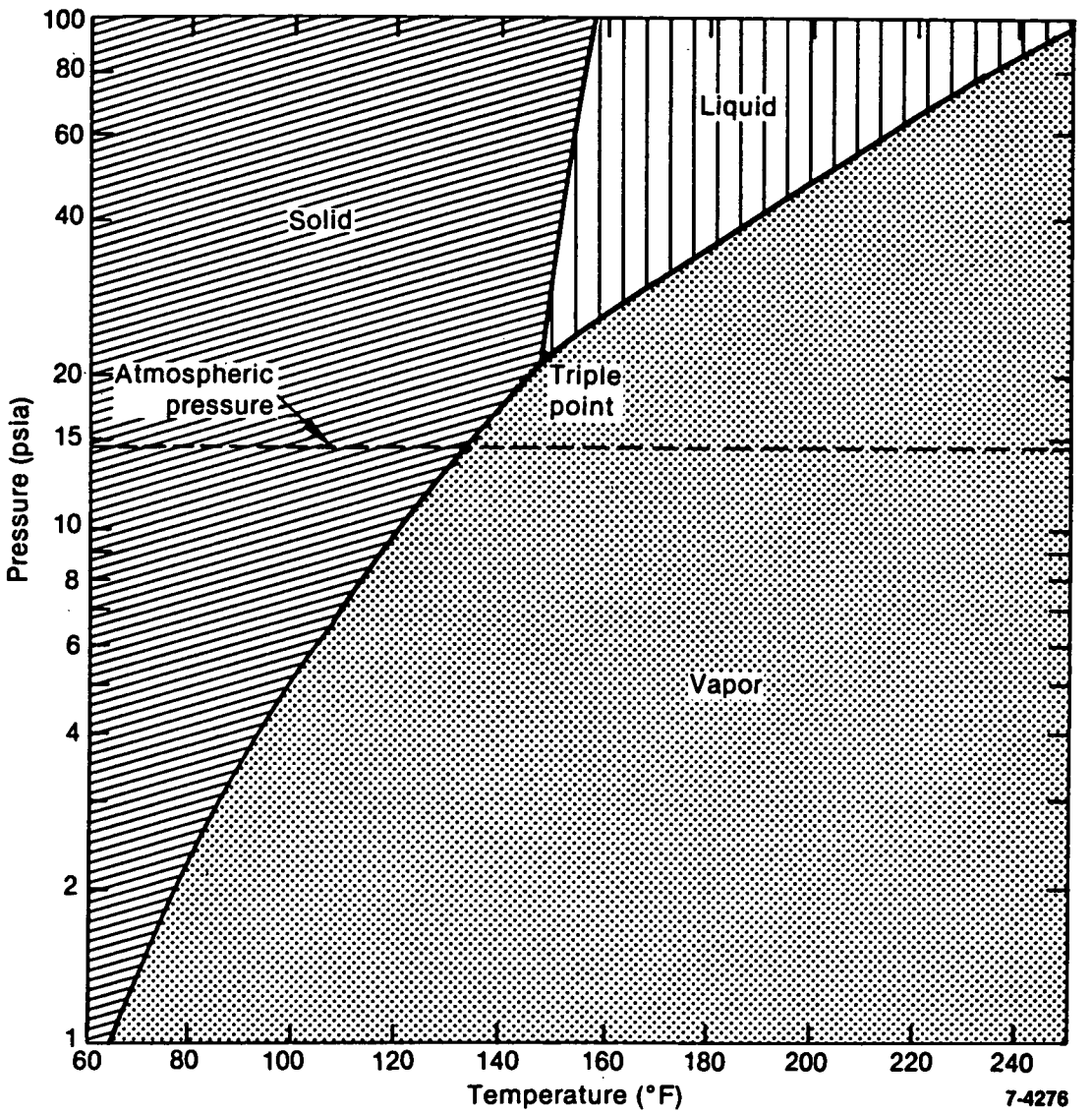

Figure 3-3. UF_6 phase diagram.

A stage consists of a motor, compressor, and a converter (contains the barrier and the cooler). The UF_6 in a single stage (Figure 3-4), is introduced as a gas and made to flow along the inside of the barrier tube. About one-half of the gas diffuses through the barrier and is fed to the next higher stage; the remaining, undiffused portion is recycled to the next lower stage. The diffused stream is slightly enriched with respect to U-235, and the stream that has not been diffused is similarly depleted. Figure 3-5 shows how the single stages are series connected or cascaded to accomplish significant separations. It also shows the essential equipment components required for the process. In this case, axial-flow compressors, driven by electric motors, are used to move the process gas through the diffuser (or converter) that contains the barrier. Stage coolers are required to remove the heat of compression.

In a theoretical cascade, each stage would be slightly different from the stages immediately above or below. The converters that contain the largest barrier area would be located at the normal assay (0.72% U-235) feed point. Stages above the feed point would be progressively smaller and are referred to as the enriching section. The stages below the feed location, also having progressively smaller stages, are called the stripping section. The number of stages required in each section is determined by operating parameters and by the total enrichment and depletion planned. Fuel for light water reactors can be produced by diffusion cascades having only two, or possibly three equipment sizes, while a longer cascade, such as the Portsmouth GDP, is designed to produce a more highly enriched product for special reactors and, therefore, is made up of five equipment sizes (illustrated in Figure 3-6). As seen in this figure, two feed streams supply the cascade. Two high-assay streams, two commercial reactor assay streams (2.7 and 4.0% U-235), and the tails stream are withdrawn. Average high-side pressures and assays are given on either side of the schematic.

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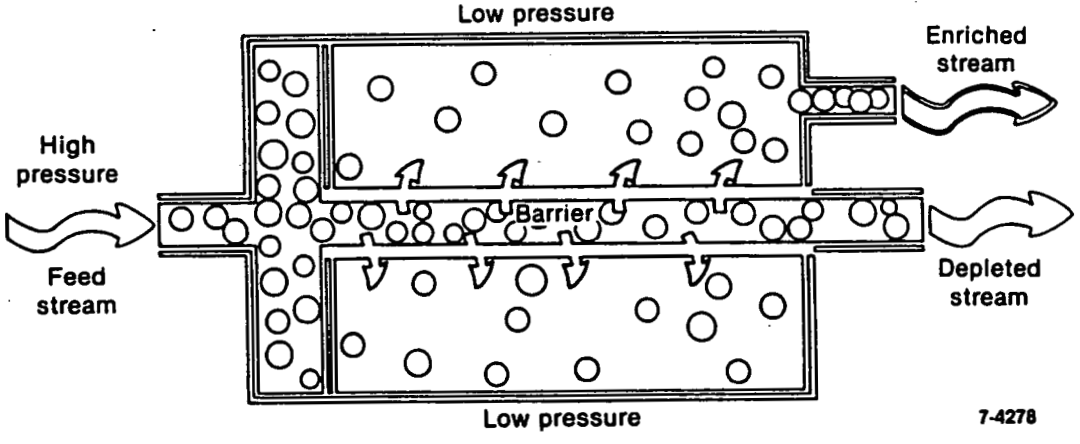

Figure 3-4. Gaseous diffusion stage.

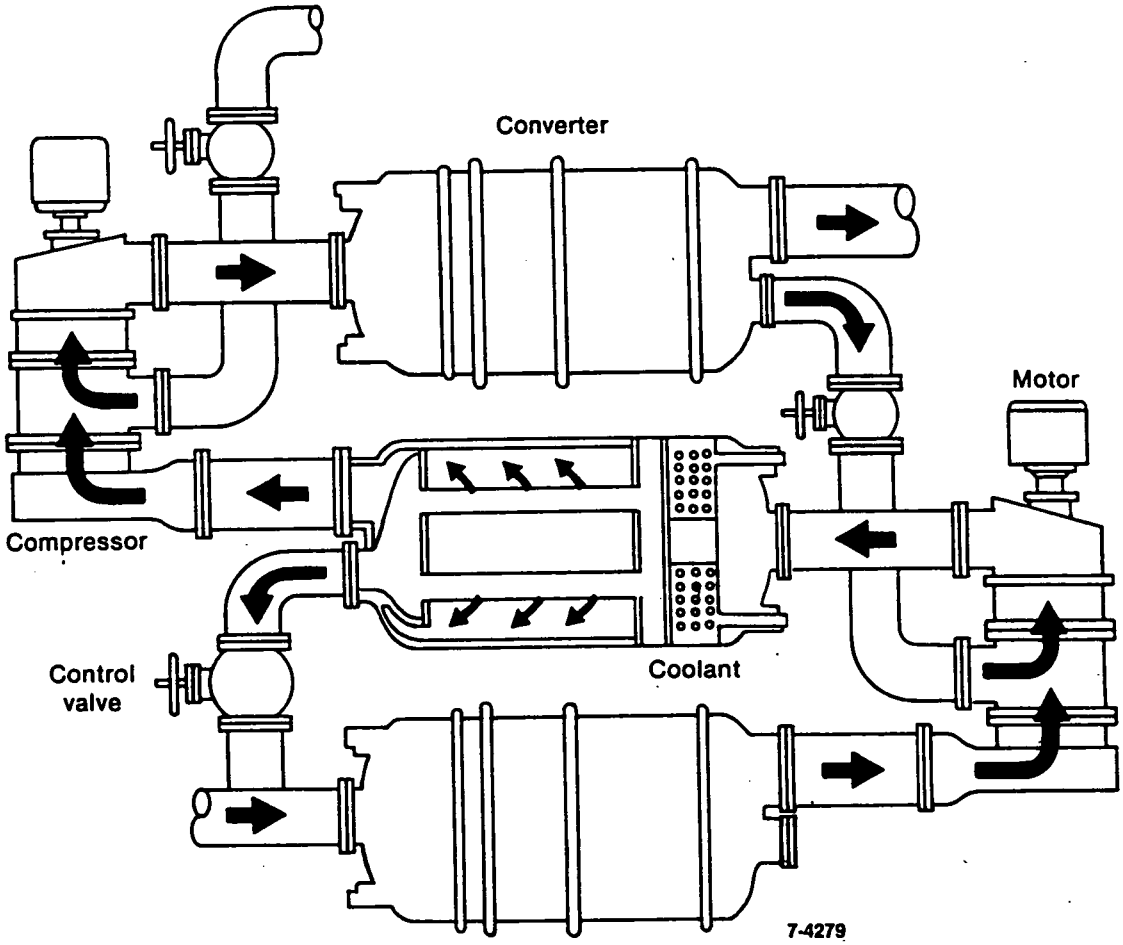

Figure 3-5. Stage arrangements.

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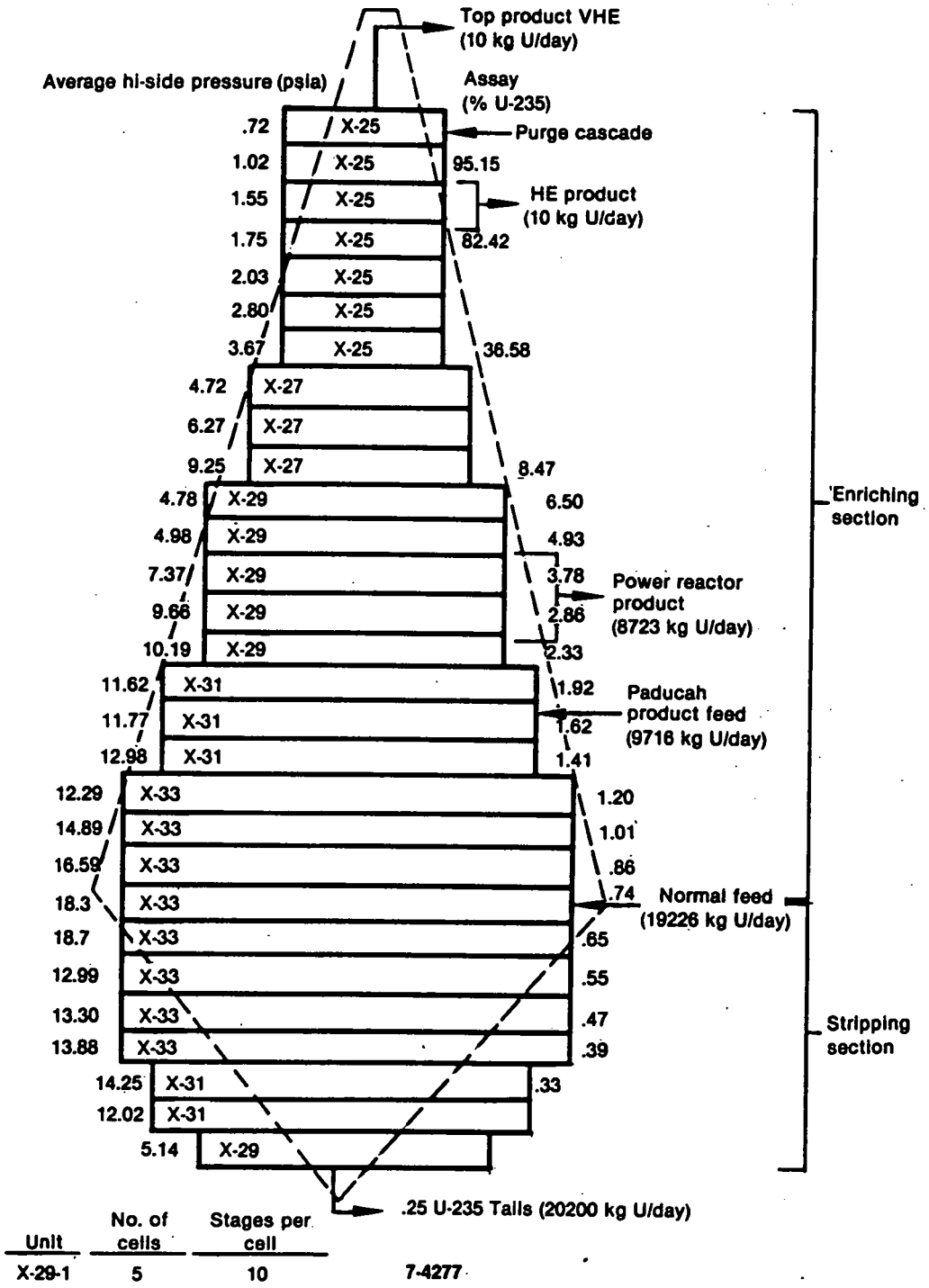

Figure 3-6. Portsmouth Gaseous Diffusion Plant cascade flow, pressure, and assays at 2200 MW.

Hazards Discussion

The primary hazard in a GDP is acute exposure of operating personnel from a major release of UF_6 from the process equipment. Low-level/chronic exposure is possible in certain phases of the process, such as in feed and withdrawal areas and during process maintenance activities. For low enrichments, the chemical toxicity remains the controlling chronic hazard; at higher enrichments, radiotoxicity becomes the controlling hazard, primarily due to the bone dose. Most of the chemical compounds encountered in gaseous diffusion plants are Class "D" (primarily uranium hexafluoride and uranyl fluoride). However, due to the reaction of UF_6 with the internal equipment surface, some class "W" compounds are present on the internal surfaces of the process equipment with their associated uranium decay products. In addition, contaminants (previously enumerated and discussed) introduced with the UF_6 feed will be present in various concentrations. In general, any contaminant of higher molecular weight than uranium (e.g., transuranics) will concentrate at or slightly below the UF_6 feed point. Those contaminants with lower molecular weight than uranium (e.g., technetium) will travel up the diffusion cascade and concentrate at the UF_6 "front" (the break point between the UF_6 and low density gases such as O_2 and N_2). While the lighter contaminants will concentrate at the higher enrichment section of the gaseous diffusion process, some of the material will deposit on internal surfaces at much lower concentrations throughout the process system. In general, the presence of uranium decay products or fission products pose no significant hazard while confined within the process equipment. The primary hazard occurs when the equipment is opened and removed for maintenance or replacement.

Due to the feeding of recycled uranium in previous years, trace quantities of fission products and transuranics are present in the diffusion plants. In the high enrichment section of the diffusion process, the fission product Tc-99 concentrates to significant levels requiring special controls during equipment removal and maintenance. The feed and

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withdrawal operations concentrate non-volatile uranium decay products in cylinder residues (heels) or in liquid UF_6 transfer lines. In the diffusion process, current levels of transuranics do not represent a controlling hazard in the uranium systems. However, they occasionally present some control problems in uranium recovery operations; these processes will be considered later.

Operations utilizing uranium hexafluoride will not generally present a hazard from penetrating radiation except in the handling of empty cylinders and in the maintenance and decontamination of process equipment. However, internal exposures may be of significant concern both from the standpoint of acute and chronic exposures. However, UF_6 has excellent visual and olfactory warning properties, hence, chronic exposures will be detected and controlled at levels far below chemical or radiological toxicity limits, by workers being aware of any significant releases of UF_6. This general principle is true except for "very highly enriched" (VHE) uranium hexafluoride (>90% enrichment). It is important to be cognizant that accidental releases of large quantities of UF_6 represent a major acute exposure concern due to its extreme toxicity. Several fatalities associated with uranium processing have occurred when individuals were exposed to high concentrations of UF_6 and its reaction products during accidental releases. (These fatalities occurred during the Manhattan Project and one recently at Gore, Oklahoma.) As a result, any operation with a potential for a large release of UF_6 (primarily liquid handling or transfer operations) should have secondary containment, if possible, as well as personnel escape equipment. Evacuation routes should be established consistent with Life Safety Code requirements. The probability of releasing UF_6 in the solid phase is extremely low. Any losses would be due to sublimation to the gaseous state with gas escaping through the opening or rupture. The liquid releases can occur at the feed, withdrawal, cylinder sampling, and cylinder-to-cylinder transfer facilities. These releases can occur at pigtail connectors, manifolds, cylinder valves, and as the result of dropping cylinders filled with liquid UF_6 from lifting cranes. Releases of UF_6 from cylinders transported between facilities

are negligible since the contents are solidified before the cylinders are transported. Finally, it is important to routinely analyze UF_6 process steams from each major source to identify the presence of contaminants, both radiological and otherwise (e.g., organics and "light" gases), which could pose health and/or safety hazards. These contaminants must be evaluated with respect to their behavior in the processes involved to determine if they will concentrate in any areas to the extent that their hazard relative to uranium will change significantly and perhaps become the controlling factor. For example, in the gaseous diffusion process, technetium compounds are concentrated in the high enrichment sections of the diffusion cascade and could become the controlling hazard.

Experience

As previously discussed the primary concern of the gaseous diffusion plant is atmospheric releases of UF_6, HF, and UO_2F_2. The probability of massive releases is greater when UF_6 is in the liquid state and smaller while UF_6 is in the gaseous state. This is because the release rate from a containment breach will be much higher in a liquid release and the UF_6 will flash to a gaseous state very quickly at ambient temperatures and pressures. In systems utilizing gaseous UF_6, the gas pressures rarely exceed 30 psia and usually are subatmospheric, and the amount of material in the equipment is thus relatively small. Contamination spread from a breach of a vessel containing solid UF_6 will generally be limited to the immediate area, as a result of the low vapor pressure of the solid UF_6 mass. In fact, the release of uranium is due to the reaction of moist air with the surface of the solid UF_6, or sublimation to the gaseous state or both.

There have been 21 major releases of UF_6 at three gaseous diffusion plants during the 18-year period between 1961 and 1978. None of these releases resulted in serious injury to personnel on or off plant site. It is estimated that the three sites performed more than 122,000 operations involving pigtail connections on UF_6 cylinders and more than

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426,000 cylinder lifts with cranes, forklifts, and straddle carriers. Approximately 135,000 of these lifts included cylinders containing liquid UF_6, the phase with the greatest release potential.

Criticality

Because of the range of U-235 enrichment involved, criticality safety is of major concern in some areas such as the Portsmouth GDP. The analyses of criticality potential at this plant under current plant design and operation show that the risks associated with inadvertent criticality are low to extremely low. No single failure has been identified that could result in nuclear criticality. A double-contingency policy is followed in the design and operation of all processes that could involve fissionable material.

The maximum uranium enrichment produced and stored at the Portsmouth GDP is 98% U-235. The bulk of the uranium inventory within the process equipment is in the form of gaseous UF_6, a phase which will not sustain a critical reaction at any mass, geometry, or enrichment level. Other phases of UF_6, i.e., liquid or solid, can become critical. The probability exists for UF_6 to solidify within the cascade due either to abnormally low temperature conditions, or to wet air entering the cascade and forming solid UO_2F_2. However, an inventory shift would be necessary to accumulate enough enriched material in a condition to cause a criticality.

Preventative Measures

To reduce the probability of a criticality, temperatures and pressures are maintained at values to prevent solidification of UF_6. Furthermore, the cascade is maintained in a leak-tight condition, preventing the inleakage of wet air. A radiation monitoring program throughout the cascade would locate any accumulation of solid uranium masses. In plant areas where uranium solutions or compounds are processed or stored, various criticality controls are employed, including geometry and batch control,

uranium concentration limitations, and other administrative controls. All processes are analyzed by the Nuclear Criticality Safety Staff before approval. Independent safety reviews are conducted on a periodic basis.

Although considered to be of low or extremely low probability, analyses of possible criticality incidents revealed that little hazard exists to personnel except those in the immediate vicinity of the incident. Rapid evacuation is effective in minimizing exposures. Criticality alarms are installed in facilities containing radioactive material. Although alarms will not prevent an incident, employees will be alerted to the need to evacuate and/or not to enter the facility. The radiation alarm system is designated a safety system for all areas of the plant.

3.1.2 Atomic Vapor Laser Isotope Separation (AVLIS) Process

AVLIS Process Description

The AVLIS process is the newest method of uranium separation and enrichment. Although no DOE production plants exist which utilize this technology, it will be briefly described.

In this process, metallic uranium is vaporized, selectively photoionized, and separated to produce an enriched product stream and a depleted tails stream. Figure 3-7 is a simplified schematic diagram of this process.

Uranium metal is introduced into a large vacuum vessel where it is melted and vaporized by large electron beam guns. (The interaction of these electrons with the uranium melt produce copious quantities of X-rays.) Uranium atoms in this vapor stream are selectively ionized by laser light. The "product" ions are extracted electromagnetically on product collectors, while the "tails" stream passes through to be collected on a tails collector.

Atomic Vapor Laser Isotope Separation (AVLIS) process. Metallic uranium is melted and vaporized. The vapor is illuminated by visible laser light, which photoionizes the selected isotope. The ions are then electromagnetically extracted. Inset shows the details of the separation process.

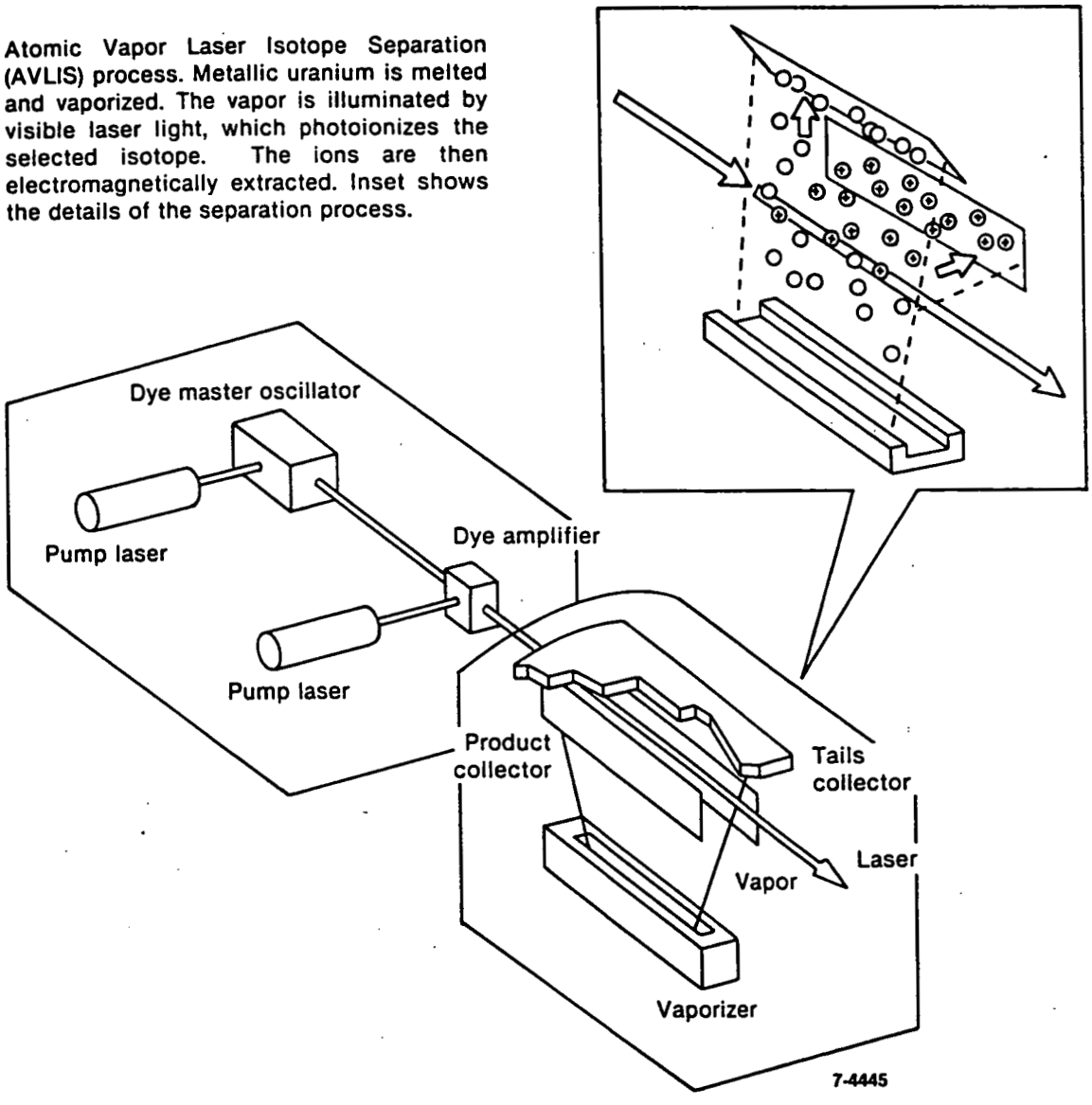

Figure 3-7. Atomic vapor laser isotope separation (AVLIS) process.

Hazards Discussion

The most serious potential health physics problem associated with this process is that of inadvertent exposure to the extremely high X-ray fields generated within the vacuum vessel. While the vessel walls provide complete shielding against these X-rays, any penetrations (e.g., instrument feedthroughs) are potential problem areas. For a more detailed discussion, see article by M. S. Singh referenced in the Bibliography.

Uranium hazards associated with this process are those associated with uranium metals and uranium oxides which are discussed in other process descriptions. It should be noted that since the U-234 does not follow along with the U-235 (as it does in conventional enrichment processes) the isotopic mix of AVLIS enriched uranium is not dominated by U-234 activity. See discussion in Section 2 of this manual (2.1.1 Radiological Properties) for comparative discussion and descriptive figures.

3.1.3 Uranium Conversion Processes/Uranium Refining

Conversion Processes

In addition to the uranium enrichment processes, much of the other DOE activities involves some type of chemical conversion of uranium compounds to produce materials to meet a specific need and subsequent physical modification of these materials. This is demonstrated by the schematic of the FMPC (Feed Materials Production Center) Facility processes shown in Figure 3-8. The chemical conversion may involve oxidation or reduction of uranium compounds, primarily uranyl nitrates, fluorides, or oxides. A brief list of uranium conversion reactions is shown in Figure 3-9.

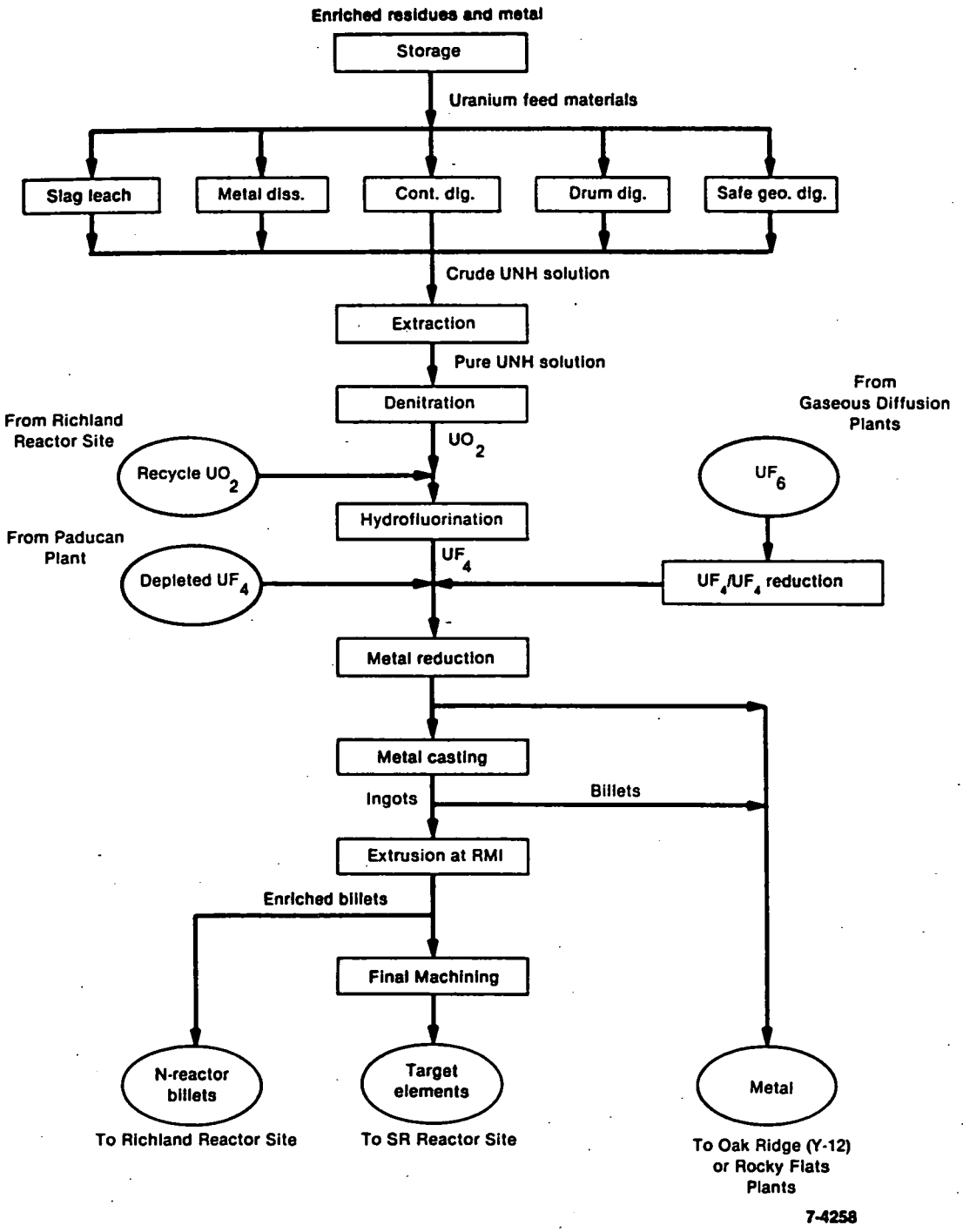

Figure 3-8. Schematic diagram of the FMPC process.

Uranium Conversion Formula

$$UO_2(NO_3)_2 \cdot xH_2O \rightarrow UO_3 + NO + NO_2 + O_2 + xH_2O$$

$$UO_3 + 2HF \rightarrow UO_2F_2 + H_2O$$

$$UO_2F_2 + 2F_2 \rightarrow UF_6 + O_2$$

$$UO_3 + H_2 \rightarrow UO_2 + H_2O$$

$$UO_2 + 4HF \rightarrow UF_4 + 2H_2O$$

$$UF_4 + 2Mg \rightarrow U + 2MgF_2$$

Figure 3-9. Uranium conversion formula.

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Hazards

Chemical conversion processes tends to concentrate uranium decay products in the waste streams. So as a general rule, fission products or transuranics will also tend to concentrate in the process waste though exceptions exist. The chemical characteristics of these contaminants will cause significant exposure levels of beta and gamma radiation from the uranium decay product activity in certain sections of the process. In some cases, the radiological hazard of transuranics and/or fission products may increase to a significant level of concern. These processes and hazards are discussed in the following sections.

From a chemical toxicity standpoint, the Class D transportable (soluble) compounds of uranium (e.g., UF_6, UO_2F_2, $UO_2(NO_3)_2$) present the greatest concern. However, with insoluble uranium compounds (uranium metal, uranium oxides, etc.) the dominant hazard is normally the radiotoxicity. In areas where enriched uranium is processed, criticality safety is a major concern.

Refining Process

This section explains some of the basic features of the uranium refining/conversion process as applicable to DOE operations. Two basic types of refining processes utilized to produce UF_6 from the uranium ore concentrates (yellowcake) are shown in Figure 3-10. The processes consists of four basic steps: sampling, feed preparation, conversion, and purification. In the "dry" process, the uranium is first converted to UF_6 and then purified by fractional distillation; this process is practiced in the United States at the Allied Chemical Uranium Refinery. In DOE facilities, some version of the "wet" process is employed; in this process, the uranium is first purified by solvent extraction and then converted to UF_6 or other compounds. A more detailed schematic of this process is shown in Figure 3-11. The various steps in the "wet" uranium refining process will now be considered.

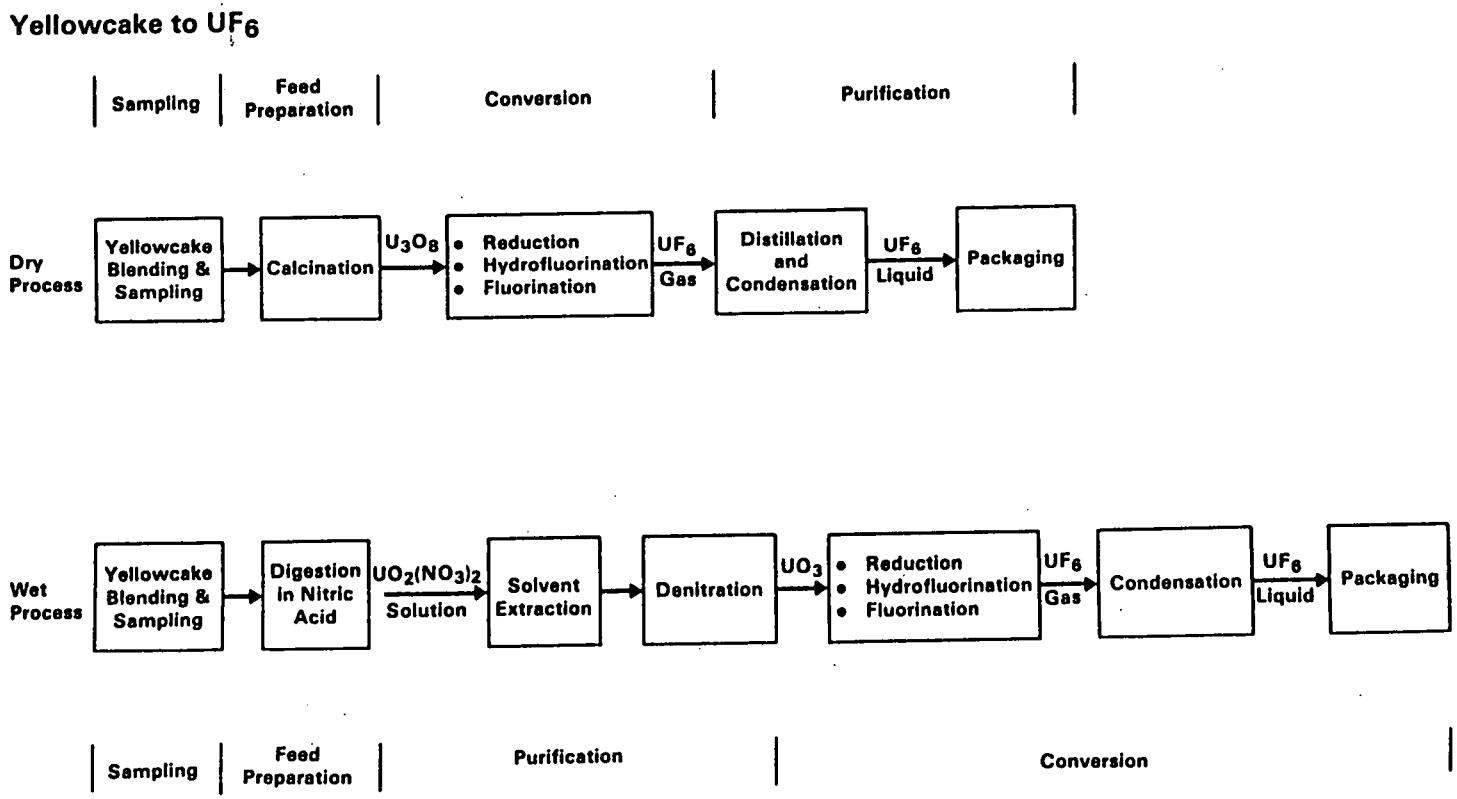

Figure 3-10. Yellowcake to UF_6.

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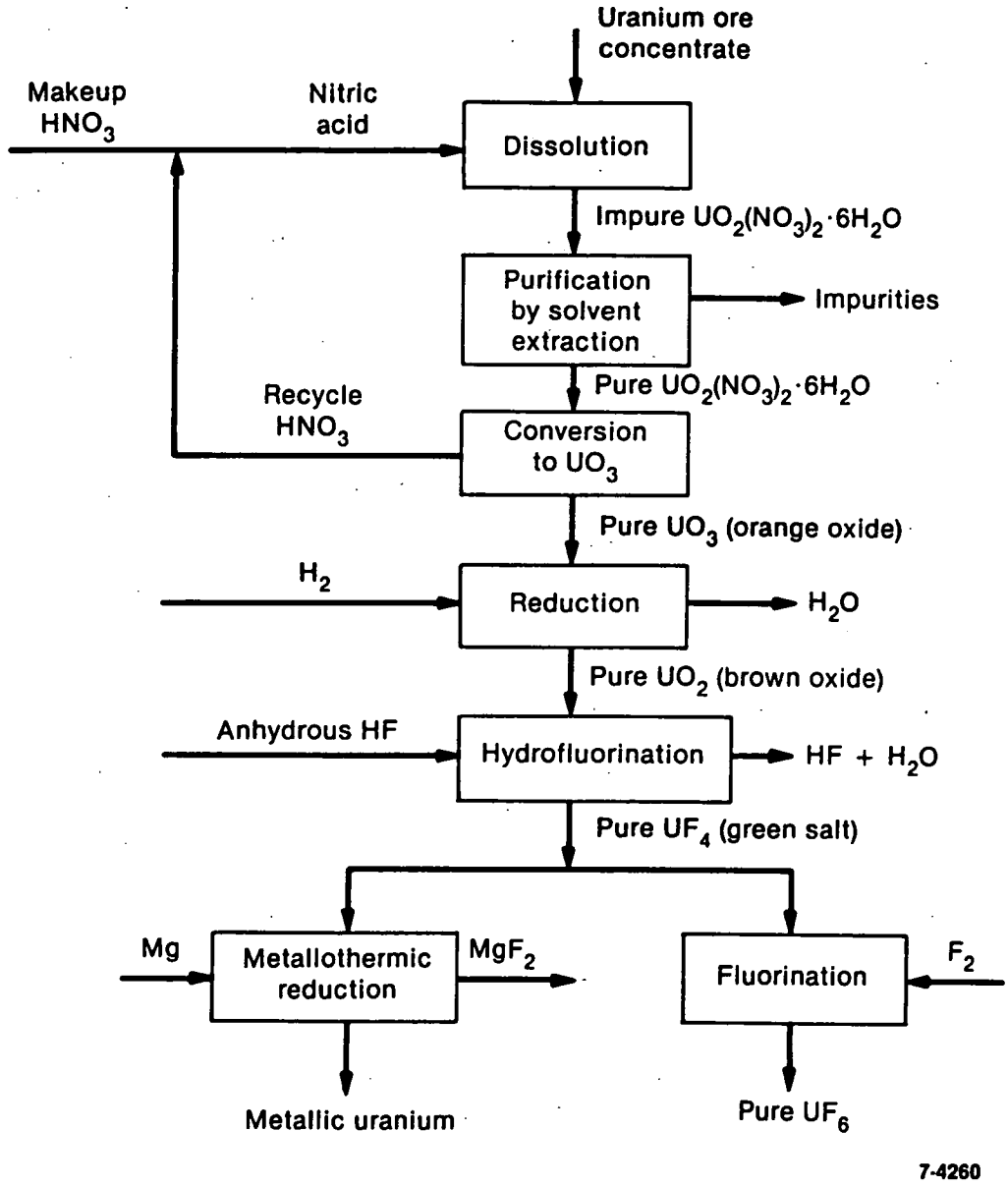

Figure 3-11. Steps in conventional uranium refining processes.

Feed Sampling and Preparation

Process. For the most part, DOE facilities do not receive uranium ore concentrates from commercial sources. However, uranyl nitrate solutions are generated through processing of spent uranium fuel, uranium contaminated waste decontamination solutions and classified uranium material.

Hazard. It is important that in any process in which feed is prepared for uranium purification/refining that the materials involved be analyzed to determine the concentration and enrichment of uranium and also to determine the presence of uranium decay products, fission products and/or transuranics. From a process standpoint, it is also important to establish the concentration of other, non-radioactive elements (e.g., metals, fluorides, chlorides, etc.) since these may affect the efficiency of the solvent extraction process. Dissolution of the feed material with nitric acid produces an aqueous solution of uranyl nitrate hexahydrate ($UO_2(NO_3)_2\ 6H_2O$) containing excess nitric acid and variable amounts of nitrates of metallic impurities and other radionuclides present in the feed.

The hazards associated with the sampling of the feed material and subsequent dissolution by nitric acid are dependent on the material source. Obviously, feed stream sampling from spent uranium fuel presents health protection challenges due to the presence of fission products and transuranics. Much of the process must be isolated from the worker by using remote controls, system containment, and careful control of waste and product streams. The sampling and dissolution of uranium metals, compounds and uranium contaminated wastes may pose both external and internal exposure hazards due to the presence of uranium decay products and/or the potential for generating airborne contamination during solids handling operations. Often, beta radiation levels from unshielded nitric acid solutions may range from a few mrad/hr to hundreds of mrad/hr. In general, they may be effectively shielded with low Z materials (i.e., aluminum), by increasing the distance between personnel and open containers of solution and by decreasing personal handling of solution containers.

Purification by Solvent Extraction

Process. Uranyl nitrate has the unusual property, shared only by nitrates of a few other actinides, of being very soluble in a number of organic solvents. When such an organic solvent is immiscible with water, it can be used in a solvent extraction process to separate uranium as uranyl nitrate, from aqueous solution, thereby separating it from associated impurities. Usually DOE processes use tributyl phosphate (TBP) dissolved in a hydrocarbon diluent (highly refined kerosene, normal hexane). The aqueous feed to the extraction equipment is highly acidic, roughly 1 N nitric acid, and contains several hundred grams uranium per liter. It is generally countercurrently extracted with the organic phase (30 volume % TBP in normal hexane) at a ratio of 10-15:1. The uranium concentration in the organic phase is 20-30% of the aqueous phase concentration while in the depleted aqueous phase it is <1% of the initial concentration. These concentrations and extraction efficiencies are typical; however, the health physicist should collect analytical data and process parameters for the specific system in question, since equipment efficiencies vary greatly depending on the constituents of the incoming solution. The depleted aqueous phase (raffinate) will contain most of the metallic impurities, the uranium decay products and the fission products. However, any transuranics present must be removed by reduction to an organic-insoluble valence state (normally trivalent state). The organic phase is then countercurrently extracted to produce a purified aqueous solution of uranyl nitrate.

Hazard. The radiation hazards from the purification process arise mainly from the concentration of uranium decay products and any fission products in the raffinate. Since the raffinate is a waste, these concerns extend through the waste treatment processes. Beta radiation fields of a few mrad/hr to a hundred mrad/hr may be experienced. The gamma field is generally inconsequential from uranium decay products; however, fission products may present problems depending on the source material. The initial task is to obtain good analytical results to identify the

radionuclides present initially and, if possible, at each stage of the process. Any spills of uranyl nitrate solution or raffinate should be carefully cleaned since the corrosive liquids will react with most surfaces and complicate decontamination. The residue from spills can lead to elevated beta levels.

3.1.4 Conversion Processes

The end product of most conversion processes at DOE facilities is uranium metal or UF_6. However, depending on the end use, many different compounds can be formed once the uranium is purified. Figure 3-12 shows a flow sheet for conversion of UO_3 to uranium metal.

Process. In this step, an aqueous solution of UNH is first concentrated by evaporation to provide a viscous liquid with the approximate composition of uranyl nitrate hexahydrate. This viscous liquid is then thermally denitrated to uranium oxide (approximate formula UO_3 or U_3O_8) as the product with waste emissions of nitrous oxides, water vapor, and nitric acid vapor.

Hazard. The major hazards occur due to the plugging of the denitration equipment, since maintenance must be performed to unplug the equipment. This operation requires special controls due to the generation of uranyl nitrate dust. The UO_3 (orange salt) is typically produced as small, dense spheroids (5-10 μm) which are not readily airborne. Since the uranyl nitrate solution is extremely corrosive, substantial fixed uranium contamination with associated decay products will be present on internal equipment surfaces. For low enrichments of uranium, this could result in significant beta dose rates when maintenance is being performed. Also, spills of uranyl nitrate solution should be promptly cleaned up since the corrosive solution will react with most surface leaving significant fixed uranium beta activity. Decontamination techniques should take into account the less soluble oxides.

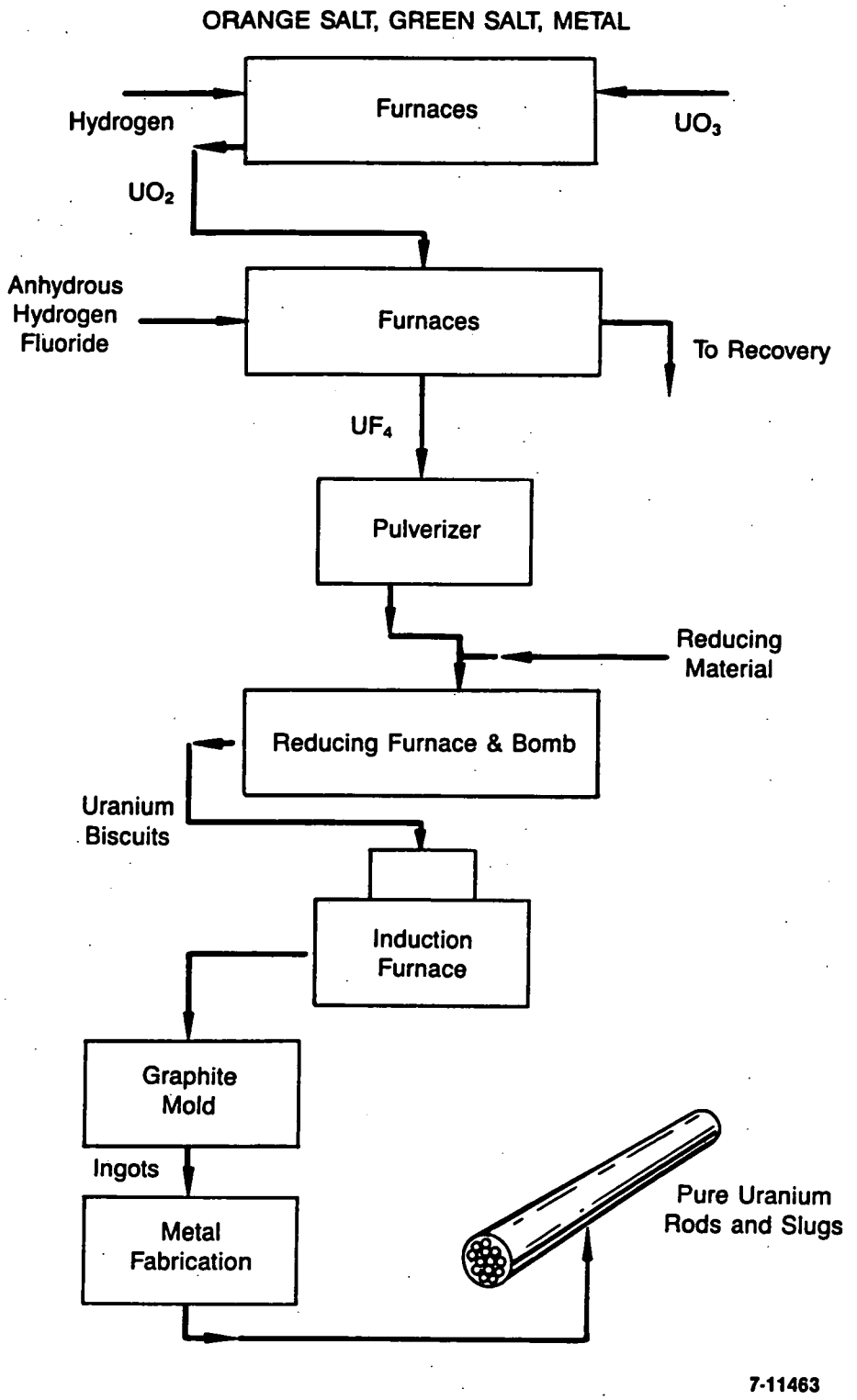

Figure 3-12. Orange salt, green salt, metal.

Process. The UO_3 is reduced to UO_2 by reaction in a countercurrent fluidized bed reactor with cracked ammonia gas (3 H_2: 1N_2) at a temperature of approximately 600°C. The exhaust gas is filtered and cooled, and the excess hydrogen is burned.

Hazard. When uranium decay products are present in the UO_3 feed, they will tend to concentrate in the unreacted waste. The collection and removal of UO_2 presents some potential for airborne contamination. Any system maintenance presents significant potential for creating airborne contamination. Localized containment should be utilized, if practical. The UO_2 is only slightly soluble (between typical W and Y compounds). Decontamination techniques must consider the limited solubility of the UO_3 and UO_2 compounds.

Process. In DOE facilities, the hydrofluorination of $UO_2$to UF_4 (green salt) utilizes two countercurrent fluidized bed reactors. The first reactor runs at 300°C and partially converts UO_2 to UF_4 while reducing the HF content of the gas stream to approximately 15%. The second reactor is feed anhydrous HF and the partially converted UO_2, and converts approximately 95% of the UO_2 to UF_4.

Hazard. Where uranium decay products or other impurities are in the UO_2 feed, they will concentrate in the unreacted waste and may produce significant exposure rates. The UF_4 is a powdery, easily dispersed material. Stringent control measures should be incorporated into the product unloading area to limit the spread of contamination and minimize personnel contact with the UF_4 powder. Again, due to the range of transportability classes (solubility), both urinalysis and in-vivo counting are advisable. Decontamination techniques should be developed to handle the various compounds that may be present. Lastly, maintenance of equipment could involve significant beta dose rates from internal surfaces of equipment.

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Process. At the DOE facility at Paducah, UF_4 is converted to UF_6 by reaction of the UF_4 with F_2 in a tower reactor. The reaction takes place at a flame temperature of 1600°C; the reactor walls are then cooled to 500°C. The UF_6 produced is condensed as a solid in cold traps; fluorine is recycled and secondary traps remove residual UF_6. The UF_6 produced is exceptionally pure; UF_6 content is 99.97%.

Hazard. Any uranium decay products in the feed will be highly concentrated in the unreacted tower waste; exposure rates greater than 1 rad/hr may be detected. System maintenance is another source of significant contamination and exposure potential. System containment must be assured due to the high chemical toxicity of UF_6 and F_2. Both UF_6 and F_2 have excellent visual and olfactory warning properties. Again, internal equipment surfaces will have significant beta exposure rate from the presence of uranium decay products.

Process. Enriched or depleted uranium is usually produced in the form of UF_6 from the United States gaseous diffusion plants; but it is normally utilized as metallic uranium or as UO_2. Both UO_2 and metallic uranium can be produced from UF_4; thus the first step is often to reduce UF_6 to UF_4 by vapor-phase reduction with hydrogen. The UF_4 may be hydrolized with steam to form UO_2, with hydrogen fluoride as a by-product.

Hazard. Since the UF_6 is reacted as a gas, any uranium decay products or other radionuclide impurities will be left in the solid phase in the UF_6 container. Thus, the UF_4 produced will exhibit only uranium hazards. However, the presence of fixed uranium contamination on internal surfaces with the associated decay products will provide some beta exposure when equipment is opened for maintenance. Another hazard is that the reaction vessels develop cracks and will need to be regularly inspected and replaced periodically.

Production of Uranium Metal. The production of uranium metal is accomplished at DOE facilities utilizing two basic processes: (1) Reduction of UF_4 by magnesium (2) Reduction of UF_4, by calcium in the "oralloy reduction process."

Magnesium Reduction

Process. The Magnesium Reduction Process is shown schematically in Figure 3-13. Green salt, UF_4, is converted to the metal by reduction with magnesium. In this process, the uranium fluoride is mixed with magnesium powder and placed in a reaction vessel called a bomb. After the mixed feed material is placed in the container, the container is closed and placed in a furnace. The temperature is raised to about 1300°F and a reaction occurs. The uranium separates from the mixture and becomes a molten metal in the center. The vessel is cooled and then opened. The slag or refuse material must be broken away and the uranium metal, or derby, removed.

Hazard. Most of the daughter products and most of the radioactivity, is concentrated in the slag. Dose rates up to several hundred mrem per hour may be encountered in low enriched uranium processes. The derby is cleaned to remove slag that may be stuck to the outer surface. If a more pure metal is needed, the derby may be reprocessed by arc melting to remove some of the impurities. In each processing step, the waste material concentrates the radioactivity, primarily from the removal of the uranium decay products.

Oralloy Reduction

Process. In the calcium reduction process, uranium tetrafluoride is reduced to the metal "button" form (see Figure 3-13). Granular calcium metal (minus 4 plus 20 mesh) is mixed with the "green salt" and loaded into a stainless steel reactor lined with a magnesium oxide liner. An igniter capsule and a lithium biscuit are added to the reactor. The igniter will

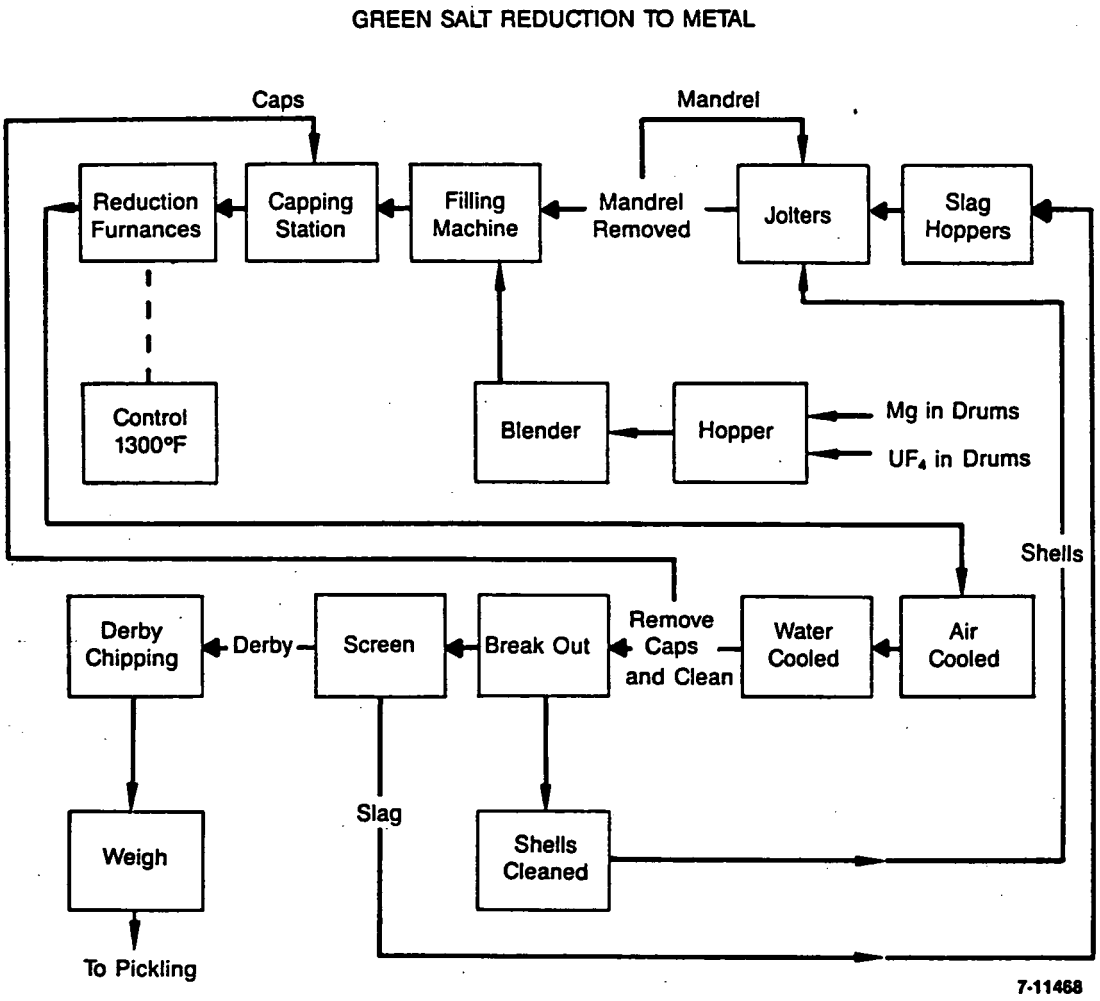

Figure 3-13. Flowsheet for the production of uranium metal by the reduction of UF_4 with magnesium.

aid in initiating the reaction when the reactor is heated in an induction furnace; the lithium will form a eutectic mixture with the calcium resulting in a lower-setting temperature for the the calcium fluoride slag. This procedure will have the advantage of producing a sound "button" with a smooth top, shown in Figure 3-14. The reactors are purged of air and fired under an argon atmosphere. The reaction will usually occur at about 600°C, as measured by a thermocouple attached to the outside wall of the reactor. The yield in converting uranium tetrafluoride to metal by this process is usually greater than 99.50%. The calcium fluoride slag, liner, and sand fines are routed to the chemical recovery area; the coarse sand, after screening, is reused in the reduction process.

Hazard. As in the magnesium reduction process, most of the uranium decay products are concentrated in the slag. However, since most of the uranium produced in this process is very highly enriched, the penetrating radiation levels are relatively low from the uranium decay products.

3.1.5 Physical Conversion Processes/Foundry Operations

Much of the DOE activity with uranium involves Foundry operations for the physical conversion or metallurgical processing of the metal. This includes such operations as melting and casting, forming, extruding, shearing, etc. Each activity has its unique hazards.

Melting and Casting

Process. Melting and casting involve a change of state of elemental uranium. The first metallurgical process generally is the melting and casting of uranium into billets, ingots or special cast shapes for further processing. Additionally, some alloying of uranium with niobium (2% and 6%) and titanium is also accomplished in induction furnaces.

Hazard. This change of state tends to concentrate on the surface any impurities in the uranium, for example, uranium decay products. This leads, in turn, to significant beta and gamma radiation exposure potential

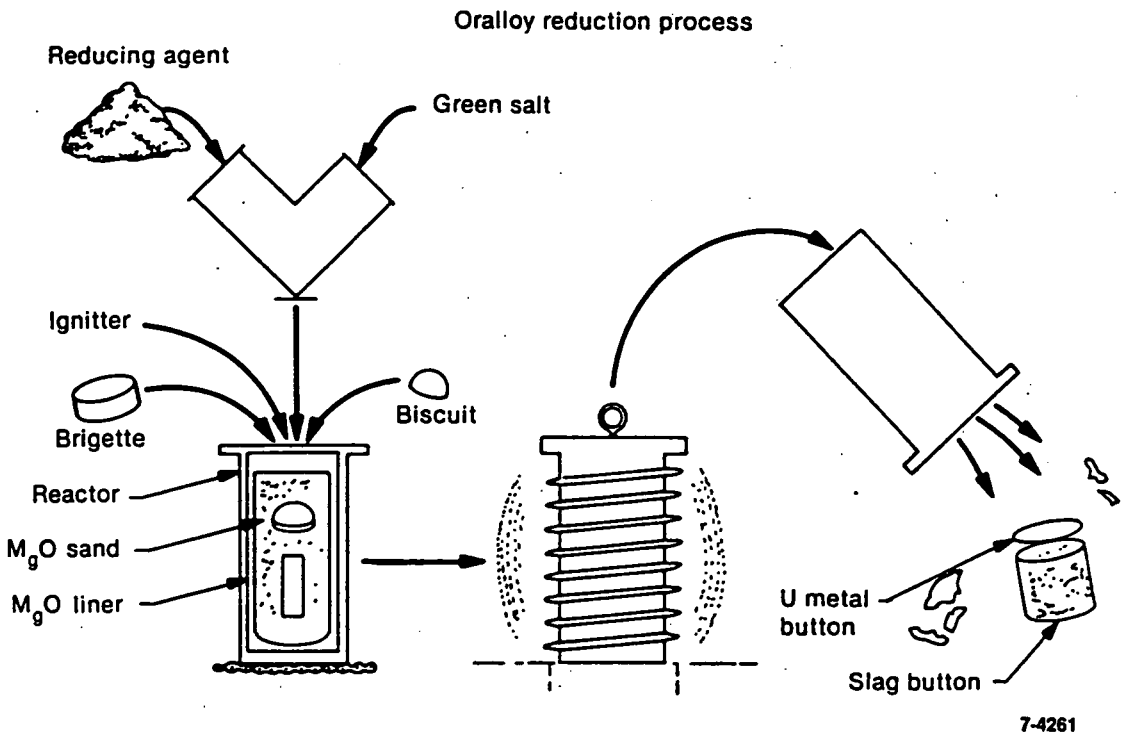

Figure 3-14. Enriched uranium reduction process.

in the melting operation with the beta radiation the controlling hazard. There is also a significant problem with uranium oxide dust generated from these operations. Finally, the process may concentrate transuranics or fission products, or both, if they are present in the uranium feed.

Mechanical Conversion

Process. Forming, extruding, and swaging are examples of mechanical conversion processes. After the derby or metallic uranium billet has the surface impurities removed, it may be mechanically processed into various shapes. It may be extruded into rods or tubes, forged into complex shapes, or swagged into bars or plates. For each of these processes, the uranium is prepared for the subsequent operations by cutting into the correct size pieces, encapsulating it in protective containers, providing a surface lubricant, or other steps, depending on the process. For example, in the extrusion of depleted uranium to form rod, the derby is preheated to several hundred degrees, then placed in a hydraulic extrusion press in which a ram forces the material through a hole or die. A lubricating material is applied during the extrusion to minimize galling. The extruded bar is usually rolled during cooling to reduce warping. Forging is another type of pressure forming process in which a preheated piece is placed in a press or die and pressure applied to force the material into the shape of the die. The uranium is preheated to a plastic state to reduce the pressure required. Swaging is a type of hammer forging in which the material is repeatedly hit and formed into the shape desired. Again, preheating is used to make the material easier to work.

Hazard. A number of hazards may be encountered in these operations. With respect to external exposure hazards, the radiation dose rate from a depleted uranium rod is about 250 mrad per hour at contact. Since most of the contact dose rate is due to beta radiation, it is relatively easily shielded. A pair of heavy gloves or a 2-ft distance will reduce the dose rate by a factor of 100. However, extremity doses (hands) should be measured and records maintained if a significant amount of manual handling

is performed. In each of these processes, uranium contamination is generated from oxidation of the surface and transfer of material to the dies, presses, hammers, and handling tools. Uranium is highly reactive in most atmospheres. Surface oxidation, similar to rusting of iron, produces removable contamination on the bare material. This contamination can be transferred to gloves, transport carts, tools, and equipment. Frequent surveys should identify and quantify contamination spreads. Gloves, in particular, may concentrate contamination and become as active a source of hand dose as the uranium itself. Airborne contamination may also become a problem if contamination levels are not adequately controlled.

One method used in the extrusion process to reduce contamination and exposure potential is to place the uranium billet inside a copper container, evacuate the air, and extrude the whole unit. The copper acts as a lubricant for the die and as a protective cladding for the uranium. The copper cladding on the resulting uranium rod, although very thin, protects the uranium from oxidation and acts as a shield for the alpha and beta radiation. This technique reduces contamination levels to non-detectable and surface dose rates to less than 10 mrem per hour. Of course, the uranium is now copper clad, and if this is not desirable, the copper cladding will have to be removed. Other control methods, such as working in inert atmospheres, inside glove boxes, and in ventilated hoods, are used to minimize and confine the contamination that may be generated.

Each operation must be evaluated for the radiation dose that may be received and the contamination that may be generated. The requirements of the operation must be considered and controls established to protect the workers. A typical supplement to the engineered controls is the use of protective clothing. Gloves, lab coats, coveralls, and shoe covers are routinely used as barriers between people and contamination. Caution must be exercised in the designation of protective clothing to ensure that the hazard from wearing the clothing does not outweigh the protection afforded; i.e., loose clothing around rotating machinery, flammable material around very hot operations, etc. Protective clothing may concentrate contaminants--providing higher skin and/or extremity doses.

Metal Working Machining

Process. The physical modification of uranium metal by various metal-working operations also tends to disperse uranium metal and uranium oxides. Usually, these operations (e.g., machining, lathing, sawing, drilling, etc.) do not tend to concentrate contaminants from the uranium as do melting or casting.

These operations present an external exposure hazard (both whole body and extremity) from the mass of uranium present and from the concentration of decay product or surface. Such operations also present an internal exposure hazard from the generation of finely dispersed uranium oxide dusts or fumes containing uranium. In general, the machining of uranium metal does not produce uranium metal particles of respirable size since the machining is generally performed utilizing a water-based coolant. However, the control of uranium chips generated by machining requires controls to minimize contamination and trackout by chips becoming embedded in shoes.

Hazard. Machining and lathing operations on depleted uranium material expose the operators to potential radiation dose to the whole body, extremities (hands primarily), eyes, and skin. Safety glasses provide shielding to the eyes from the beta radiation as well as protection from flying chips. Gloves also provide some shielding from the beta radiation as well as a barrier between loose contamination and the skin.

Machining and lathing steps are frequently production line operations. Consequently, other pieces are normally in the processing line. Material not actually being worked on should be located away from occupied areas or shielded to reduce the radiation dose to the operator. Entry of personnel into the work area may be controlled to reduce the exposure of people to the radiation. Periodic surveys should determine the radiological conditions and dosimeters needed to detect and record personnel doses.

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The hazard from internal exposure to uranium during these operations must also be controlled. There is a potential for inhalation of airborne material, ingestion of contamination and injection into the body from cuts and punctures. Machining and lathe operations frequently produce fumes at the point of contact between the tool and the uranium. Some of the fumes contain uranium. Also, the movement of the work piece may cause some of the oxidized uranium on the surface to become airborne. If not controlled, the airborne material may be inhaled. Contamination generated by the work may be transferred to gloves and protective clothing and to the skin. Injection of radioactive material directly into the body and the blood stream is almost always the result of accidents. Examples of injection accidents are the puncture of the skin through contaminated gloves by burrs and shavings and cuts on the hands and fingers by sharp (and contaminated) tool bits.

A significant hazard in machining and lathe operations on uranium results from its pyrophoric properties. Chips, filings, and turnings of uranium may spontaneously ignite and burn. The burning of uranium not only produces airborne material which may be inhaled, but may ignite other materials. The potential for spontaneous ignition may be reduced by using large quantities of a water based lubricant for cooling during cutting operations and adjusting the cutting speeds to reduce the temperatures. The work area should be kept as clear as possible of chips and turnings and other combustible materials. Metal chips and shavings should be stored under water to keep them cool and to reduce their contact with air. For some operations, it may be necessary to provide a dry, inert gas atmosphere to control spontaneous ignition. Fire fighting equipment must be readily accessible near these operations. Dry sand to cover burning material and dry powder extinguishers, such as MET-L-X, can be used to put out uranium fires. Water should not be used on uranium fires because of the potential for hydrogen generation. Storage of waste chips and turnings should be provided well away from work areas, preferably in a separate, ventilated and filtered building. For a discussion of pyrophoricity see the article by R. B. Smith referenced in the Bibliography.

3.1.6 Uranium Material Handling

Process. In connection with various uranium conversion processes and with the fabrication of fuel elements, material handling is an area of importance. In many cases, the uranium compounds are in a physical form that is readily dispersible and are also in a relatively insoluble chemical form. Materials handling processes must be managed to prevent airborne contamination which could result in internal dose and to prevent the spread of surface contamination.

Hazard. One of the most important radiation controls used is ventilation and filtration. It is necessary to provide uncontaminated air in the work place for personnel to breathe and to assure that only uncontaminated air is discharged to the environment. Ventilation systems should be designed to move contaminated or potentially contaminated air away from occupied areas. Equipment such as hoods and glove boxes should be used during dust and fume generating operations. The exhaust from contaminated areas should be filtered before discharge. High Efficiency Particulate Air (HEPA) filters are typically used. It is important to control the use of portable, temporary ventilation, such as fans, to reduce contamination spread.

3.1.7 Decontamination Activities

Process. Any facility that processes uranium compounds must have an active decontamination program to support maintenance activities and to cleanup after process outages. Good industrial housekeeping practice is, in itself, an effective method for maintaining contamination levels within acceptable levels.

In general, most decontamination processes can be classified as "wet" or "dry." Wet decontamination processes, as the name implies, utilizes aqueous solutions to collect uranium contamination. The solutions may be acidic (nitric, citric, etc.) or they may be basic (carbonates); in any event, they should be tailored to the compound to be decontaminated.

Hazard. Aside from the chemical hazards from the solutions themselves, a number of potential problems exist. First, there is the potential of resuspending the contamination which may create an inhalation hazard. Second, the decontamination may involve exposure to uranium decay products or RU contaminants. Third, wet methods may leave less soluble radionuclides, such as thorium, behind which could continue to present a hazard even though the area was decontaminated of other radioactive material. Dry decontamination methods also present a variety of hazards. The potential for resuspension of dry contamination is greater than it is with wet methods. Specially designed HEPA filtered vacuum equipment should be utilized to minimize resuspension hazards. Whenever enriched uranium is present, engineering and administrative controls must be implemented to minimize criticality hazards. The equipment design of collection vessels should be critically safe.

3.1.8 Uranium Recovery

Process. The recovery of uranium from contaminated equipment, waste materials, and spent fuel is essential from economic and ALARA standpoints. Most uranium recovery uses solvent extraction and subsequent calcination of the uranyl nitrate (UNH) solution. In this process, solutions bearing UNH are countercurrently extracted with an organic phase of tributylphosphate in high grade kerosene. The UNH is preferentially extracted into the organic phase. The UNH-bearing organic phase is then contacted with deionized water and the UNH is released into the aqueous phase. The UNH bearing solution is concentrated and fed to a calciner where it is thermally decomposed into uranium oxide.

Hazard. In this process, fission products, such as technetium, and uranium decay products are concentrated in the aqueous waste stream (raffinates). Depending on the source of solutions, the raffinate may exhibit high beta dose rates (up to several hundred mrad/hr) on contact. Generally, the dose to personnel may be effectively reduced by utilizing

low Z material for solution containers or shielding and/or putting distance (>2 ft) between the solution and personnel and limiting personal handling of solution containers. Where spent fuel is reprocessed, there is a wide variety of fission products of high activity that are of primary concern from a health protection standpoint. A detailed discussion of irradiated uranium fuel reprocessing is beyond the scope of this manual.

3.2 Uranium Forms and Uses

In summary, a knowledge of the processes, chemical and physical forms of uranium, the U-235 enrichment and the absence or presence of RU contaminants is essential to provide an effective health protection program. Of particular importance is knowledge of the solubility of the uranium compounds involved, their particle size, and dispersibility of the material. It is also important to verify the effect of the particular processes on the relative concentration of RU contaminants and uranium decay products to uranium. This knowledge of the process and the uranium materials involved will guide the design of the health physics program.

3.3 Summary Discussion and Guides

Overall, DOE facilities/programs have been very successful in controlling radiation exposures and protecting worker health. Radiation doses to workers have historically been well below standards and, in general, average annual radiation doses have been reduced over the years as more knowledge has been gained and radiation protection practices improved. However, accident/incidents have occurred and will likely continue to occur. The challenge is to develop programs that limit the consequences of a loss of control. The development and implementation of good health physics practices will minimize the probability of serious problems. In addition, good practices will result in a reduction in external and internal dose to personnel. However, the first step is to fully understand the uranium materials handled and the processes involved.

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In general, the types of hazards present in DOE facilities are a function of the enrichment level, the chemical form of the uranium, the presence of contaminants, and the processing methods and controls utilized. While the individual processes have been discussed in the previous sections, some basic principles may be derived from the DOE experience in the evaluation and design of health physics programs in uranium facilities.

Operations utilizing uranium hexafluoride will not generally present a hazard from penetrating radiation except in the handling of empty cylinders and in the maintenance and decontamination of process equipment. Internal exposures may be of significant concern both from the standpoint of acute and chronic exposures. However, UF_6 has excellent visual and olfactory warning properties. As a result, chronic exposures will be detected and controlled at levels far below chemical or radiological toxicity limits since workers will be aware of any significant releases of UF_6. This general principle will remain true except for "very highly enriched" (VHE) uranium hexafluoride (>90% enrichment). It is important to be cognizant of the extreme toxicity of UF_6 in the event of a large release; most fatalities associated with uranium processing have occurred when individuals were exposed to high concentrations of UF_6 and its reaction products during a release. Any operation that could result in a large release of UF_6 (primarily liquid handling or transfer operations) should have secondary containment, if possible, and personnel should be provided with escape equipment, and evacuation routes should be established consistent with Life Safety Code requirements. Finally, it is important to routinely analyze UF_6 from each major source to identify the presence of contaminants, both radiological and otherwise (e.g., organics and "light" gases) which could pose health and/or safety hazards. These contaminants must be evaluated with respect to their behavior in the processes involved to determine if they will concentrate in any areas to the extent that their hazard relative to uranium will change significantly and perhaps become the controlling factor. For example, in the gaseous diffusion process, technetium compounds are concentrated in the high enrichment sections of the diffusion cascade where they become the controlling hazard.

Chemical conversion processes involve a wide range of uranium compounds and transportability classes from D to Y. The chemical and physical forms of uranium utilized must be considered in the establishment of controls and monitoring programs. In DOE experience, a few individuals have received internal exposures in excess of established limits in some of these operations. Generally, these occurred due to the use of inappropriate monitoring strategies (e.g., urinalysis for compounds having Class Y transportability) and/or a lack of adequate control. Controls should be evaluated with reference to radiotoxicity and chemical toxicity of the materials. Many of the conversion processes will concentrate uranium decay products and contaminants in the waste streams to levels representing significant hazards. These will represent both internal and external exposure hazards. In most cases, the primary external hazard will be due to beta radiation from uranium decay and/or fission products especially to the extremities.

In the physical conversion processes, remember that low enriched uranium metal will present a beta radiation field of 200-250 mrad/hr on contact. Also, while gamma radiation levels will be much lower (typically 5 mr/hr), if large amounts of uranium metal are stored the cumulative levels may be significant. Those physical conversion processes that involve a change of state will tend to concentrate uranium decay products and impurities on the metal surfaces which may dramatically increase surface radiation levels. Finely divided uranium chips and turnings can be a fire hazard due to pyrophoricity of uranium. Finally, uranium metal exposed to the atmosphere will form a coating of oxide very quickly which will be easily removed as dust during the handling of the metal. Contamination control practices are very important in such operations.

An aggressive nuclear criticality safety (NCS) program is required for facilities which process enriched uranium. Since the consequences of a criticality are so great, double contingency is required in the design and operation of all processes involving fissionable material. No single failure should be capable of causing a nuclear criticality accident. NCS

issues must be evaluated by qualified professionals trained in the NCS discipline. A brief discussion of NCS is presented in Section 8; which is intended only to provide information sufficient to alert health physics personnel of the need to obtain specific plant related training and the assistance of NCS professionals in evaluating the specific operations and conditions.

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SECTION 4

RADIATION PROTECTION MANAGEMENT

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SECTION 4

RADIATION PROTECTION MANAGEMENT

The responsibility for safety in the broadest sense rests with management. However, every employee must take personal responsibility in working according to established rules and industry standards to assure personal protection and a safe working environment. The requirements for the establishment of a Radiation Protection program at uranium processing facilities are dictated by DOE orders, and are based on sound radiological requirements such as those recommended in ANSI and NCRP documents. But the foundation of a successful Radiation Protection program is the support of the ALARA (As Low As Reasonably Achievable) concept by upper management. Support and implementation of good radiation protection practices by Radiation Protection management and all employees is no less important.

The purpose of a Radiation Protection program is to maintain a radiation-safe environment. The Contractor will have a written policy on radiation protection (including ALARA). The specific program responsibilities include:

a. Establishing and maintaining operational procedures so radiation exposure to workers and to the public is kept as low as reasonably achievable below regulatory levels

b. Instructing personnel in safe work practices and the nature of hazards associated with exposure to ionizing radiation

c. Assuring that personnel-monitoring duties are conducted as directed

d. Conducting periodic radiation surveys as required

e. Investigating each case of excessive or abnormal exposure, and

f. Maintaining all pertinent and required records.

This section establishes the basics of a sound radiation protection program.

4.1 Organization

A clear organizational structure established by the individual contractor to best meet the business needs of the facility is necessary. Although no one organizational structure is best for every facility, there are some common characteristics which are considered basic to supporting an effective radiation protection program:

a. Management commitment to safety principles in general and to ALARA specifically

b. Radiation protection program independent of operating/ manufacturing functions and clear delegation of authority to Radiation Protection management

c. Adequate resident radiation protection staff within each facility

d. Adequacy of personnel, equipment and funding to achieve radiation protection goals

e. Specific, formal assignment of ALARA responsibility.

4.1.1 Management Commitment

Management commitment to safety is the most important characteristic. If the management commitment to safety is strong, the Radiation Protection program will be valued/respected with adequate authority to perform necessary assignments and program implementation. Adequate personnel, equipment, and funding will also be available.

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4.1.2 Radiation Protection Organizational Independence and Reporting Level

Since there are many functional organization arrangements, each facility should design an organization specific to its needs and circumstances. Whatever the organizational structure, independence of the Radiation Protection functions should be assured through effective access to top management. A recommended organizational structure combines all the occupational health and safety activities under one manager at a general manager reporting level. Figure 4-1 is an example of an organization with a centralized health and safety program.

A mid-sized facility with a limited scope of work could combine a number of functions and still maintain the independence of the Radiation Protection function. Figure 4-2 illustrates such a structure.

Finally, a small, single function facility can maintain the independence of the health physics program through a highly placed radiation protection committee. In Figure 4-3, the Health Physicist reports to the Engineering manager but maintains access to high levels of management and decision making through the Health Physics or ALARA committee.

4.1.3 Adequacy of Personnel and Equipment

Evidence of the commitment to a sound Radiation Protection program and ALARA can be observed in the adequacy of the instrumentation and equipment and the competence and enthusiasm of the personnel involved in meeting these goals. Basic to production of this evidence is an adequate budget. Specific recommendations on staffing and staff qualifications are made in Sections 4.2 and 4.3.

4.1.4 Assignment of ALARA Responsibility and Authority

Responsibility for the ALARA program should be clearly defined and placed with a specific individual or organization and be recognized as a

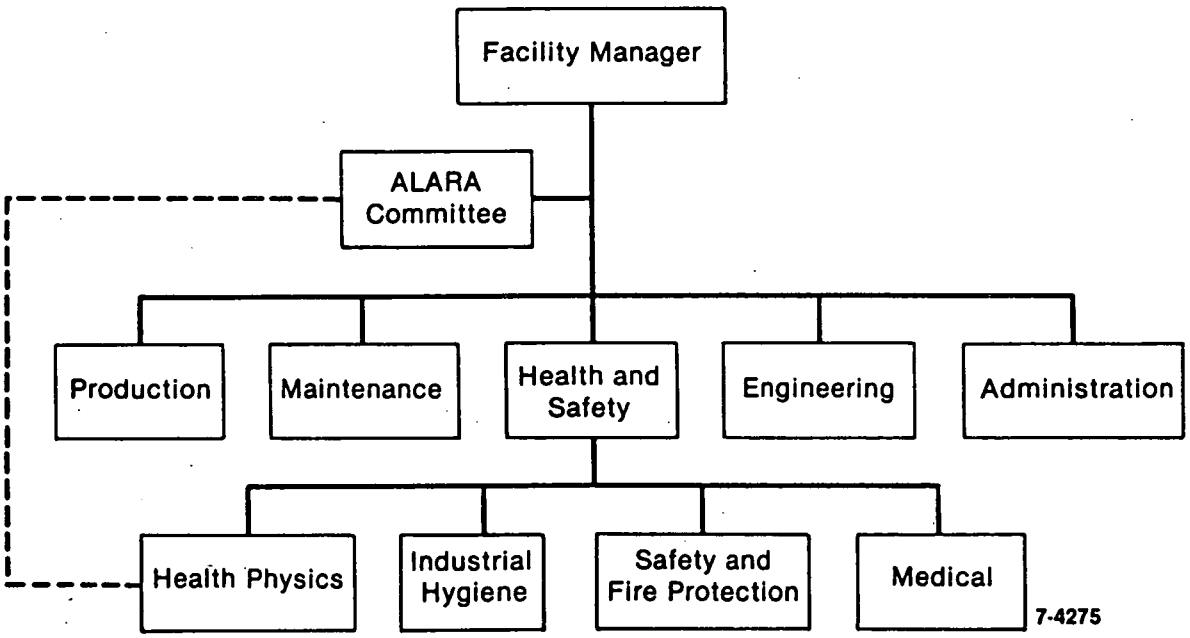

Figure 4-1. Organization chart with centralized health and safety program.

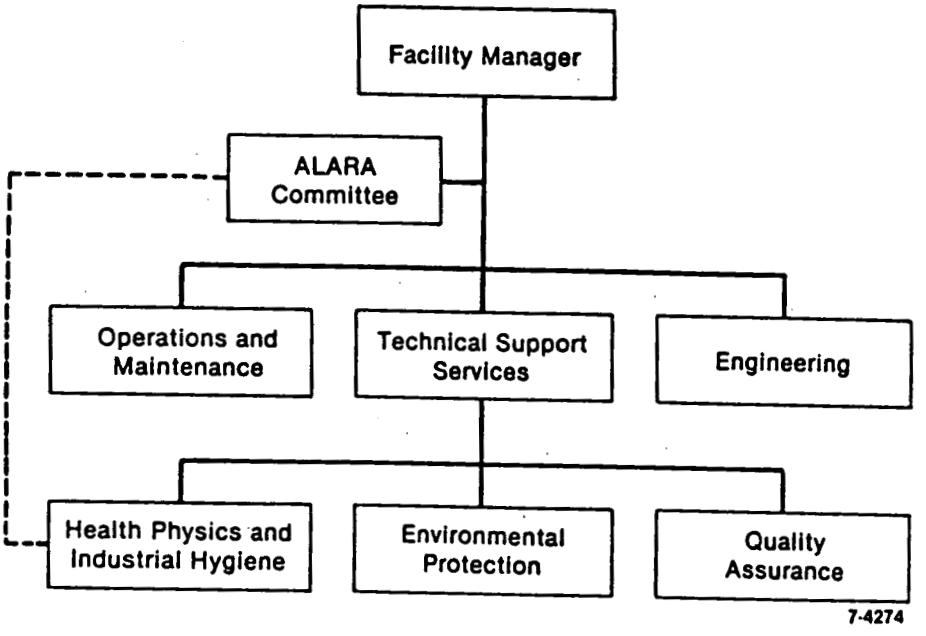

Figure 4-2. Organization chart for mid-sized limited scope facility with health physics independence.

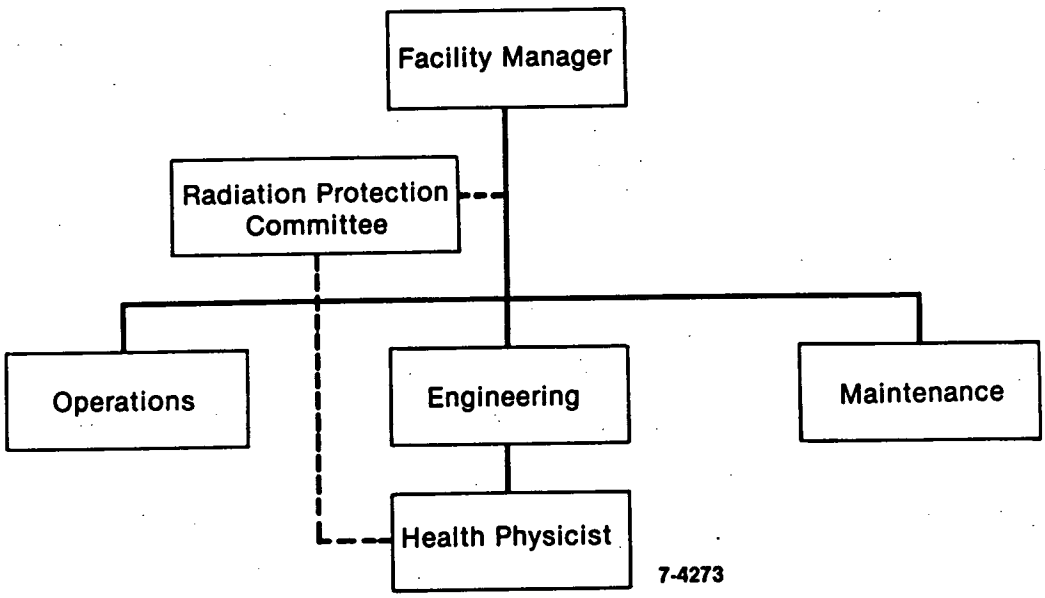

Figure 4-3. Organization chart for small single function facility with health physics independence.

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major/valued function. Operational management is the appropriate choice. Responsibility clearly defined would include:

a. Overall expectations of higher management for conduct of the program

b. Time schedules

c. Goals to be achieved.

Further, basic goals should be established by operational organizations with the help of Radiation Protection management where exposure problems are most likely to occur.

Assignment of authority in addition to responsibility is necessary to achieve ALARA goals. While the basic authority and responsibility for employee safety rests with the facility manager and line manager, in the ideal situation, the line management Radiation Protection and other staff work together to achieve working conditions where safety is maximized.

Specific authority granted to Radiation Protection management and staff should include:

a. Approval of construction plans including facility modification wherever radioactive materials are used, stored, or generated

b. Issuance and/or approval of radiation work permits

c. Determination and approval or both of operational protective measures to ensure ALARA

d. Training development and qualification of radiation workers

e. Emergency shutdown of imminently hazardous operations.

4.2 Staffing and Staff Qualifications

Central to maintaining a safe, operating facility and achieving ALARA goals is a staff of competent dedicated and motivated personnel. In this section staff requirements and qualifications will be discussed for both professional and technician categories.

The number of professionals and technicians required to meet the ALARA goals of a facility will depend on many factors such as the magnitude of source inventory, the hazard potential of sources, and the size of the organization. While no formula exists for making decisions about staffing, some general guides will be outlined.

4.2.1 Professional Staffing and Qualifications

At least one professional health physicist should be on the coordination staff of a uranium facility as it is being built. This professional should be a certified health physicist or have several years of experience in the operation of a uranium facility. As construction nears completion, the remainder of the staff should be hired and trained under the guidance of the experienced health physicist. This step allows personnel to grow with the facility. Once operation of the plant begins, potential problems will already have been identified and administrative or engineering changes made to correct them. Professional staff hired after start-up should have several months of orientation through direct involvement under experienced personnel before working independently.

Professional health physicists, generally at the graduate level, should have appropriate training and experience in radiation protection to make necessary radiation measurements, evaluate their significance, and devise corrective measures. General competency in the profession can be verified by testing and continuing education through the American Board of Health Physics. Other organizations certifying Health Physicists are the American Board of Radiology and the American Board of Industrial Hygiene (Radiological Aspects).

4.2.2 Technician Staffing and Qualifications

Routine activities required for work place surveillance and documentation are usually performed by technician(s) following methods and procedures established/approved by a professional health physicist. The technician may also offer advice to employees on matters related to radiation safety, assist in training other employees, and relieve health physicists of routine tasks. The need and use of technicians is dependent on the nature and extent of radiation protection coverage. The training, experience, and competence required of and/or possessed by the technicians as well as the technical complexity of the radiation protection program should be considered in specific responsibility assignments.

Employees hired as health physics technicians should have a high school diploma with one year of high school physics as a minimum. A one or two-year vocational technology degree in a radiological science curriculum is highly desirable. Most companies provide specialized health physics technician training directly (in house) or through company sponsored programs. Testing of both knowledge of subject matter and job specific skills should be used to verify the competency of the health physics technician. The general competency level can be established by registration by the National Registry of Radiation Protective Technologists. Refresher training and testing at specified intervals (2 to 3 years minimum) is recommended.

4.2.3 Staffing Levels

The professional health physics and health physics technician staffing levels required vary depending on the types and quantities of nuclear material being handled, the complexity of the operations performed, and the number of workers involved. Some staffing surveys show a ratio of all Health and Safety staff to total plant population of one to six percent. When the health physics staff is compared to radiation workers, the range is from 3 to 15%. A general guide to the "average" facility is one health physicist or HP technician for every twenty radiation workers (5%).

4.3 Training and Education

In order for employees in a uranium facility to have the understanding necessary to work safely and take personal responsibility for their own safety, they need training appropriate to their work. This training is provided by the company through supervision scheduling. Training for workers at a uranium facility should be offered to four groups of employees. These groups are the health physicists, health physics technicians, radiation workers, and all of the other employees. There is also a need to educate the public.

4.3.1 Training for Health Physicists

It is important that Health Physicists keep current with the radiation safety field and especially that phase related to the facility for which they are responsible. Certified health physicists must maintain their credentials by obtaining continuing education credits through a program of continuous study and upgrade. These credits are earned by attending classes and seminars offered by professional organizations and universities. Participation in local Health Physics organizations also helps maintain knowledge and encourage the exchange of information. Because health physicists are professional, it is expected that they will also spend personal time in professional reading to keep current. Management should formulate a professional development plan for each health physicist which would include encouragement to conduct plant-related research and submit papers for delivery at conferences or for publication in journals.

4.3.2 Training for Health Physics Technicians

Health physics technicians perform most of the routine surveillance activities associated with maintaining a safe working environment at a uranium facility. The training provided should be thorough in covering both the theory as well as the skills required. The theory is important in

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assuring that the health physics technicians understand why, as well as how, to complete an assigned task. The training program should be competency-based rather than time-based. While the subject matter covered may vary according to the particular operations and practices at the facility, training should cover the following topics, as a minimum:

a. Basic nuclear physics

b. Metric system

c. Monitoring techniques

d. Radiation instrumentation

e. Respiratory protection and engineering ventilation control systems

f. Emergency procedures

g. Radiation biology and effects

h. Air sampling and internal dose determination

i. Decontamination

j. Radiation safety guides, standards and orders

k. ALARA guidelines and procedures

l. Responsibilities of working groups and personnel.

It is recommended that a qualification program be established for health physics technicians which would include both written and practical testing. It is also recommended that refresher training and retesting on a regular basis be part of the program. Management may wish to recognize and

reward technician registration by the National Registry of Radiation Protection Technologists.

Supervisors of the health physics technicians should also be well trained. In addition to supervisory training, supervisors should be knowledgeable in all of the same topics listed above plus have additional knowledge of overall facility operations and procedures.

4.3.3 Training for Radiation Workers

Radiation workers are defined as individuals who are reasonably expected to receive an exposure greater than 100 mrem/yr. Workers should be well informed of the hazards, safe operating procedures, and materials in the work place as a basis for an effective program.

It is recommended that Radiation Workers complete classroom training on such topics as radiation fundamentals, radiation biology, and measurement and control of radiation. On-the-job training specific to the materials and equipment in the work area should be conducted for each worker. Radiation workers should be qualified or certified to work alone through competency-based, written and practical tests. Refresher training and retraining should occur at regular intervals, usually every two years.

4.3.4 Training for All Other Facility Personnel

All employees at a uranium facility should have basic knowledge of radiation protection, which should give all employees a sense of the facility mission, the basis for the rules and procedures and may also allay any fears or uncertainties regarding their personal safety.

All employees should receive an orientation to radiation safety as they begin work at the facility, regardless of job assignment. This initial orientation should include such topics as:

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a. Risks of low-level occupational radiation exposure and prenatal exposure

b. Basic radiation protection concepts

c. Radiation protection policies and procedures

d. Employee and management responsibilities for radiation safety, and emergency procedures.

Reorientation every two years is suggested to maintain employee awareness.

4.3.5 General Public Education

The need for public education in radiation safety will vary greatly depending on the materials and processes used at the facility and the interest displayed by the public concerning the facility. If the public is interested, the need to inform the public about any matters which may affect its well-being should be accepted. Hence, facility management should provide the public, press, and health officials with appropriate information on the facilities/processes, hazards and safeguards.

If a new facility is to be constructed, thorough discussions of the technical and health aspects should be held early in the planning stages with state and local officials as well as in open discussion with the public. These discussions set the stage for open communications throughout the life of the facility.

4.4 Legal Aspects

Even with the most modern facility using the most stringent safety policies and procedures, legal considerations (litigations) are inevitable. Documentation is needed as evidence of the existence and effectiveness of a comprehensive radiation safety program. All records

should be complete to the extent that they document the patterns of radiation exposure and working conditions at the facility. "Negative" data (survey results which indicate no detectable contamination, air activity, etc.) are as important as positive/detectable data. A complete record of procedures, instruments, calibrations, data, calculations, interpretations, and final evaluations which establish the degree of protection achieved are all valuable in the event of litigation. If an injury or damage is attributable to radiation, a complete record can establish the existence of a comprehensive program and could represent a defense against a charge of negligence, as well as to establish that radiation hazards were under competent control.

4.5 Records

The systematic generation and retention of records relating to occupational radiation exposure are essential to describe the occupational radiation exposure received by workers and the conditions under which the exposures occurred. In addition to the internal dosimetry records, training records, and records related to individual external radiation exposure, radiological conditions under which individuals were exposed, historical records that establish the radiation protection policies and standards of the facility, methods for interpreting and evaluating individual exposure data, medical records, and equipment calibration and maintenance records should be provided. Detailed guidance on radiation exposure records systems can be found in ANSI N13.6-1972, Practice for Occupational Radiation Exposure Records Systems (ANSI 1972). DOE requirements can be found in DOE 5480.11.

Records are the primary source of information used to verify events and trends and are essential in the case of litigation. Records are also essential in audits and for tracking and/or evaluating the progress of the radiation protection program.

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All records should be complete to the extent that they reveal the patterns of radiation exposure and working conditions at the facility. A completed summary table of occupational radiation exposure estimates and sufficient illustrative detail to explain how the radiation exposure assessment process was performed should be included. A description of any procedural changes that were made as a result of the dose-assessment process should be developed. Significant information establishing the radiation status of the facility and personnel exposure must be retained throughout the life of the facility and beyond.

It is suggested that personnel records contain the following information, as a minimum:

a. Worker identification and demographics;

b. Occupational radiation exposure data (current calendar quarter external exposure, lifetime external exposure, and internal exposure);

c. All medical data; and,

d. Worker training.

Computer databases are recommended as an efficient and desirable repository for records. Data manipulation capability is strongly suggested for maintaining extensive trend analyses and reports. Because of the right-to-privacy laws and sensitivity of personal dose records, the security and protection of both computer and hard copy records is necessary.

4.6 Quality Assurance and ALARA

All programs require periodical review/audit to evaluate the implementation and effectiveness of Radiation Protection programs and to ensure that the objectives are being met satisfactorily. Three basic elements of a Radiation Protection program audit can be titled:

a. Current status of the ALARA program

b. Quality assurance

c. Achievement of goals.

4.6.1 Current Status of ALARA Programs

Evaluation of the current status of the ALARA program involves a determination of the program relevancy to current operations and needs. As workloads and functions change, so may the means of achieving ALARA objectives. Therefore, the ALARA program should be evaluated in light of the overall mission of the facility along with specific organizational components. Areas in which most of the plant exposures are received will need special review of the application of ALARA program requirements.

4.6.2 Achievement of Goals

Periodically, the ALARA program should be measured in terms of goal achievement. Goals should be realistic and measurable, the development of which is a management function. Setting measurable goals must be done carefully with considerable thought given to the interpretation of results.

4.6.3 Quality Assurance

Quality assurance (QA) reviews should be scheduled periodically to ensure that the program activities are adequately documented and carried out in accordance with written procedures and policies. These QA audits are a useful way to determine if adequate control of the radiation safety program is being exercised by managers and staff members and to verify that identified deficiencies have been corrected.

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4.6.4 Technical Aspects

The technical aspects of the Radiation Protection program are the compilation, analysis, and evaluation of personnel exposure data. These data include exposure by job category, location, and expected exposure versus that actually incurred. Review of the methods used for personnel dosimetry, instrumentation, standardization, calibration, safety analysis, and design review are some of the areas in which technical reviews and audits can be conducted.

4.6.5 Attributes of Effective Review/Audit

Reviews and/or audits should be conducted as often as necessary with the entire program examined no less than every three years. See specific guidelines in DOE Orders 5480.11 (8.P.) and 5482.1B (9.d.). In addition, random unscheduled reviews and audits of the radiation protection program and its implementation should be conducted.

Reviews and/or audits provide the means to evaluate the effectiveness of the ALARA program through a detailed analyses of the data. Through these analyses, specific areas of 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 radiation protection 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 radiation work has occurred. Such trends should be examined at least quarterly to permit initiation of timely corrective actions. Personnel who perform reviews and audits may be drawn from many sources, but should include line management, professional health physics personnel, and senior management. Occasional use of outside consultants with proven technical expertise is particularly advantageous.

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.

4.7 Administrative Controls

Radiation Protection in a uranium facility is achieved by a combination of administrative and physical controls. Equipment and facility designs can minimize personnel exposure to contamination and/or radiation in the work place. In a new facility, the physical controls can be designed so that administrative controls are minimized, which generally result in a more efficient system. Many current DOE facilities are older facilities in which the ideal does not exist. When the primary controls are administrative, a higher probability of incidents through human error exists.

Administrative controls are achievable through careful documentation of the procedures, materials, and maintenance jobs and a detailed system management and Radiation Protection program control support for the work performed in radiation environs. The objective is to carefully review all

work, establish effective control procedures, assure complete reviews and approval, and execution as planned.

Administrative controls in a uranium facility are designed to limit time in the work place, prevent contamination spread, and limit personnel radiation exposure. The controls may include:

a. Radiation exposure limits;

b. Radiation work permits;

c. Bioassay result levels at which investigations are conducted and workers are restricted from further radiation work;

d. Training required for performing radiation work; and

e. Radiation protection documentation system which allows the tracing of DOE Order requirements from orders to policy, from policy to standards and controls, and from standards and controls to procedures.

The design of the facility and the operations performed will determine the quantity of administrative controls necessary and the rigor with which they must be enforced.

4.8 ALARA at Uranium Processing Facilities

The ALARA concept has wide application and serves as a basis for sound health physics programs. The fundamental ALARA objective is to reduce radiation exposures to the lowest levels commensurate with sound economics and operating practices. Realistic numerical goals can be set and achieved; however, compliance with numerical standards is not *prima facie* evidence that the ALARA concept is fully incorporated in the health physics program. Rather, the success of a mature ALARA program is measured by many

factors including intangibles such as dedication to the concept of dose reduction. 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 as low as reasonably achievable 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.

An ALARA program can be defined in four major elements: program administration, goal setting and program evaluation, radiological design, and conduct of operations. All four elements are vital to the successful implementation of ALARA.

4.8.1 Program Administration

Not only is management commitment to the concept and success of the ALARA program important, but communication to all employees is essential. Policy statements, procedures, and manuals, as well as direct communications, should be used to ensure that facility employees are aware of:

a. The program

b. Their personal responsibility

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c. Their role in its success

d. Personal benefits of the program.

Training in the concept of ALARA, ALARA policy and program, and techniques for its implementation is important and should be provided for specific groups at all levels of the organization. This should include management; and operations, maintenance, design, and the health physics staff. The commitment and cooperation of all are essential to an effective program.

The ALARA function is a responsibility of all management levels but should be a tool of top management. High-level management should formally assign both authority and responsibility for the ALARA program to a specific individual or organizational component and identify management expectations for the program. An independent ALARA review committee may be used to review the program and findings from audits and make recommendations directly to the facility director.

4.8.2 Goal Setting and Program Evaluation

Goals for the ALARA program may be either quantitative or non-quantitative and may or may not be related to dose measurement. All goals, however, should have one or more clearly defined end points which contribute directly or indirectly to reducing personnel exposures. Reducing person-rem by a specific amount in a specific period of time is an example of a quantitative dose-related goal. Increasing staff awareness of ALARA by publishing an internal ALARA communication is an example of a non-quantitative, non-dose-related goal which may indirectly reduce personnel exposure. An ALARA Committee composed of representatives of operations, engineering, radiation protection, and others responsible for the ALARA program can often establish more effective goals than can any one special interest.

In addition to the ALARA program evaluation routinely performed as a function of management of the program, an independent evaluation should be conducted periodically. The evaluation should be commissioned by senior management and the personnel conducting it should report directly to them. It may be appropriate to use an evaluation team if the size of the facility and the extent of radiation work activities warrant. The individual or team members conducting the evaluation should, individually or jointly, have knowledge of and experience in health physics, facility operations, design, management systems, and ALARA. A formal report should be issued to senior management and include an overall assessment of the program, findings of the evaluation, areas of strengths and weaknesses, and recommendations for change and improvement.

There are several measures or indices of performance or the degree of ALARA achievement. Some commonly used measures of achievement are:

a. Mean individual dose equivalents for penetrating and nonpenetrating dose to whole body

b. Specific organ doses from external and internal sources

c. Mean individual dose equivalents by job classification, location, and task

d. Number of workers exceeding administrative dose levels

e. Size of radiation and contaminated areas

f. Effluent release quantities and types

g. Worker training.

Not all performance measures are necessary for all programs, nor is any one or combination of these necessarily suitable for all facilities. Rather,

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those responsible for the ALARA program should decide which measures are most appropriate and the weighing factors to give each one. The ultimate goal of the ALARA program is reducing radiation exposure to levels that are as low as reasonably achievable, or maintaining them at such levels. However, many activities and actions that ultimately affect the radiation dose received may not be directly measurable using dose. These activities and actions are important to the ALARA program and may, in many instances, result in significant dose reductions.

4.8.3 Radiological Design

Design of facility features to accommodate anticipated presence of radioactivity or radiation-generating devices is important in reducing radiation doses received in the conduct of work with radioactive materials to ALARA. For both new facilities and the modification of existing facilities, ALARA considerations should be introduced into the design process at the earliest possible stage. During these early stages of design, incorporation of design features to minimize exposures is most cost and exposure effective. Design engineers should be trained in exposure reduction techniques and ALARA practices. Radiation protection and ALARA personnel should be included in the development of design criteria and review of completed designs to ensure that dose reduction measures have been included and are adequate.

Some of the design features that relate to the successful realization of ALARA objectives are:

a. The layout of the facility; specifically, the use of sequential radiation or contamination zones and the control of traffic patterns;

b. The ventilation system, which should use pressure differentials and high-efficiency particulate air filters to trap airborne radionuclides and prevent the spread of airborne contamination within or from the facility;

c. The waste removal systems, which should minimize the amount of radioactive waste material permitted within a facility, provide storage and handling to control exposure of personnel to casual doses, and minimize the potential for fires, spills, and leakage.

4.8.4 Conduct of Operations

The application of the ALARA principles to the performance of work in the field is the objective. ALARA design, engineering, planning, and administration come to fruition in maintaining radiation exposures to workers and the public as low as reasonably achievable. The operational application of ALARA requires the cooperation and coordination of many functional groups including radiation protection, operations, maintenance, planning and scheduling, training, engineering, and administration.

The primary responsibility for controlling radiation exposure during operations rests with the individual and his or her immediate supervisor. The support functions provide the training, resources, guidance, and measurements but it is in the application that the effectiveness of the ALARA program may be realized. Operational measures for controlling exposure must be applied to assure that any work with radioactive materials is carried out in the safest manner reasonable. Both engineered and administrative control measures should be used for limiting exposure.

As previously stated, engineered controls should be utilized whenever possible. In addition, periodic verification of the continued effectiveness of the engineered controls should be performed. Ventilation and filtration systems should be routinely checked and inspected to assure that operation within the design criteria is maintained. The integrity of shielding, the reliability of equipment, etc., should likewise be routinely verified.

Although administrative controls are not an adequate substitute for engineered features, they are necessary. They are the management systems developed and implemented to provide guidance, direction control, and

limitations for activities. Administrative controls include the documents that describe organizational interfaces and prescribe controls for radiation protection. Administrative control, especially procedures, should be reviewed by those responsible for ALARA to ensure that radiation exposure activities include dose reduction considerations.

In summary, the successful implementation of an ALARA program requires the commitment, support, attention, and efforts of all members of an organization. In facilities in which the radiation exposures are relatively low, implementation of the ALARA concept is particularly challenging. The reduction of radiation doses to as low as reasonably achievable demonstrates to workers and the public a continued emphasis and concern for health and safety.

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SECTION 5

CONTAMINATION CONTROL

The control of contamination in the work place is a significant part of the overall radiation protection program. Contamination control to limit personnel exposure is primarily concerned with minimizing ingestion or inhalation of uranium compounds and controlling external exposure to uranium decay products. Another major objective is to prevent the spread of radioactive materials into uncontrolled areas. The control of contamination can be a valuable part of an aggressive ALARA program since the monitoring and control of contamination provides an indication of the effectiveness of engineering controls and work practices in preventing the release of radioactive material.

This section addresses the basic features of an effective contamination control program and the technical considerations of implementing various program features.

5.1 Air Contamination Control

The primary route of entry of uranium into the body is through the respiratory tract. Release of contaminants from design containments and suspension of radioactive particles into the work place atmosphere result in the principle potential for internal intake.

5.1.1 Internal vs. External Dose Philosophy

It is the policy of DOE to avoid internal exposure of personnel "under normal operating conditions to the extent (reasonably) achievable" (5480.11 89(1)3. A variety of methods are used, including:

a. Containment in process equipment, or handling facilities, i.e., hoods, glove boxes;

b. Conservative use of respiratory protection which dictates the use of protective devices not only when air activity has been measured but when the potential exists due to work on contaminated surfaces, etc.;

c. Isolation of and/or restricted entry to areas of known or potential contamination;

d. Extensive detection and alarm systems for prompt alert to loss of control or increased exposure potential.

The widespread application of these methods in DOE facilities have resulted in a history of relatively minor internal exposures. The majority of significant/reportable exposures to date have been the result of accidental releases/exposures.

A variety of reasons have driven this approach and resultant experience:

1. The assessment of internal dose is difficult, inaccurate, time consuming, and offensive to personnel as compared to external dosimetry. For example, an accidental internal uptake may require:

   a. The subject to submit dozens of biological samples over many months' time,

   b. Extensive analytical support,

   c. Considerable time of trained professionals to analyze data and calculate the internal dose, and

   d. Long time lapse before the dose estimate is available which handicaps the occupational exposure status of the worker.

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2. Prevention of internal exposure is more feasible and successful than for external exposure. Contained radioactive material may continue to produce external penetrating fields of radiation but no internal exposure potential. Protective devices (respiratory equipment) can minimize internal exposure.

3. Recent changes in recommendations of the International Commission on Radiological Protection (ICRP) in formulating a Dose Equivalent limit system results in a combining of internal and external dose. Again, the difficulty and time delay of internal dosimetry make elimination of significant internal exposure an economic incentive.

However, in facilities which handle/process large quantities of uranium there may be situations in which exposure to work place air activity may occur. The fact that tons of material are handled rather than gram quantities and that the material is less toxic (on a gram basis because of low specific activity) make total containment less practical. This difficulty, coupled with worker and management complacency ("it's only uranium") have resulted in a few situations in which routine internal exposures at low levels are experienced. Generally, these situations do not represent "good practice" and should be solved (considering the economics, practicality, and hazards evaluation) consistent with the "no internal exposure" philosophy.

5.1.2 Purpose of Air Monitoring

The primary purposes of most air sampling/monitoring in DOE facilities is to identify, evaluate, and control internal dose received by workers from occupational exposure to airborne radioactive materials, to confirm that source controls are functioning properly and to assess the exposure significance of process upsets. There are two general aspects of air sampling that must receive equal consideration in a properly executed monitoring program. The first involves the methods and equipment, filters, etc., by which a sample is collected and analyzed to yield an accurate

measurement of the specific radionuclides. The second is the procedures/protocol which establishes the sampling location, duration, and frequency which focuses on determination of the radionuclide concentration in the work area.

In order to meet these two purposes three types of samples are collected: General Area Sampling (GAS), Breathing Zone Sampling (BZ), and Personal Air Sampling (PAS).

GENERAL AIR SAMPLERS

Area sampling is performed in the general area of a work site where work with radioactive materials is being performed. These methods are typically used to measure airborne radioactivity for the following purposes:

a. Assure that the work place environments are free of contamination and are inherently safe for routine occupational activities.

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

c. Detect unexpected loss of containment/malfunction of systems and provide the basis to initiate corrective procedures.

d. Detect low level trends in activity which can signal loss of confinement in early stages.

e. Occasionally estimate personnel exposure and/or evaluate compliance with federal orders.

BREATHING ZONE SAMPLERS

Breathing Zone Sampling is performed through a concerted effort to place the air monitor/sampler in the immediate area in which the workers

will spend the majority of their time. The intent is to sample the air and define the air activity concentrations to which the worker(s) were actually exposed. The purposes of BZ sampling are the same as those for GAS with the emphasis on detecting low level trends in activity which can, in the early stages, signal the loss of confinement.

PERSONAL AIR SAMPLERS

Personal air sampling is performed with a small, battery-operated, low volume (approximately 2 ℓ/min) sampler worn by the worker with the filter located near the worker's face. This method more positively defines the concentration of air activity which the worker actually inhales.

Active Versus Passive Monitoring

A further distinction in air monitoring is defined by:

1. Active (constant activity monitoring) and

2. Passive samples (retrospective documentation).

Continuous air monitoring (CAM) provides for immediate alarm and warning workers of an accidental release of high levels of radioactivity. Passive samples collect activity for a period of time after which the sample is analyzed and the activity concentrations calculated, providing retrospective information on what the levels were. Each of these sampling/monitoring approaches are utilized for specific purposes and justification. Active monitoring is essential for high hazard and high potential areas to provide immediate and timely protective response, while the passive sampling provides high sensitivity activity records, trends, etc.

5.1.3 Regulations and Limits

The regulations, standards, and limits pertaining to exposure of radiation workers to air activity in the work place are based on the probability of injury to internal organs and the total body of radioactive materials taken into the body. To facilitate control in the work place, standard setting authorities have calculated Derived Air Concentrations (DAC) and Annual Limits on Intake (ALI) which are designed to limit uptake and resultant dose to internal organs. Operational hazards are directly controlled by the DAC and/or ALI VALUES.

The International Commission on Radiological Protection (ICRP) and the National Council on Radiological Protection (NCRP) are independent, non-governmental organizations which set standards and guidance for control of radiation hazards. The EPA and DOE implement these recommendations by establishing federal policy for the protection of workers within these areas of DOE application. EPA FR 52:17, "Radiation Protection Guidance to Federal Agencies for Occupational Exposure" and DOE Order 5480.1, Chapter 11, "Requirements for Radiation Protection" establish the requirements related to air monitoring. These requirements are:

a. ALARA is an overriding principle requiring monitoring and documentation

b. DAC and ALI guides, as established by ICRP 30, are defacto limits for control of the workplace and require demonstration of compliance

c. Monitoring is required to assure control of potential exposure sources

d. Any workplace area in which >10% DAC could occur must be monitored

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e. Monitoring Systems (ambient) adjacent to work place to permit representative measurements are required

f. Calibration of instruments/samplers to measure 8 DAC-hrs is required

g. Design objective is to avoid inhalation of radioactive material to the extent achievable

h. Areas in which >10% DAC exist must be posted.

5.1.4 Uncertainties and Limitations

A discussion of the uncertainties and limitations should prove useful in placing air sampling/monitoring programs in their proper perspective. In general, air sampling should not be used to estimate internal dose, except in unusual circumstances where bioassay information is unavailable and/or unobtainable. However, evaluation of worker exposure potential utilizing DAC-hrs may be a legitimate control measure and demonstrate compliance with federal directives.

Uncertainties Using Different Air Sampling Methods

An appropriate air-sampling method should provide samples which accurately represent the air volume under study, but should not be used to establish and/or document individual exposures except in unusual circumstances. If air activity data must be used for exposure records, these samples must be collected from the breathing zones of the workers, from a volume of air known to have air concentrations representative of the air actually inhaled by the workers, or a known conversion factor applied. In contaminated areas subject to significant temporal and spatial variations in the activity concentrations, only personal air samples or virtually continuous grab samples collected from within the breathing zone of workers can provide reliable breathing zone (BZ) samples. Almost any

restricted area with good ventilation and with one or several point sources of contamination will have substantial variations in the activity concentrations observed at various locations, particularly if the activities of the workers cause resuspension of the activity. Several researchers have investigated the relationship between fixed air samplers and spot samples collected at various locations in typical working areas, and there is general agreement that variations as great as one or two orders of magnitude are not unusual.

Gonzales et al. investigated the variability of air concentrations around a single release point in a simulated glove box working environment and found that air concentrations in the breathing zone of a worker ranged up to 250 times those recorded by a fixed air monitor located a few feet away. In this experiment, a single glove box was centrally located in a 20 x 20 x 8 ft room and the release point for the test aerosol was a 1/8-in. diameter hole at one glove location. Ventilation of the room ranged from 6 to 12 air changes per hour.

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

Breslin et al. conducted a study in a uranium extrusion plant and concluded that reliable exposure estimates could be obtained using GAS methods coupled with careful time studies of working patterns. Breslin conceded that the effort required would be prohibitive for routine applications.

The potential for release of gaseous UF_6 and subsequent generation of its soluble hydrolysis product (UO_2F_2) necessitates changes in the air-sampling objectives in uranium conversion and gaseous diffusion plants relative to those plants handling less reactive uranium compounds. In

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these plants, effective processing, as well as worker safety, requires a fairly high degree of containment. Continuous general area air samples to detect loss of containment in the process, coupled with spot air samples constitute the typical sampling strategy. A 1963 study by Schultz and Becher, conducted at the Oak Ridge Gaseous Diffusion Plant, concluded that shift-long air samples collected in the general working areas of a gaseous diffusion plant were virtually useless in predicting worker's urinary uranium excretion. The slight correlation observed was not statistically significant at the 95% confidence Level. These researchers also found that smear samples of alpha activity on work surfaces in the area may provide a better indicator of uranium uptake than the GAS records. This finding is similar to Kruger's observations in uranium mills.

Although transuranic material is handled by DOE Uranium facilities only as feed contamination, the unusual characteristics of the transuranic elements make them worthy of separate consideration. The low maximum permissible concentrations specified for these elements and their frequently low specific activities cause extreme difficulties in the detection of significant airborne activity. These radioisotopes and their daughters generally produce little penetrating radiation, and are not easily measured by a whole body count (WBC) at levels less than a body burden. The maximum permissible concentration (MPC) for soluble plutonium (DAC Pu = $2 \times 10^{-12} \mu ci/m^3$) corresponds to the activity produced by a single particle of pure plutonium only 1.6μ in diameter per cubic meter of air. Consequently, replicate measurements of airborne activity of plutonium and other transuranic elements can be quite variable, even when relatively large sample volumes are considered. This in itself can cause difficulty in obtaining representative samples for estimating individual exposures. Operations involving significant amounts of plutonium should be conducted in a ventilated glove box environment and with monitoring and systems capable of detection of small unsuspected releases involving a few times one DAC. Special alpha CAMs (GAS) and fixed BZ samplers are the standard air-sampling methods used in United States' facilities of this category.

Another serious deficiency of GAS monitoring for individual exposure records can be traced to the high dilution factors that tend to reduce the air concentrations before the contaminated air reaches the filter head. Most restricted areas where contamination can occur are well ventilated by several air changes per hour. A release of activity due to a malfunction of the containment system can produce large activity concentrations at the BZ of a worker. These concentrations can be diluted in an unpredictable manner by one or two orders of magnitude before the contamination reaches a monitor located only a few feet away. Further, it has been demonstrated that in some operations (such as welding over a short time) period differences of as much as a factor of 5 between the right and left lapel PAS samplers can be expected.

An excellent example of the wide variations in observed air-activity concentrations that can occur with different sampling techniques is provided by data from the Three Mile Island Nuclear Generating Station which is typical of operations in a large open building.

Between June and September 1983, over 40 multiperson entries were made into the containment building providing 949 manhours of PAS data. Five stationary air monitors were operated continuously at strategic locations throughout the building, and each entry was preceded by the collection and analysis of a high volume grab sample. All samples were analyzed by a gamma spectrometer, primarily to detect cesium-137, and by gross beta counting. A graph of the average air-activity concentrations determined by gross beta counting by each of the three sampling methods is shown in Figure 5-1.

The five continuous air samples exhibited good internal agreement when averaged over either 12- or 24-h periods indicating little differences in the averaged readings for the separate locations. However, the continuous air-sampler readings averaged a factor of 3 lower than the grab samples and a factor of 34 below the personal air-sample averages collected in the BZ of workers. The major reason for this large difference was attributed to

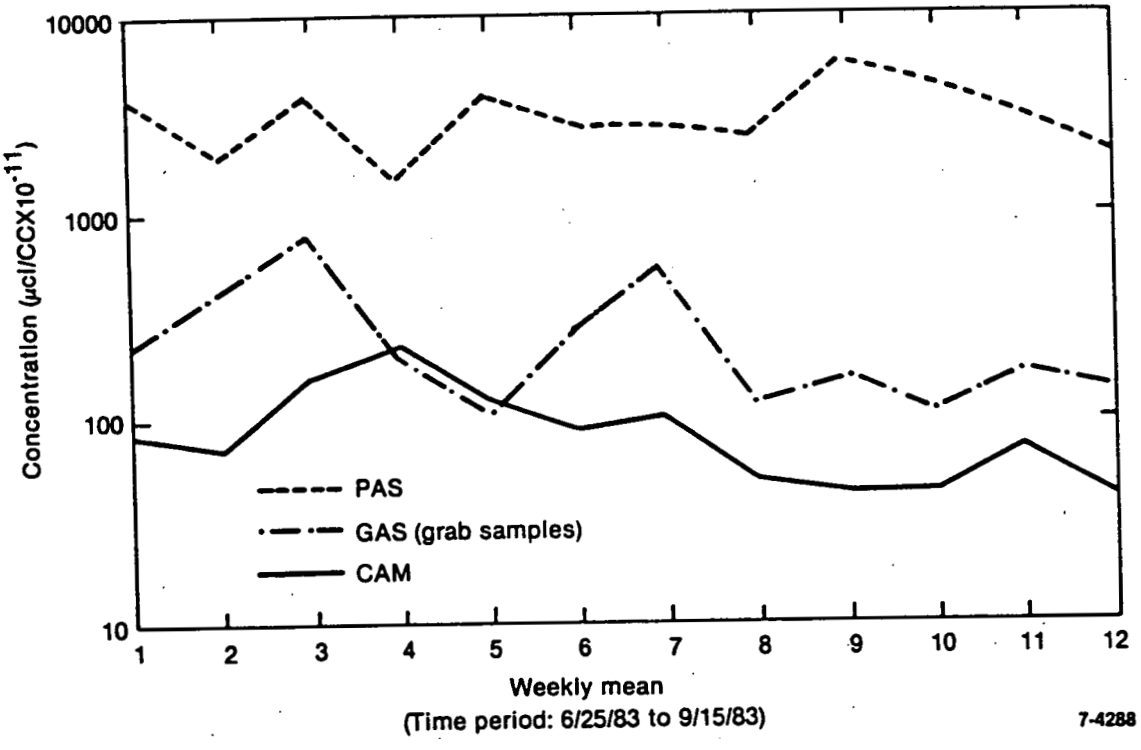

Figure 5-1. PAS vs GAS vs CAM example of the degree of correlation between type of sampling TMI-1983.

resuspension of the surface contamination by the work in progress. These data, coming from a thoroughly monitored and carefully analyzed air-sampling effort, are further evidence that GAS methods should be viewed with caution.

Other Limitations

Although they play similar roles, there may not be an equivalence or fixed relationship between breathing-zone sampling and bioassay. It is usually not possible to accurately estimate individual uptake or the resulting internal dose, from an air activity exposure estimates. It is also difficult to accurately estimate previous internal uptake from bioassay measurements.

Even when the air-activity concentrations in the BZ workers have been reliably determined, there are other physical and physiological parameters which can produce large uncertainties in dose assessment. The established DAC are derived for each radionuclide assuming a standard volume of air breathed in occupational situations, specified pathways to critical organs, the "standard man" metabolic and elimination patterns, and the physical and biological properties of the isotope.

While these generally conservative assumptions are justified when calculating an appropriate MPC or DAC for control purposes, the actual internal dose received by an individual worker will depend on how closely the actual physical parameters of the aerosol and physiological parameters of the worker corresponded to the standard conditions used to calculate the DACs. Large variations are encountered in breathing rates and tidal volumes (which depend on working conditions), and there are individual variations in such physiological parameters as lung clearance and metabolic rates. The particle-size distribution of the aerosol and the actual solubility of the inhaled aerosol can significantly affect the deposition and retention of airborne activity in the respiratory tract. Obviously, the potential uncertainty in the total dose assessment cannot be less than the uncertainty in any one of these parameters.

Particle-Size Distribution

In the absence of actual measurement of particle-size distributions, an activity median aerodynamic diameter (AMAD) of 1 μ and a geometric standard deviation (GSD) of two is often assumed as a conservative estimate, per ICRP 30 methodology. Size-selective and size-measuring inlets for air samplers have been developed to give more accurate estimates of deposition. Cascade impactors can provide information on the AMAD and the GSD of airborne contamination. As yet, the use of impactors for continuous activity measurements in the BZ, in an attempt to catch the brief and unexpected release of activity that often accounts for an individual exposure, is of unproven practicality in most work situations.

Size-selective inlets (e.g., cyclones) have been designed for both GAS and PAS equipment which separate the nonrespirable fraction of airborne dusts and allow collection of the respirable fraction on a filter. These devices could be useful for minimizing the dose assessment errors resulting from lack of knowledge of the actual aerosol-size distribution; however, they are expensive and require additional handling and care. They do not provide a simple means for estimating total airborne activity.

Solubility

The health physicist may have some prior knowledge of the solubilities and/or chemical compounds of the nuclide or nuclides in question and be willing to classify as D, W, or Y. The new ICRP 26 dosimetry model provides for a lung class designation of aerosols depending on the rate of dissolution; however, actual determination of the lung class for dose assessment can probably be determined only after the exposure utilizing appropriate chemical and/or bioassay data.

Breathing Rates/Tidal Volumes

The actual air intake of a worker can vary from 5 L/min to 100 L/min, although average variations from the assumed 20 L/min standard will probably be no larger than a factor of 2 or 3. The air intake depends both on the rate of breathing and on the volume of air exchanged (tidal volume).

Deposition/Clearance

While deposition fractions in the various lung compartments depend primarily on particle-size distributions, these can change somewhat with breathing rates and tidal volume, and may vary greatly from individual to individual. Lung clearance and metabolism can depend on the general state of health and on physical activity, perhaps to a factor of two or more.

Since these physical and physiological parameters are normally not measured as part of the air-sampling program, accurate air-activity measurements alone are not adequate for reliable individual dose assessment. It may be argued that conservative dose estimates can be made from reliable BZ air-sampling measurements (with uncertainties of perhaps a factor of about five), but such estimates should not be considered adequate for compliance with the federal regulations when significant air-activity concentrations are encountered.

5.1.5 Samplers and Instrumentation

Air sampling equipment and monitors exist in a wide range of designs and capabilities, the characteristics of which generally being specific to the application and need. Samplers range from small portable units that can be worn by a worker to large, high volume units permanently mounted in the facility. Flow rates (as well as the preoperational detection limit or sensitivity range) from a few liters per min to a few cubic meters per min are used.

Key factors in selecting air samplers and/or instruments are:

1. Sensitivity of detection needed

2. Type of sample to be collected

3. Convenience (size, weight, ease of operation, etc.)

4. Power requirements

5. Accuracy required

6. Reliability and maintainability.

Sensitivity

In general the sensitively required is at least DAC levels, however, in some applications small fraction of DAC is desired to detect early loss of control, low level trends, etc. Also constant and alarming air monitors may only need to alarm at multiple DAC levels in order to be effective in preventing or mitigating personnel exposures to accidental airborne releases.

Type of Sample

In most uranium facilities particulates in the air are the concern, although gaseous constituents may be of occupational interest. It may be of interest to collect samples which will allow characterization of the particle size distribution or define a "respirable fraction." In each application the samples type will dictate the sampler design, filter media, etc.

Convenience

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

Power Requirements

Battery powered vs 110-220 line power requirements are obvious criteria which dictate sampler/instrument design.

Accuracy

Some sampling is performed to "detect" or make relative measurements (is the activity rising or falling or is "any" escaping) and the accuracy requirements are not great. In other situations the need for accurate measurements of the air breathed by personnel impress an entirely different sampler design on the situation.

Selection and Use

Continuous monitors provide a real-time measurement and real-time record of air activity concentrations. Typically these monitors utilize pumps or air movers in the range of a few cfm and appropriate detectors, electronic analyses and recorders to provide visual and audible indication and alarms. They are generally used in the GAS mode.

Passive or cumulative air samplers collect samples at a specific desired flow rate for a period of time after which the filter is removed for counting and/or analyses.

Filters

Filters should have high collection efficiencies (i.e., >99%) for particles over a wide range of sizes. Table 5-1 lists particle-collection

efficiencies in the particle size range of lowest efficiency for most filters. Most cellulose ester (acetate, nitrate, or mixed ester) or fiber glass filters meet these requirements and are commonly available. Other filters with reasonable high collection efficiency may be used if required for special applications or assay methods. Selection of a filter type generally involves compromises between filter efficiency, resistance, and requirements imposed by the desired assay method.

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

b. Fiber glass filters are made of a mat of randomly oriented glass fibers. They have lower flow resistance than most membrane filters, and trap an appreciable fraction of the particles within the filter mat. Table 5-1 also presents data on filter permeability, e.g., flow resistance.

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

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

TABLE 5-1. SUMMARY OF FILTER CHARACTERISTICS

Filter	Material	Pore Size (mm)	Filter Permeability Velocity, cm/s (ΔP = 1 cm Hg)	Filter Efficiency Range (%)[a]
Cellulose Fiber				
Whatman				
No. 1	Cellulose	—	6.1	49 to 99.96
No. 2	Cellulose	—	3.8	63 to 99.97
No. 3	Cellulose	—	2.9	89.3 to 99.98
No. 4	Cellulose	—	20.6	33 to 99.5
No. 5	Cellulose	—	0.86	93.1 to 99.99
No. 40	Cellulose	—	3.7	77 to 99.99
No. 41	Cellulose	—	16.9	43 to 99.5
No. 42	Cellulose	—	0.83	92.0 to 99.992
Glass Fiber				
Gelman				
Type A	Glass fiber	—	11.2	99.92 to >99.99
Type A/E	Glass fiber	—	15.5	99.6 to >99.99
Spectrograde	Glass fiber	—	15.8	99.5 to >99.99
Microquartz	Glass fiber	—	14.1	98.5 to >99.99
MSA 1106B	Glass fiber	—	15.8	99.5 to >99.99
Paliflex				
2500 QAO	Quartz fiber	—	41	84 to 99.9
E70/2075W	Quartz fiber	—	36.5	84 to 99.95
T60A20	Teflon coated glass fiber	—	49.3	55 to 98.8
(another lot)	Teflon coated glass fiber	—	40.6	52 to 99.5
T60A25	Teflon coated glass fiber	—	36.5	65 to 99.3
TX40H12D	Teflon coated glass fiber	—	15.1	92.6 to 99.96
(another lot)	Teflon coated glass fiber	—	9.0	98.9 to >99.99
Reeve Angel 934AH	Glass fiber	—	12.5	98.9 to >99.99
(acid treated)	Glass fiber	—	20	95.0 to 99.96
Whatman				
GF/A	Glass fiber	—	14.5	99.0 to >99.99
GF/FB	Glass fiber	—	5.5	>99.99 to >99.99
GF/C	Glass fiber	—	12.8	99.6 to >99.99
EPM 1000	Glass fiber	—	13.9	99.0 to >99.99
Plastic Fiber				
Delbag Microsorban-98	Polystyrene	—	13.4	98.2 to >99.99
Membrane				
Millipore				
MF-VS	Cellulose acetate/nitrate	0.025	0.028	99.999 to >99.999
MF-PH	Cellulose acetate/nitrate	0.1	0.16	99.999 to >99.999
MF-PH	Cellulose acetate/nitrate	0.3	0.86	99.999 to >99.999

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TABLE 5-1. (continued)

Filter	Material	Pore Size (mm)	Filter Permeability Velocity, cm/s (ΔP = 1 cm Hg)	Filter Efficiency Range (%)[a]
Membrane (continued)				
Millipore (continued)				
MF-HA	Cellulose acetate/nitrate	0.45	1.3	99.999 to >99.999
MF-AA	Cellulose acetate/nitrate	0.8	4.2	99.999 to >99.999
MF-RA	Cellulose acetate/nitrate	1.2	6.2	99.9 to >99.999
MF-SS	Cellulose acetate/nitrate	3.0	7.5	98.5 to >99.999
MF-SM	Cellulose acetate/nitrate	5.0	10.0	98.1 to >99.99
MF-SC	Cellulose acetate/nitrate	8.0	14.1	92.0 to >99.9
Polyvic-BD	Polyvinyl chloride	0.6	0.86	99.94 to >99.99
Polyvic-BD	Polyvinyl chloride	2.0	5.07	88 to >99.99
PVC-5	Polyvinyl chloride	5.0	11	96.7 to >99.99
Celotate-EG	Cellulose acetate	0.2	0.31	>99.95 to 100 >99.999 to 100
Celotate-EH	Cellulose acetate	0.5	1.07	99.989 to >99.999
Celotate-EA	Cellulose acetate	1.0	1.98	>99.99 to >99.99
Mitex-LS	Teflon	5.0	4.94	84 to >99.99
Mitex-LC	Teflon	10.0	7.4	62 to >99.99
Fluoropore				
FG	PTFE-polyethylene reinforced	0.2	1.31	99.90 to >99.99
FH	PTFE-polyethylene reinforced	0.5	2.32	99.99 to >99.99
FA	PTFE-polyethylene reinforced	1.0	7.3	99.99 to >99.99
FS	PTFE-polyethylene reinforced	3.0	23.5	98.2 to 99.98
Metricel				
GM-6	Cellulose acetate/nitrate	0.45	1.45	99.8 to >99.99
VM-1	Polyvinyl chloride	5.0	51.0	49 to 98.8
DM-800	PVC/Acrylonitrate	0.8	2.7	99.96 to >99.99
Chemplast				
75-F	Teflon	1.5	3	83 to 99.99
75-M	Teflon	1.0	6.6	54 to >99.99
75-C	Teflon	1.0	32	26 to 99.8
Gelman Teflon	Teflon	5.0	56.8	85 to 99.90
Ghia				
S2 37PL 02	Teflon	1.0	12.9	99.97 to >99.99
S2 37PJ 02	Teflon	2.0	23.4	99.89 to >99.99
S2 37PK 02	Teflon	3.0	24.2	92 to 99.98
S2 37PF 02	Teflon	10.0	—	95.4 to >99.99
Zefluor				
P5PJ 037 50	Teflon	2.0	32.5	94.6 to 99.96
P5PI 037 50	Teflon	3.0	31.6	88 to 99.9

The surface-particle collection properties of membrane filters can be an advantage when sampling for alpha and weak beta-emitting materials, since this minimizes self-absorption of the combined filter/particulate counting sample. Membrane filters are especially useful if the sample is to be analyzed by alpha spectrometry, since surface collection of particles reduces degradation of energy resolution. Membrane filters are also advantageous if the assay procedure for the sample involves ashing or dissolution of the filters.

Table 5-1 information on filter permeability may be used to compare filter resistances for different filter materials and sizes as follows:

$$\Delta P = \dot{V}\left(\frac{cm^3}{s}\right) \cdot \frac{1}{A_f}\left(\frac{1}{cm^2}\right) \cdot \text{Perm}\left(\frac{cmHg \cdot s}{cm}\right)$$

where

$\dot{V}$ = volume flow through the filter

A_f = the area of the filter through which the sample passes. This will be a fraction of the total filter area. This fractional area will be constant for any specific filter holder, so resistance for different filters can be compared directly using total filter area, provided either would be used in the same filter holder

Perm = the filter permeability from Table 5-1

cmHg = pressure drop in centimeters of mercury.

Filter Holders

Criteria for filter holders are simple but critical. For the collection of large volume work room air samples, filter holders should be open-faced such that sample air is drawn directly onto the filter surface from the atmosphere without passing through a tube, orifice or other obstruction. This precludes loss of the radionuclide to surfaces upstream from the filter. However, closed-face cassettes are recommended for small PAS. Research studies of commonly available types of closed-face cassettes with 4 mm inlets indicate that these designs have good particle-collection characteristics (at a flow rate of 2 L/m), in addition to reducing sample contamination problems. Other closed-face filter inlet diameters, geometrics, and flow rates may also be acceptable, but have not been characterized. See Figure 5-2 for a typical closed-face cassette.

According to present theory the diameter of the filter opening should be held within prescribed limits, depending on flow rate, to avoid biasing particle collection by the effects of sedimentation and inertia. For the range of more commonly used flow rates, 5 to 50 ℓ/min, the theory indicates proper diameters to be on the order of 2 to 5 cm which are consistent with sizes that are likely to be selected to satisfy other criteria. However, the theoretical treatment applies to calm air, which is atypical of most occupational environs, and it has not been confirmed experimentally. Therefore, these biasing effects have not been shown to be of practical significance in routine circumstances.

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

Filter holders may be made of various metals or plastics. Generally, metallic holders are more reliable and durable.

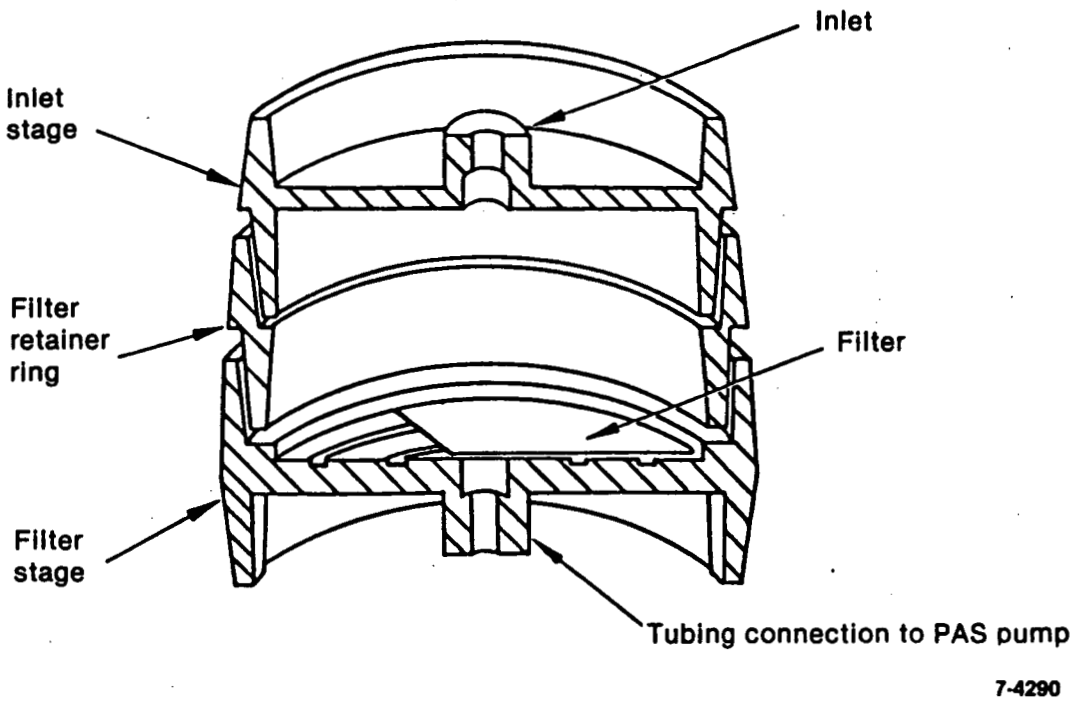

Figure 5-2. Typical closed-face cassette.

Size-Selective Devices

Size-selective devices fall into two categories: respirable fraction collectors and devices for measuring complete particle-size distributions. Respirable fraction collectors are generally simpler to use, readily available, and have well-characterized collection and separation properties, since the mining industry, National Institute of Occupational Safety and Health (NIOSH), and Mine Safety Health Administration (MSHA) have done considerable research and development in this area.

Particle-size distribution measuring (particle-sizing) devices are typically more complex and require more sample analysis. The major advantage in using these devices is that the size distribution of airborne contamination is useful information for estimating regional deposition of inhaled particles in the respiratory tract. Particle-size measurement should be performed only by properly trained individuals, as an investigative tool for evaluating the health hazard posed by a process and/or procedure suspected of generating airborne contamination.

Radiological safety workers may wish to use respirable fraction-size selectors, if they intend to use PASs solely for more accurately estimating individual uptake, particularly if the work site is dusty and activity is distributed over a wide range of particle sizes (for example, airborne material in a uranium milling plant).

Measurement of only the respirable fraction of airborne contamination will reduce sensitivity to larger airborne radioactive particles. Unfortunately, these larger particles may be of interest for a more general evaluation of the quantity of airborne contamination generated by the worker's activities. Total airborne-activity measurements are probably more useful in evaluating control of airborne contamination, and are also satisfactory for conservative estimation of individual exposure.

a. Particle-sizing devices. The cascade impactor is the most commonly available particle-sizing device. Air samples passing through a cascade impactor are forced through a series of increasingly rapid changes of velocity. The inertia of the

particles causes them to deviate from the direction of the airstream at locations where the particles' speed and direction are changing most rapidly. Particles of different aerodynamic size deflect to different extents so that some contact surfaces are impacted. Impactors are designed so that most (ideally all) impaction occurs on sample collection (i.e., impaction) surfaces at each stage. This is usually accomplished using carefully designed air jets directed at impaction surfaces at each stage. The quantity of material in a particular size range can be estimated by examining the material deposited on the impaction surfaces at each stage.

There are some limitations and drawbacks to use of impactors. Cascade impactors subdivide the sample, so that more sensitive assay methods may be required for successful use. There is a limit to the mass of material which can be collected on each stage before overloading; inactive dust is always present and may limit useful sampling time. Each impactor stage (i.e., stage in the cascade) is a separate sample which must be analyzed. This multiplies capacity requirements of the sample-counting system. Impactors (and most other aerodynamic size selectors) are sensitive to the airflow characteristics which are compatible with their products, or the impactor should be calibrated for particle-sizing characteristics with the PAS pump which will be used for sampling. This calibration requires equipment and facilities which are not available to most safety personnel. There are no standards yet for impactor performance. Users who intend to use cascade impactors should consult the literature on the topic.

b. Respirable Fraction Samplers. A number of respirable fraction samplers have been developed, but the cyclone separator is the most extensively used and characterized type, since it is specified by NIOSH/MSHA for personal respirable mass sampling in coal mines. NIOSH and MSHA currently certify entire sampling systems (PAS pump, cyclone, filter head, and filters) for

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personal respirable fraction sampling. This "system" approach may be modified as the result of recent research; however, it does provide an interim standard for performance. The performance of cyclones, pumps, and filters may be characterized to allow intermixing of sampling-train components in future work; but, at present, theoretical prediction of performance of mixed systems is not reliable.

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

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

5.1.6 Sample Analyses

Most sample analyses at uranium facilities are performed by quantifying the radioactivity on the samples collected. There are some fluorometric analyses performed with equivalent sensitivity.

Alpha Counting

Alpha particles can be counted with ionization, proportional, scintillation or other solid state detectors. The major draw back is that relatively little particle penetration in the filter or dust loading can result in low reading due to self absorption of the alpha particles.

Alpha Spectrometry

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

Beta Counting

Thin window G.M., ionization, proportional and solid state detectors are used for beta counting. Due to the wide range of beta particle energies of even a "single energy" emitter, careful energy calibration is necessary. Beta counting is less dependent upon self absorption.

Beta Spectrometry

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

Gamma Spectrometry

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

Precautions

The intricacies and procedures of sample analysis is beyond the scope of this manual. However, one or two general precautions are important to mention.

The naturally-occurring radionuclides, radon and thoron and their decay products, are present in all atmospheres in widely varying concentrations. These radionuclides are typically in higher concentrations

than the isotopes of interest and, tend to interfere with radiometric analysis unless they are given time to decay after sample collection. Radon daughters, which are much more abundant than thoron daughters in most areas, decay with an effective half-life of about 30 minutes and a counting delay of three hours may be adequate. Thoron daughters decay with an effective half life of 10.6 hours and where they exist in significant concentrations, a counting delay of several days is advisable. The presence of either radionuclide on a filter can be detected by recounting two or three times at intervals of a few hours.

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

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

The contaminated or upstream sides of filters collected in clean atmospheres are often difficult to identify and some convention that is understood by the analytical staff should be followed by sampling personnel to assure that the proper sides of filters will be counted. This may consist of marking the exposed side of the filter or placing the filter in the sample envelope consistently with the exposed side toward the identifying number or label on the envelope or marking the unexposed side of the filter.

5.1.7 Monitoring Strategies and Protocols

Designing an air-sampling program for the work place is a complex task since each facility has unique environments for the generation of airborne activity. It is important that the radiation safety personnel coordinating the sampling program have a thorough understanding of the basic operations at their facility, especially with respect to the potential each operation has for generating airborne material. In addition, these personnel should be familiar with the working habits of potentially exposed workers. The success of most sampling programs depends on a proper selection of workers to be sampled, and, therefore, on the ability of radiation safety personnel to accurately assess workers' risk of exposure. This can only be accomplished by well-trained, observant safety personnel.

Two basic questions should be considered for an airborne activity hazard evaluation:

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

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

Potential Sources of Airborne Contamination

Virtually every worksite has at least one of the fundamental mechanisms for the generation and suspension of particulate material. The following descriptions of some of the basic mechanisms of aerosol generation, are intended to help radiation safety personnel recognize processes which have inherently higher risk. Vohra compiled a list of the more common mechanisms for aerosol formation in nuclear industries. They are:

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

b. Combustion. Burning material produces smoke, fumes, etc.

c. Heating. Many materials produce aerosols when heated, without actually igniting.

d. Formation from bubbles, foams, or highly agitated liquids. Fine solid particles can form from larger, evaporating liquid droplets.

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

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

g. Formation of solid, radioactive nuclides from gaseous parent nuclides. These radionuclides usually attach to existing, nonradioactive aerosol particles.

h. Adsorption of gaseous, radioactive nuclides on nonradioactive aerosols.

A similar list developed by Leidel et al. for identifying industrial processes which may be sources of airborne particles is shown in Table 5-2. this information is applicable in many nuclear industries.

All potential sources of airborne contamination should be identified as a first step in developing a safety program.

Worker Routine and Procedure Evaluation

The program designer should be familiar with the routines and working habits of workers, especially those in situations were there is a greater

TABLE 5-2. SOURCES OF AIRBORNE PARTICULATES

Hot Operations

- Welding
- Chemical reactions
- Soldering
- Melting
- Molding
- High temperature solidification
- Burning
- High temperature cutting

Shaping Operations

- Cutting
- Grinding
- Filing
- Milling
- Molding
- Sawing
- Drilling

Solid Operations

- Screening
- Pouring
- Mixing
- Extraction from solids
- Crushing
- Conveying
- Loading
- Bagging
- Sieving

Liquid Operations

- Painting
- Degreasing
- Dipping
- Spraying
- Brushing
- Coating
- Etching
- Cleaning
- Dry cleaning
- Pickling
- Plating
- Mixing
- Galvanizing
- Solvent extraction
- Electrolytic reactions
- Other chemical reactions

Pressurized Spraying

- Cleaning parts
- Degreasing
- Sand blasting
- Painting
- High pressure jet cleaning

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potential for generating locally high concentrations of airborne contamination. This will assist in planning for prevention of exposure, and in selection of suitable sampling methods for estimating individuals' workplace control.

- Worker location and mobility. If the worker stays in a fixed location, fixed breathing-zone sampling may be useful for individual exposure estimation. This sampling may be performed using moderate flow rate pumps (1 to 3 cfm) which can be located within a few feet of the worker. Mobile workers should be surveyed using PAS to obtain a BZ sample.

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

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

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

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

Characterization

For the purpose of evaluating workplace controls, worksites can be characterized as either "tightly controlled" or "loosely controlled." Some of the characteristics of tightly and loosely controlled work areas are described in Table 5-3. Tightly controlled work areas are preferred in all cases, but there are situations where good control is difficult or not reasonably achievable. PAS surveys can help define those operations which pose the greatest radiation control problems and, thereby assist in decisions to improve specific work situations.

Jones has noted that significant exposure incidents in highly controlled, i.e., tight, areas usually are the result of isolated and unforeseeable events, which are complete departures from the normal material processing routine. These events usually include loss of containment. In tightly controlled areas, PAS can serve as a means for detecting a failure of containment, since work locations may be located near potential release points, and inadequate physical controls may be apparent only during an operation performed by a worker. Tamas's observations on the usefulness of PAS in the field are an example of this. Her experience with PAS at a ^{238}Pu-handling facility (glovebox-type containment) was that in those circumstances, PAS was the single most effective means of initially detecting a loss of containment resulting in airborne contamination. It would have been difficult to predict this finding without field studies, but the results indicate that safety workers may find uses for PAS in addition to exposure estimation.

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TABLE 5-3. CHARACTERISTICS OF TIGHTLY AND LOOSELY CONTROLLED WORK PLACES

Process or Work Place Characteristic	Tightly Controlled Work Places	Loosely Controlled Work Places
Manipulation of radioactive material	Remote handling of radioactive material	Hands-on procedures for working with radioactive material
Distribution of radioactive material through the work area	Restricted to a few work stations	Widely distributed, e.g., in process flows or at numerous work stations
Worker mobility among potential release locations	Low mobility. Workers are restricted to one or a few work stations and tasks	High mobility. Workers move throughout the work area and perform a variety of tasks
Ventilation of the work area	Air flow in the area is stable and well controlled (This does *not* indicate uniform concentration of airborne activity.)	Air flow in the area is unpredictable. Little effort is made to control air movement, or efforts are ineffective
Surface contamination levels	Surface contamination maintained at relatively low levels, and contaminated areas are well defined. Prompt cleanup of surface contamination occurs following detection	Relatively high surface contamination levels. Easily suspended contaminants in poorly defined or random distribution in the work area
Physical properties of radioactive material being processed	Material is tightly bound to a substrate; as an inert, nonfriable solid	Friable, and/or reactive solids; liquids and gases

As a general rule, loosely controlled operations result in a greater likelihood of exposure of individuals to high local concentrations of airborne material as a result of conditions which represent an undesired but foreseeable consequence of some normal working situation. These releases may not be reliably detected by GAS, let alone quantified.

5.2 Surface Contamination Control

5.2.1 Plant Surfaces

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

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

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

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

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

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

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

5.2.2 Reporting/Documenting Contamination Levels

Contamination levels are reported in a variety of units (e.g., cpm/sq ft, cpm/100cm2), but should be reported in activity per area (e.g., Bq/m2, dpm/100 cm2). This permits interpretation of the recorded data without requiring knowledge of instrument efficiency or geometry. Contamination control limits are generally specified in terms of "fixed" or "removable" contamination. In many contamination applications, the fixed contamination can be five times the removable contamination levels, as measured by wiping the surface with an absorbent material with moderate pressure. Although this can be a useful generalization as "rule of thumb" scoping the hazard potential implied by contamination monitoring data each situation may be different. The relative proportion of removable versus fixed contamination is obviously a function of the chemical and physical characteristics of the uranium compounds in question.

5.2.3 Characteristics of Uranium Contamination

The characteristics of uranium contamination are a function of the physical and chemical properties of the uranium compounds. Additional factors involved in the determination of surface contamination are the radiological properties of the contamination, i.e. the isotopic composition.

The chemical and physical characteristics of uranium contamination encompass a wide range of solubility classifications and particle sizes, significantly influencing the dispersibility and thus the transferability of the contamination. For example, uranium hexafluoride, when released from process equipment, reacts with moisture in the air to form a finely dispersed aerosol of uranyl fluoride and hydrogen fluoride, which is thermodynamically buoyant. The released uranium settles as a very fine dust over a large area and is easily dispersible and transferable and presents entirely an alpha radiation hazard since the non-volatile daughters are not released. On the other hand, the soluble uranyl nitrate that is a product of uranium recovery processes is normally associated with aqueous solutions and is normally not as easily dispersible in the event of a leak from process equipment. However, dusts from drying spills and

insoluble oxides of uranium may be of small particle size and thus be very dispersible. Each process should be evaluated to characterize the type of contamination present since this will dictate the contamination monitoring procedures and techniques as well as the interpretation of the monitoring data.

Physical factors which can affect the form of the uranium contamination available are demonstrated by activities such as grinding, machining, buffing, etc., of surfaces which generate airborne activity in many cases even though the chemical form was not dispersible. The range of particle sizes from these operations is large. Some are respirable while others are too large to reach the broncheal or pulmonary regions of the lung, but even those could present an ingestion hazard.

The radiological characteristics of the uranium compounds also influence the hazard potential. The U-235 enrichment of the uranium will change the concern level of the contamination monitoring data. As shown in Table 5-4, the specific alpha activity increases by a factor of more than two orders of magnitude as the U-235 enrichment increases to 90%; this is primarily due to the increase in the fraction of the U-234 present. Also, Table 5-4 shows that, the relative contribution of uranium daughter beta activity to the alpha activity decreases significantly as the enrichment increases. Though the radiological hazard from the enriched uranium is significantly greater than for natural or depleted uranium, the relative chemical toxicity hazard remains constant. Therefore, the range of enrichment that may be encountered should be considered in establishing contamination limits. Alpha to beta ratios as illustrated in Table 5-4 provide a field method for evaluating the radiological characteristics of the contamination. Alpha monitoring is often the method of choice for recently separated and highly enriched uranium. A graph comparing the alpha/beta ratio for various U-235 enrichments is shown in Figure 5-3.

TABLE 5-4. CALCULATED ALPHA AND BETA ACTIVITY OF URANIUM IN EQUILIBRIUM WITH INITIAL SHORT-LIVED DAUGHTERS

% U-235 Enrichment	Alpha DPM Per Microgram	Beta DPM Per Microgram
0.7	1.3	2.1
1	1.6	2.1
10	12.5	2.0
20	23.6	1.9
30	32.3	1.8
40	43.0	1.7
50	53.7	1.6
60	64.9	1.4
70	79.4	1.3
80	99.3	1.2
90	127.8	1.1

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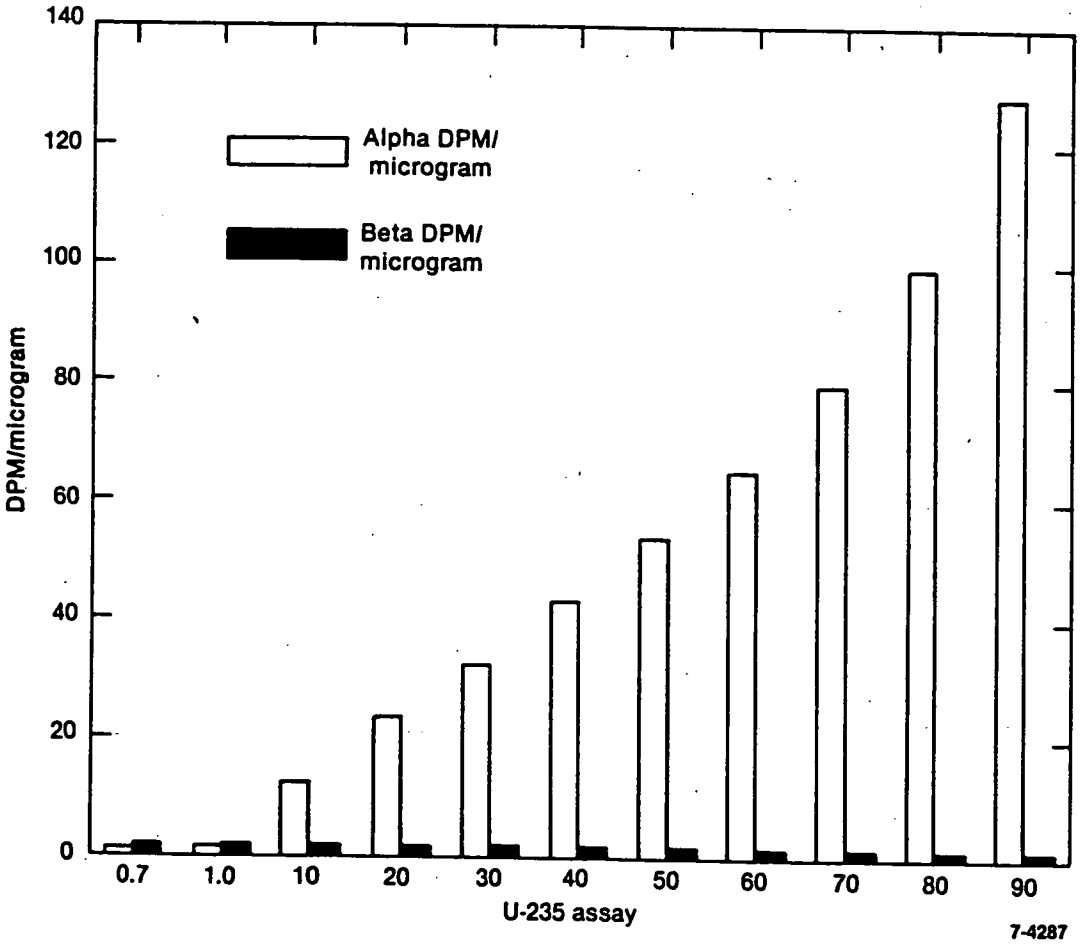

Figure 5-3. Alpha beta ratios vs enrichment.

Contaminants

Another factor that must be considered in evaluating contamination levels and establishing limits is the potential presence of contaminants from recycled uranium. In the USA fuel cycle, there is little, if any, "virgin" uranium and fission products and/or transuranics contamination will be expected. The uranium will usually be the controlling hazard; however, the process should be evaluated and analyses performed for various recycled uranium (RU) contaminants, since a chemical or physical process may concentrate RU contaminants in certain equipment or process streams. For example, the gaseous diffusion process concentrates the volatile compounds of the fission product Technetium in the high enrichment gas stream. In another case, various RU contaminants will be concentrated in various waste streams of uranium recovery operations (e.g., fission products will concentrate in the raffinates from the solvent extraction process while transuranics may remain with the uranyl nitrate, depending on the recovery process utilized).

Uranium decay products may be present in various quantities relative to uranium depending on the age of the uranium contamination. Figures 5-4 and 5-5, depict the time required for freshly separated uranium to achieve equilibrium with various decay products. Figure 5-6 provides the decay scheme for U-235 and U-238 listing the decay products that may be encountered in some quantity during typical ages of uranium compounds at DOE facilities. In general, the uranium decay products exist as compounds that are non-volatile and show limited solubility (although they are generally soluble in nitric acid solutions). Usually, uranium daughter products are separated in solvent extraction processes designed to recover uranium. The decay products also tend to separate from the uranium in phase change operations such as vaporization of UF_6 (the decay products remain in the cylinder heels) or melting (the decay products concentrate in a dross that forms on the surface of the melt). Therefore, any contamination monitoring program must take into account the radioisotopes that may be encountered in the workplace based on an analysis of the process and on analytical data from various process streams. It is never safe to assume that only uranium is present.

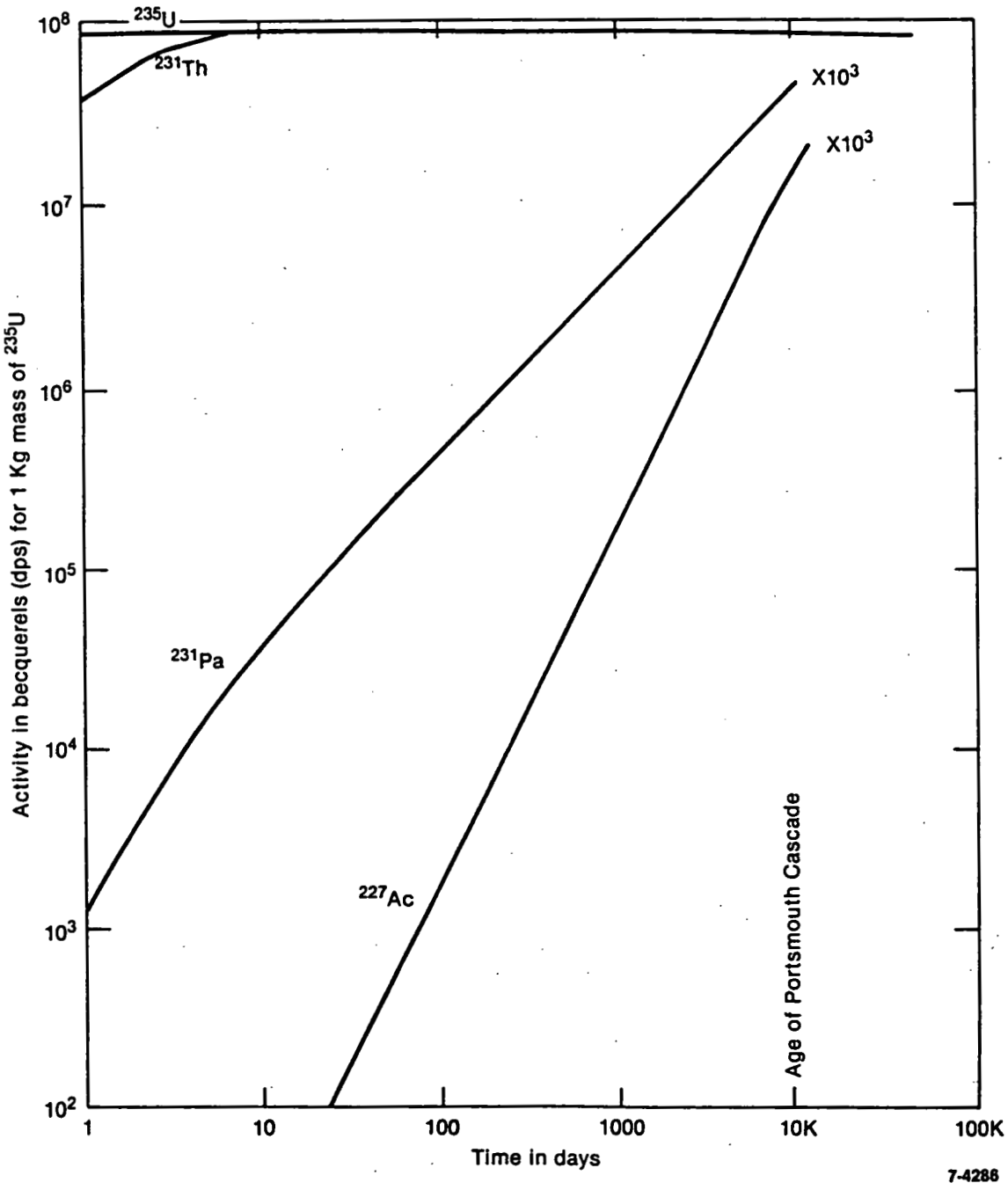

Figure 5-4. Activity of 1 kg mass of U-235.

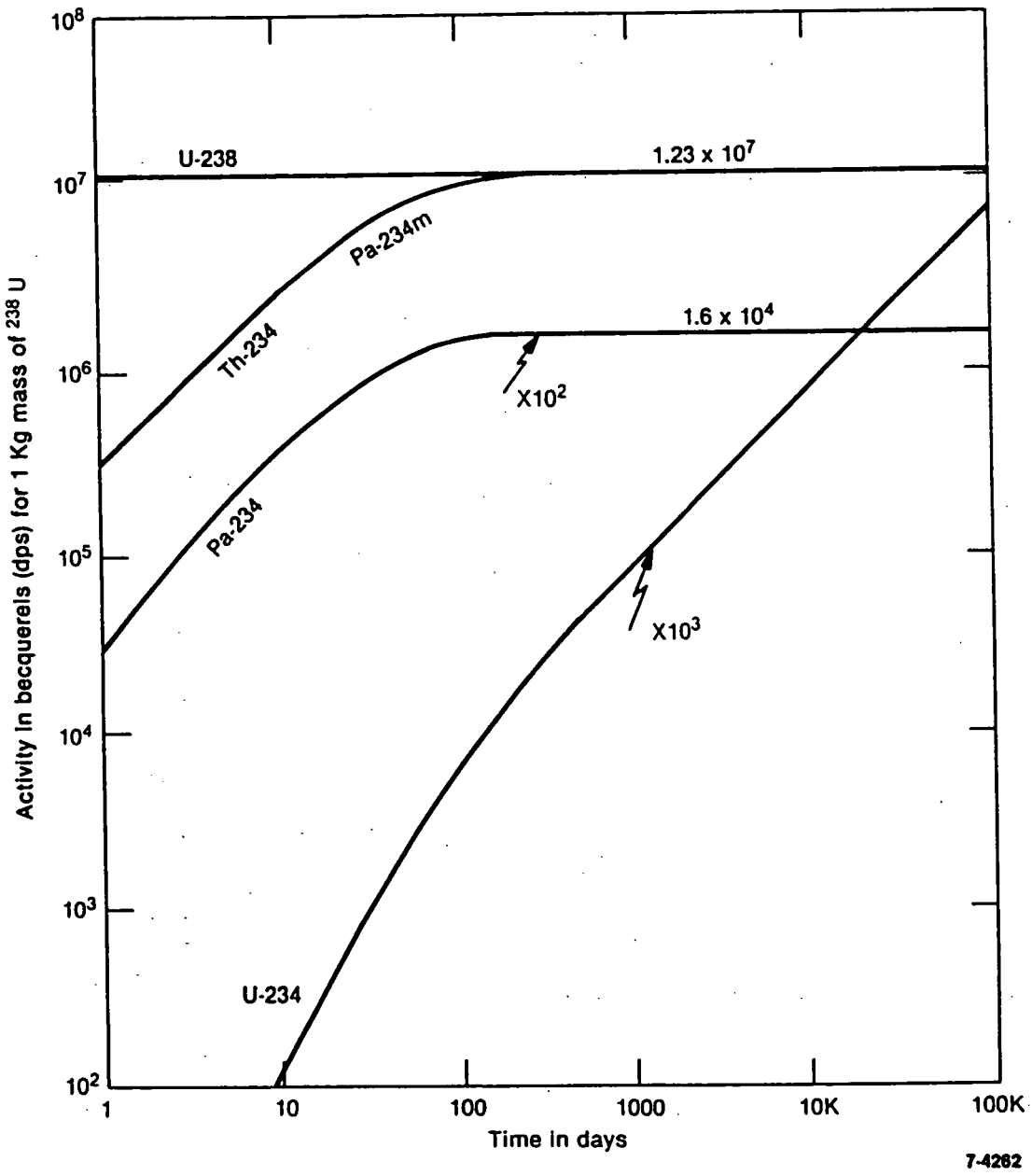

Figure 5-5. Activity of 1 kg mass of U-238.

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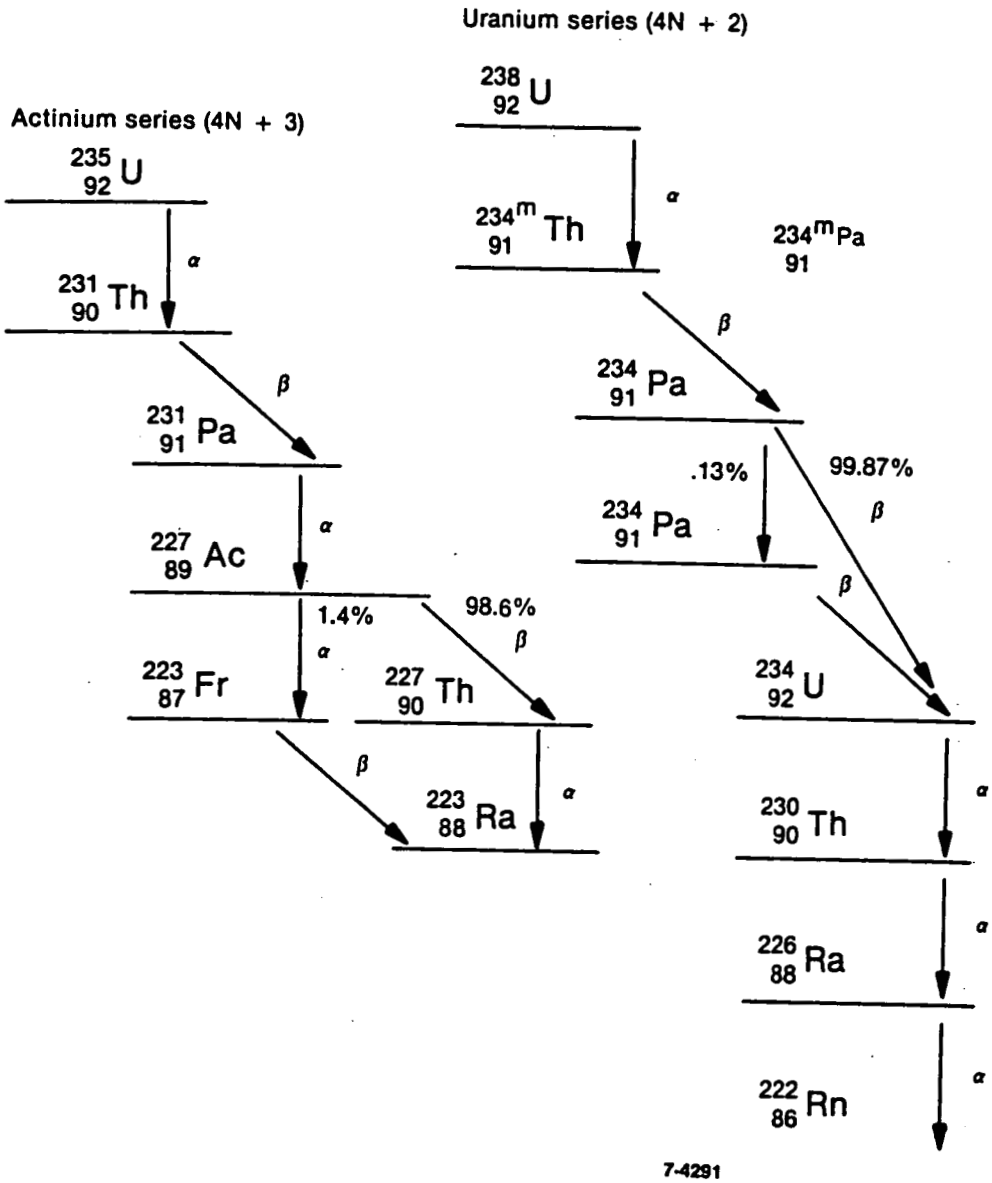

Figure 5-6. Actinium and uranium decay series.

5.2.4 Resuspension

It has been recognized by national and international radiation protection bodies that there is not a well-defined correlation between surface contamination and internal exposure to workers. However, the use of surface contamination monitoring results to estimate the potential for worker exposure is a common practice since the measurements are easily made and are done prior to the work activity. Additionally, it may provide a rationale or basis for establishing respiratory protection requirements based on precontamination levels especially for radionuclides which present primarily an internal exposure hazard (for example, uranium). Correlations have been defined in terms of resuspension factors. Resuspension factors (defined as the ratio of air contamination to the surface contamination) have been measured under a wide range of conditions. Resuspension factors typically range from $10^{-3}m^{-1}$ to $10^{-7}m^{-1}$. A factor $10^{-6}m^{-1}$ has been utilized as an appropriate resuspension factor for average work situations and general surface contamination and routine work activities. If contamination levels are utilized to specify worker protection requirements, a more rigorous approach is needed. Resuspension factors should be chosen based on the work activity. A range of resuspension factors can be utilized with work activities categorized according to their potential for creating airborne activity. If specific operational data is lacking, conservative resuspension factors should be utilized.

5.2.5 Site Contamination Limits

The previous discussion illustrates a few of the complexities of establishing a contamination monitoring and control program. Because of the complexities involved, professional health physicists and monitoring technicians must be trained to identify the contamination monitoring constraints of various facility processes. Generally, operating personnel should not be expected to perform more than simple meter readings and comparing results to "go/no-go" criteria.

An evaluation of jobs or processes for exposure potential should be performed by professional health physicists or health physics technicians. As discussed, there is generally no simple correlation of surface contamination levels to actual exposure of personnel to uranium or other associated radionuclides. However, contamination monitoring and control can be effective parts of an aggressive radiation protection program. It is important to clearly define the program objectives in terms of health protection and public relations considerations (e.g., the need to prevent off-site contamination) and then to plan the program to accomplish those objectives, taking into account the nature of the uranium contamination and processes at the specific site. While in practice each site utilizes program criteria based on DOE Orders and field office requirements, Table 5-5 lists recommended contamination guides for use in uranium facilities; other guides are discussed in the following section.

5.2.6 Control Zones

An integral part of a successful contamination control program is the establishment of a control zone system designed to confine contamination to the source (production, decontamination, and maintenance areas). A major design objective of the zone system must be to minimize the removal of uranium contamination from the plant on personal clothing or tools, equipment, etc. While the definition of zone criteria is dependent on the specific plant and the uranium operations involved, it is generally good practice to employ a three-zone system. Each zone has unique control requirements designed to protect personnel and to prevent the spread of contamination. Typical zone designations and zone descriptions follow; note that the actual terminology and zone designations for each site will utilize program criteria based on DOE Orders and field office requirements.

Unrestricted Zone

Unrestricted areas are those zones where radioactive materials are not handled except for materials in sealed containers or in conjunction with

TABLE 5-5. PLANT SURFACE AND PERSONNEL URANIUM[a] CONTAMINATION LIMITS WITHIN RADIOLOGICAL CONTROL AREAS

	Contamination Guides--dpm/100 cm^2 Removable[b]	
Area Designation	Good Practice[d]	Area Limit
Unrestricted	<200 α[c]	<1,000 α
Regulated (Controlled)	ALARA, generally <1000 average α	<10,000 α
Contamination Area	ALARA, generally <10,000 average α	Dependent on conditions at operation. Will always require protective measures.
Personnel and Personnel Clothing/Shoes	<200 α (direct reading)	<1000 α (direct reading)
Company-Issued Clothing/Shoes	ALARA, generally <5000 (direct reading)	<15,000 α (direct reading)

a. U-Ntl U-235, U-238 and associated decay products.

b. Measured by wiping surface with material and techniques designed to collect the contaminants and counting with and appropriate instrument of known efficiency (calibration with plated Uranium alpha standard source is recommended).

c. Depleted Uranium α/β ratios can reach 0.1 levels. For depleted material processes β counts should be used with the same limit applicable.

d. Monitoring techniques and equipment should provide the best feasible detection levels but should meet the "good practice" levels where achievable.

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procedure-prescribed movement of contaminated equipment. In this zone, there are no restrictions on work activities, or movement of personnel, or equipment. Contamination levels in this area should be "non-detectable" utilizing sensitive monitoring methods, but no higher than those area limits listed in Table 5-5.

Regulated Zone (Controlled)

The regulated zone designation refers to those controlled areas where radioactive materials are handled but the actual contamination or the potential for contamination on surfaces is low, provided that procedures are followed and equipment performs as designed. The controls required will be less stringent than those required for contamination zones. Monitoring of personnel and equipment should be performed prior to movement to an unrestricted zone. Monitoring equipment should be provided for employee use prior to breaks, lunch, or exiting the plant site. Some operations performed in regulated zones may require personnel protective equipment; however, activities should not require extensive use of protective clothing or similar stringent controls. For specific work activities (e.g., maintenance involving opening of uranium process equipment) with the potential for creating significant contamination levels, a temporary contamination control zone may need to be established. Workers in this zone should wear company-issued clothing.

Contamination Area Control Zone

A contamination area control zone designation applies to those areas where radioactive material is handled and significant levels of contamination exist, or can be generated, even though procedures are followed and equipment performs as designed. Controls are established to prevent the removal and spread of contamination from the zone via clothing, shoes, equipment, etc. Contamination monitoring of personnel and equipment prior to exit from the zone is essential. Usually, special work practices will be required to limit potential employee exposure and to control the

contamination generated. Anti-contamination (anti-C) apparel will be utilized and respiratory protection will often be required. Pre-job monitoring and evaluation by health physics technicians will usually be required. The job evaluation providing a summary of the contamination levels detected and specifying any personnel protection requirements will be performed and be readily available at the work site. A contamination control zone may be either permanently or temporarily established; the operating requirements and controls would be the same in either case.

To achieve containment of contamination in the Contamination Zone, all anti-Cs must be removed at a step-off area (pad) prior to exit from the zone. Additionally, personnel, tools, and equipment should be monitored for contamination each time they exit the zone. If the monitoring is to be performed by other than Health Physics personnel, specific exit guidelines should be provided preferably as conservative, "go, no-go" criteria. Post-job contamination monitoring of a temporary Contamination Zone is required with whatever decontamination activity is necessary to return the area to its previously designated contamination status; at which time the zone barriers may be removed.

In all areas, the primary requirement is the control of contamination by two basic methods. First, positive contamination prevention through job planning that stresses the use of reasonable and innovative precautionary measures to eliminate, reduce, and/or contain contamination generated by the work activity. Second, an active decontamination program administered according to established contamination limits. In this context, the area limits in Table 5-5 are those levels of contamination which provide an <u>upper boundary condition</u> for the particular zone in question. Although each installation will need to establish a policy for when decontamination is required, it would be appropriate to require decontamination, establishing boundaries, etc., whenever the <u>average</u> contamination level in the zone area exceeds the area limit value. Additionally, smaller sections of the zone exceeding area limits should be designated with boundaries if of significant size (>5 m^2).

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5.2.7 Monitoring

The effectiveness of radiological controls are evaluated through various monitoring programs. These programs include monitoring both the workplace environs and personnel. Monitoring resources should be allocated to provide an adequate database for evaluating employee exposures, assessing the effectiveness of uranium contamination control measures, and providing a means to identify ALARA goals. Surface contamination monitoring includes instrumentation requirements, monitoring techniques, and the structure of a surveillance program.

Instrumentation

To operate a contamination monitoring program at a uranium facility, several types of instrumentation are required. As noted earlier, uranium contamination may include a variety of uranium isotopes, depending on the U-235 enrichment, fission products, and transuranics. This mix of isotopes, and the range of contamination requiring detection, will determine the monitoring equipment required. In addition, the types of monitoring to be performed will also provide criteria for instrument selection. For example, the monitoring of personnel, equipment, scrap, and surfaces may present different instrument constraints.

Alpha Contamination Monitoring

For most surface contamination monitoring, alpha monitoring instruments will be the most sensitive and effective except in those facilities handling highly depleted uranium. However, the range of alpha particles should be considered when determining whether to utilize alpha monitoring to detect the presence of uranium contamination. For example, the use of alpha contamination monitoring to detect the presence of uranium contamination is inappropriate in the case of porous or dirty surfaces where appreciable self-absorption can occur.

There are three general types of portable instrumentation widely utilized for the monitoring of alpha surface contamination: gas flow proportional, air proportional, and scintillation detectors. Window density thickness for each of these detectors should be no more than 1-2 mg/cm^2. The gas flow proportional detectors are effective in detecting alpha contamination; however, they tend to be bulky for field use and are more often utilized in stationary applications, such as hand and foot monitors. Air proportional detectors are rugged to withstand field use; however, variations in the ambient atmospheric conditions (humidity, temperature and pressure) often result in instrument response fluctuations and instabilities that may make the readings unreliable. Scintillation detectors are effective in measuring alpha contamination, particularly in the presence of beta contamination or low gamma backgrounds. The scintillation detector is also relatively unaffected by humidity and temperature variations. However, the probes are delicate and will not withstand rugged field use. Any light leaks through the detector surface covering renders the instrument inoperable.

In general, using the "normal" field survey techniques, the lowest contamination level that can be reliably detected is 200 to 500 dpm/100 cm^2. This varies with the type of detector, probe active area, electronic set-up, etc. The types of radionuclides present should be considered, along with the need to discriminate between them. For example, if uranium contamination is present with beta emitting isotopes, such as, technetium or uranium daughters, the gas and air proportional detectors will detect both alpha and beta contamination to some degree. While proper discriminator settings will minimize the problem, a scintillation detector may be a better choice if low alpha detection levels are required in the presence of significant beta contamination levels. Isotopic analysis should be performed to identify relative ratios, interfering radiation, etc. If very low alpha removable detection levels are desired, wipes should be taken and counted in a laboratory counter. Instruments should be chosen that permit discrimination between alpha and beta radiation through

the choice of detector type (e.g., ZnS scintillator for alpha detection), and the choice of appropriate HV and discrimination settings on gas-filled detectors.

Techniques for Contamination Monitoring

The effectiveness of a contamination monitoring program is as much a function of the monitoring techniques as of the instrument response characteristics. The general directions for scanning for contamination is to scan at a low speed (5 to 10 cm/sec at a distance of approximately 0.5 cm. The scanning speed and the instrument response time will determine the level of contamination that can be detected. In some studies, using an alpha scintillation probe and a scanning speed of 10 cm/sec., in a low background area, the limit of detection for a distributed source was 125 dpm per 100 cm^2, and for a point source at a 50% probability of nondetection, the minimum detectable level was 1000 dpm/100 cm^2. Thus monitoring techniques and the geometry of the surface contamination bear directly on the minimum detectable level of contamination. Also, a determination of surface contamination levels of a large area requires a number of measurements to provide a statistically valid description of the contamination status. The contamination program documentation should cover these areas by procedures and training should be provided to technicians to assure consistency of monitoring practice.

Surveillance Program

The design of the surveillance or monitoring program is crucial to achieving the objectives of the overall health protection program. The design and organization of the contamination monitoring program is usually a matter of balancing available program resources (e.g., personnel and equipment) against the degree of assurance required to protect the health of personnel and to meet organizational objectives. In general, three basic types of surface contamination monitoring are performed:

1. Routine monitoring of equipment for contamination to specify decontamination requirements and/or to specify personnel protection/contamination control requirements for subsequent work activities. A means of communicating monitoring results is essential. In many cases, a tagging system can be effectively utilized.

2. Job specific monitoring of tasks that may involve the generation of significant contamination or that are performed in areas where significant levels of contamination may be present (for example, contamination zones). This monitoring would be performed by trained technicians and would include a pre-job hazard evaluation.

3. A routine contamination monitoring program of selected areas is required to determine the degree of general workplace contamination and to assess the effectiveness of contamination control practices.

The routine program should also include periodic surveys of unrestricted areas to assure that no significant contamination is present and that controls are functioning. The frequency of surveys should be developed with the objective of identifying deviations in program control objectives and assuring that any significant contribution to personnel exposure is identified. While this would of necessity be a site-specific activity, some general guidelines are as follows:

1. Unrestricted zones that are near or in the midst of restricted areas (for example, break areas or locker rooms) should be monitored on a frequent basis, weekly to daily, as dictated by monitoring experience.

2. Restricted (controlled) zones should be monitored at a frequency designed to detect a trend that exceeds the zone alert values in adequate time to assure that the zone guide values are not

exceeded. This frequency is heavily dependent on monitoring experience and the variability of operations as well as the aggressiveness of the contamination control program. Generally, weekly to monthly minimum.

3. Annual or more frequent audit surveys of unrestricted areas should be performed if there is any significant potential for the spread of contamination to the areas.

4. Annual audit surveys of restricted areas that are unlikely to have any significant contamination from routine operations to verify the status of the area's contamination.

5. Frequent surveys of any permanent contamination zones should be performed if there is routine worker activity in the zone. Otherwise, the areas should be monitored prior to any work activity.

5.2.8 Release Criteria

An important aspect of contamination control is the monitoring and the control of contamination on equipment being released for uncontrolled use. DOE Orders in the 5400 Series on Radiological Protection specify release criteria. The primary areas of technical concern deal with two major issues; first, the difficulty of accurately monitoring for uranium contamination on irregular, inaccessible and/or porous surfaces, and second, the need to evaluate specific isotope activity contributions to the surface contamination levels when fission products or transuranics are present. From a program standpoint, a formalized control program for material to be excessed must be implemented and enforced. The monitoring methods and release criteria must be defined and documented; program performance should be verified, and audited periodically for adequacy.

5.2.9 ALARA Guidelines

In strict terms, surface contamination levels are not controlled with radiation exposure of personnel as the basis. The control of contamination from an ALARA standpoint is a matter of policy as opposed to a matter of structured dose reduction goals i.e. even detectable activity on personnel and in the public is unacceptable though not of significant personnel hazard at the "levels of concern". In general, contamination should always be reduced through good industrial housekeeping practices. Additionally, work philosophy and practices should be established for containing radioactive materials and preventing the spread of contamination to other areas. The application of ALARA to the area of contamination control requires several basic programmatic features. First, a formal, documented contamination control program is required; this program must be communicated through training and work procedures to all parts of the organization. Second, a rigorous monitoring program must be developed and integrated with a routine cleaning/
decontamination program. Lastly, a system of data analysis is necessary to identify trends and to provide for field audits of contamination control program performance. In conjunction with the data analyses program, specific ALARA goals may be established to reduce contamination levels in a particular area over some time.

5.3 Personnel Contamination Control

5.3.1 Monitoring Philosophy

Although the primary hazard to personnel from uranium is an internal exposure hazard, contamination of personnel is also of concern because of potential ingestion hazards and potential skin doses. Additionally, an objective of the contamination control program is to confine uranium contamination to production/work areas and to minimize, to the extent practicable, any spread of contamination to areas outside the plant or to the public. Therefore, guidelines for allowable contamination on personnel and personal clothing/shoes both inside the plant and prior to exiting the

restricted areas are required. Also, a personnel monitoring program must be developed with adequate monitoring equipment and sensitivity to provide assurance that contamination is effectively controlled. The guidelines should be developed with consideration of the following factors:

a. The need to prevent detectable activity from appearing in the public.

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

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

d. Commitment to the policy of keeping contamination to the minimum practical level.

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

5.3.2 Monitoring Program

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

A major aid to control of personal contamination will be the use of company-issue clothing and shoes for employees with permanent work assignment in regulated or contamination areas. For personnel surveying at contamination zone boundaries, portable instruments utilizing pancake GM probes have proven to be effective. At boundaries to unrestricted areas, fixed hand/foot and/or portal monitors (if beta levels are equivalent to the alpha levels) are generally more effective. The monitors utilized should be chosen for maximum sensitivity to alpha and/or beta contamination based on the materials being processed. Monitors are available which discriminate between the two types of contamination; and provide separate alarm levels with appropriate warning messages displayed.

5.3.3 Protective Clothing (PC)

Employees should wear protective clothing in any contamination zone. Protective clothing should be removed at a zone "step-off" area and personnel monitoring for contamination should be performed. If monitoring is not practical, strict control of the movement of personnel from the step-off area to a location where monitoring can be performed is necessary. It is not good practice to allow personnel wearing protective clothing that has been worn in a contaminated area to co-mingle with individuals in personal (street) clothing. Protective clothing shall not be allowed in noncontrolled areas such as offices, lunch rooms, control rooms, etc. The choice of PC garments will be based on the type of job and the form of contamination hazard. For example, a job sampling uranium oxides will have significantly different contamination mechanisms than a wet-decontamination operation. In addition, protective clothing should meet both PC and chemical hazard protection requirements. For example, in a decontamination operation using acid solutions, the PCs should provide protection against both the uranium and the acid. This may be a difficult matter where organic solvents are utilized along with aqueous solutions in the total cleaning process. The various aspects of the work activity should be carefully analyzed to select appropriate protective clothing; this will require a coordinated effort with other health protection

groups. Company-issued clothing and shoes should be utilized under the PCs; so that if there is a tear, an employee's personal clothing will not become contaminated. Personnel must be trained routinely on the effective use of PCs and their job performance audited periodically.

5.3.4 Respiratory Protection

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

As with personnel protective equipment, respiratory equipment utilized must also provide protection from the full range of airborne hazards that may be encountered in the work environment. For example, a uranium metal machining operation may have both an airborne uranium oxide hazard and an airborne hazard from solvent vapors. The respirator utilized must be effective for both types of hazard. Also, one airborne contaminant may interfere with the effectiveness of the canister in an air-purifying device that is designed for a different contaminant. For example, a corrosive gas, such as hydrofluoric acid (HF), may attack a HEPA filter and render the filter ineffective. It is important to coordinate the use of respiratory protection requirements with other health protection groups. The respiratory protection program must also be in compliance with ANSI Standard Z88.1 requirements. In specifying respirators for various applications, one should always know the applicable protection factors to assure that the range of hazard that may be encountered will be covered. While the specification of respiratory protection should normally be made a result of personal and/or area sampling results, the use of respirator guides based on surface and wipe contamination monitoring results may be utilized.

5.3.5 ALARA Guidelines

The primary ALARA guideline relating to personnel contamination is that any contamination should be kept at the lowest level practicable. While the actual hazard may be minimal, the social implications of contaminating the home or family of an employee are immense. In plant operations, contamination levels should be reduced wherever practical, especially in areas where personnel are routinely assigned.

5.3.6 Release Criteria

Generally, it is appropriate to set release limits for personnel at the level of detection of the monitoring instrumentation (the reliable limit of detection). However, if an individual is contaminated and decontamination efforts are not successful in reaching normal limits, specific guidance is needed for safe, but detectable release limits, to prevent decontamination efforts causing potential injury if no significant hazard is involved. An example of guidance that may be provided is shown in Table 5-5.

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SECTION 6

INTERNAL DOSE CONTROL

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SECTION 6

INTERNAL DOSE CONTROL

In most uranium facilities, the primary radiological hazard is the potential for internal intakes of uranium. This hazard is usually properly controlled by appropriate facility and equipment design, contamination control procedures, and protective clothing. An integral part of an internal dose control program is a bioassay program for uranium.

Bioassay for uranium involves the use of in-vivo measurements and analysis of biological samples (urine, feces, etc.) to make estimates of depositions of uranium activity within the body. Bioassay programs serve a multitude of purposes including:

a. exposure control (detection of intakes),
b. assessment of doses from intakes,
c. documentation of results, and
d. compliance with regulatory requirements.

Bioassay programs for uranium typically involve the collection of urine samples to monitor intakes of transportable uranium, and in-vivo measurements for the deposition of nontransportable uranium. Fecal samples are also analyzed to assess inhalation intakes of nontransportable uranium.

6.1 Internal Dosimetry Surveillance

The workplace should be designed to prevent intakes of radionuclides to the extent reasonably achievable. Radiological controls for the workplace should ensure that radionuclides are confined and handled properly, and that intakes, if they occur at all, are negligible, to the extent achievable with state-of-the-art technology.

Internal exposure to radionuclides may occur as the result of inadvertent intakes of radioactive material. Internal dosimetry surveillance programs should be designed to rapidly detect a release in the event of a loss of radioactive material containment.

When workers are inadvertently exposed to radioactive material, it will be necessary to take appropriate corrective action to ensure that control and confinement have been reestablished. To control occupational exposure for the worker, an early assessment of probable severity, preferably within the first two hours of the intake, is desirable. Preliminary assessment of severity (based on estimates of probable intakes) is necessary to provide guidance for both medical professionals and the employee. The probable severity of each suspected intake is to be evaluated as soon as possible using whatever data are available. Examples of such data, which may be of use in specific cases, include:

a. A description of the internal exposure scenario
b. Air concentration and exposure duration
c. Skin and clothing contamination
d. Nasal or saliva smears
e. Direct bioassay measurements (in vivo)
f. Indirect bioassay measurements (in vitro).

6.1.1 Performance Capabilities for Internal Exposure Monitoring

It is the policy of the Department of Energy that facilities are designed and operated to prevent intakes of radioactive material to the extent reasonably achievable. The requirements in this section define a minimum capability for internal exposure monitoring programs.

Internal exposure monitoring programs should include both prospective monitoring, and retrospective monitoring and dose assessment. The objectives of the prospective monitoring program are to verify the integrity of the process confinement and to detect inadvertent releases of

radioactive materials in the workplace. Prospective monitoring should also indicate possible intakes of radionuclides by workers. The objective of the retrospective monitoring program is to measure radioactive material in the body and in samples of excreta collected from workers with known or suspected intakes so that dose assessment may be performed. Retrospective monitoring also involves measurements to confirm an intake of radioactive material, if any, to identify radionuclides involved, to quantify the magnitude of confirmed intakes, to verify biokinetic models used with bioassay data to estimate intakes, and to document the annual effective dose equivalent and the committed effective dose equivalent received by the worker.

Performance Capabilities for Workplace Monitoring Programs (Prospective Monitoring)

Programs for prospective monitoring of the workplace should be designed and operated to detect potential intakes of radioactive materials by workers that could result in a committed effective dose equivalent of 100 mrem, considering all occupationally derived radionuclides to which the individual may be exposed during the year in the workplace. The primary method of monitoring the workplace is air monitoring. However, prospective bioassay measurements may complement the air monitoring program.

In the design and evaluation of the monitoring required to meet DOE 5480.11, the specific characteristics of the contaminants potentially involved should be considered, including radionuclide composition, particle size distribution, potential modes of intake, transportability from the lung to other organs, and absorption into the systemic circulation. If such information is not available, assumptions based on conservative evaluation of facility operations and recommendations of advisory groups such as the NCRP and the ICRP should be applied.

If a release of radioactive material is detected that could result in an intake delivering a committed effective dose equivalent of 100 mrem, workplace monitoring results should be associated with the workers involved.

It may be difficult to meet the above performance capabilities for some forms of uranium such as enriched uranium oxides. If the performance capabilities cannot be achieved, the following should be applied in lieu of the above requirements:

1. The best practicable (state-of-the-art) prospective monitoring capability should be used.

2. Internal exposures should be prevented by enhanced facility design, operation, controls, and personnel protection equipment.

3. Exposure control measures should be documented for auditing purposes.

Performance Capabilities for Individual Dose Assessment (Retrospective Monitoring)

Performance capabilities for retrospective monitoring and dose assessment are specified in terms of the internal component of the annual effective dose equivalent. The minimum detection capability does not include contributions from external dose. As a minimum, the measurements used to assess and document internal dose should be capable of confirming and assessing:

1. exposures that could result in an annual effective dose equivalent in excess of 100 mrem from all intakes of radionuclides occurring during the calendar year, and

2. the presence of radionuclides in the body from all previous intakes, irrespective of the year of intake, that would deliver an annual effective dose equivalent in excess of 500 mrem.

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If these retrospective performance objectives cannot be achieved, the organization responsible for radiation protection should take administrative action to:

1. ensure that additional control measures are applied to prevent intakes of radioactive material to the extent reasonably achievable,

2. document these additional control measures for auditing purposes,

3. upgrade bioassay measurement systems and air monitoring equipment to provide state-of-the-art measurements, and

4. ensure that the best technology is available for internal dose assessments.

The significance of all confirmed occupational internal exposures, regardless of magnitude, should be assessed. Results of all bioassay and other measurements needed to support the quality of measurements, and dose assessment should be recorded and maintained.

The assessed calendar year effective dose equivalent resulting from intakes of radionuclides during the year, based on confirmed bioassay measurements, should be recorded. If the annual effective dose equivalent from such intakes is less than 10 mrem, it need not be reported to DOE for compliance with 5480.11. Separate recording of the calendar year individual organ dose equivalent is not required if less than 100 mrem. However, the dose contribution from all organs must be considered in the computation of the effective dose equivalent. The committed effective dose equivalent should be recorded for any confirmed intake.

6.1.2 Protection of Unborn, Minors, and Students

Additional administrative controls to protect the unborn should be established for radiation workers who have declared their pregnancy. Enhanced control of internal exposure to minors and students should be exercised since the effective dose equivalent limits for these individuals are the same as for the general public.

6.2 Characterization of Internal Hazards

Bioassay programs for uranium must be specifically designed for the type of uranium hazards presented by the facility. These hazards must be well-characterized before the scope of the program can be defined and sampling or measurement techniques selected.

The first step in the design of a bioassay program is to thoroughly characterize the potential internal hazards presented by the types of uranium present and operations performed in the facility. Most uranium facilities deal with more than one chemical or physical form of the element. As discussed earlier, most "uranium" is in reality a mix of uranium isotopes, daughter radionuclides, and impurity radionuclides.

Characterization of uranium internal hazards should cover the following areas:

Radionuclides present:

a. What range of enrichments are present?
b. What mix of isotopes can be expected over this range?
c. What uranium daughters will be present?
d. How will the ratio of daughter to uranium activity vary?
e. What impurity radionuclides are present?

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Chemical/Physical forms:

a. What chemical forms of radionuclides are present?
b. What clearance classes are appropriate for these forms?
c. What physical forms of uranium are present?
d. What particle size distributions can be expected?

Intake mechanisms:

a. What routes of intake are probable or possible?
b. What activity dispersal mechanisms are present?
c. Are intakes likely to be acute, chronic, or intermittent?
d. What other indicators of possible intakes will be available?

Such a characterization is easy to describe, but may be difficult to perform in the field. An appreciation for such real world difficulties may be gained by considering the following comment from a recent colloquium:

> ". . . consider the standpoint of a lathe operator working with a large piece of uranium metal which has a potential for oxidizing and becoming airborne. I see an occasional urinary uranium excretion which obviously results from this event. I am also going back to look at the kind of compound to which he was actually exposed. Frankly, I don't know and can't figure any way to determine it. I know the final form of oxidation is probably U_3O_8 but from the time its twirling off that machine and getting up where he can breath it, I don't believe it goes to U_3O_8, and I don't think it is UO_2. It might be UO_3, but I really think it is going from one oxide to the other as it comes up, probably going through the reaction by the time it is inhaled. . . All we know is that we started with metal that went to some kind of oxide, maybe not, as it got into his lungs"

6.3 Scope of Bioassay Program

The rationale for each category of bioassay measurement must be established, as well as rationale for selection of personnel or groups of

personnel to be monitored. The frequency of sampling or measurement should be established based on either air activity measurements or a "missed dose" concept.

Selection of bioassay measurement techniques and associated measurement frequencies plays a critical role in establishing the effectiveness of a bioassay program. A poorly designed bioassay program may be worse than no program at all. For example, a continual stream of "negative" results from a program with very poor detection capability due to inappropriate selection of bioassay types or frequencies can result in a false sense of security.

6.3.1 Classification of Bioassay Measurements

Once the range of potential hazards from uranium has been characterized, bioassay samples or measurements should be classified according to their primary function. A recommended classification scheme follows.

Category	Purpose
1. Baseline (preparatory)	To establish the individual's status relative to any previous uranium intakes. Such a measurement should be made on new hires and on workers transferring to uranium processing facilities.
2. Prospective Monitoring	To detect intakes of uranium, thus providing feedback on the effectiveness of containment and control of uranium.

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Category	Purpose
3. Retrospective Monitoring (Dose Assessment)	To provide estimates of the intake or deposition of uranium for use in assessing dose to organs of concern.
4. Separation or Termination	To document the status of uranium deposition at time of termination of employment or a transfer from uranium work.

Such classification is important because the purpose of a sample can greatly affect the collection method and subsequent analysis. For example, "spot" urine samples (of a few 10s of cc) may be adequate for exposure control purposes, however, "24-hr" samples may be necessary for dose assessment purposes.

Appropriate baseline measurements should be made for all new hires who will be working with uranium. Results of these measurements should be available and reviewed before the new hire begins actually working with uranium. In some circumstances, measurements taken upon separation from the previous facility may be acceptable.

Routine prospective exposure control measurements are, in a sense, the "bottom line" of the internal dose control program. A comprehensive prospective bioassay program which normally returns insignificant results is proof that the internal dose control program is working properly. In contrast, a program which frequently detects significant intakes is conveying a message about the facility's contamination control program.

Retrospective bioassay measurements are made to confirm positive results and to allow the health physicist to make estimates of the quantity of activity taken into the body or deposited in organs of concern. Since estimates of initial deposition are based on time-dependent retention

functions, the date (and sometimes the time) of sample and measurement becomes critical. Urine or fecal samples which can be related to a daily excretion rate are usually required.

Retrospective measurements and samples should begin as soon as possible after an acute intake. Samples should not be collected in the workplace where the possibility of inadvertent contamination exists. If a significant uptake is indicated, the worker should not be exposed to further intake during the assessment period.

Separation/termination samples or measurements are assuming a more important role in today's environment. Such measurements can document the level of intake or deposition of uranium in the terminating employee at the time of separation from the facility.

6.3.2 Prospective Monitoring Requirements/Selection of Employees

Many factors affect the choice of personnel or groups of personnel for prospective bioassay monitoring. The Department of Energy has directly or indirectly established mandatory and recommended monitoring requirements. The current DOE Order 5480.11, Radiation Protection for Occupational Workers, states that:

> "Internal dose evaluation programs (including routine bioassay programs) shall be adequate to demonstrate compliance with the Radiation Protection Standards in Paragraph 8a. Programs are required for radiation workers exposed to surface or airborne radioactive contamination such that the worker may receive 0.1 rem (0.001 sievert) annual effective dose equivalent per year from all intakes of radionuclides, or if any organ or tissue dose equivalent may exceed 5 rem (0.05 sievert) per year."

In addition to providing information about exposure control, routine bioassay measurements also provide documentation regarding the magnitude of intakes for a worker. Should future questions arise regarding past exposure to uranium, such measurements can be invaluable. In light of such

considerations, it is recommended that even workers with casual contact with uranium (custodians, guards, etc.) be placed on a minimal bioassay program as outlined in the following sections. As an absolute minimum, workers who may be exposed to intakes that could result in an annual effective dose equivalent of 0.1 rem or greater should be included in a bioassay program. DOE Order 5480.11 has specific requirements for monitoring.

6.3.3 Selection of Bioassay Monitoring Technique

The following scheme should be used to guide the appropriate bioassay measurement techniques:

	Uranium Transportability Class		
Bioassay Category	D	W	Y
Baseline[a]	U	I (U)	I (U)
Prospective Monitoring	U	I+U	I+U (F)
Retrospective Monitoring	U	I+U (F)	I+U (F)
Separation or Terminology	U	I+U	I+U (F)

where

U	=	urine
I	=	In-Vivo
F	=	Fecal
+	=	in addition to
()	=	may also be useful.

a. The choice of monitoring techniques for a baseline assessment should also consider the nature of any past exposure.

Comments:

1. "Spot" or "grab" urine samples may be satisfactory for Baseline, Prospective Monitoring or Separation samples. "24-hr" or "simulated 24-hr" samples should be used for retrospective assessment.

2. Non-radiometric analysis of urine samples (e.g. uranium fluorometric) is suitable for relatively low enrichments of uranium. Isotopic analysis should be used for enrichments over 10%, particularly when workers could be exposed to various enrichments of uranium.

3. The excretion of natural dietary uranium in the feces will mimic excretion of inhaled uranium via the GI tract. Isotopic analysis of uranium in fecal samples will permit some discrimination if the enrichment is high enough. Non-radiometric analysis of fecal samples may not be particularly meaningful if the enrichment is greater than 10% or if the worker is exposed to a range of high and low enrichment.

6.4 Establishing Bioassay Frequency

A variety of factors such as measured air activity concentration, the magnitude of "missed doses" and other, less tangible factors affect the frequency at which bioassay samples or measurements should be taken. The category of bioassay measurement may also affect the frequency of measurement.

6.4.1 Frequency Based on Air Activity Concentration

The Nuclear Regulatory Commission (NRC) (NUREG 8.11, WASH-12511) has published detailed analyses and discussions of bioassay frequency based upon average and maximum quarterly air activity concentration measurements.

These schemes specify a "Minimum Bioassay Program," with increased bioassay frequencies based on air activity results. This sampling frequency scheme is shown in Table 6-1. (Suggested fecal sampling frequencies have been added for Class Y material.)

Such guidelines should be followed to the extent that they are applicable to the facility in question, subject to some limitations. If bioassay frequencies are based upon air activity measurement, care should be taken to ensure that those measurements are representative of the activity concentration being breathed. Fixed air sampler results probably will not be representative of the concentration of uranium in air being breathed.

The scheme implies that bioassay frequencies may change from quarter to quarter based upon changes in observed air activity concentrations. However, even if the air activity concentrations do change and even significantly, it may not be practical to change bioassay frequencies that often.

Bioassay frequencies established using this methodology should be reviewed in light of the "Missed Dose" concept.

6.4.2 Frequency Based on the "Missed Dose" Concept

An alternate method of establishing bioassay frequencies is based on the concept of the dose which would be "missed" at a given measurement interval. The pattern of retention of activity in the body, the sensitivity of a bioassay measurement technique, and the frequency with which that technique is applied define a quantity of intake which could go undetected by the bioassay program. An intake of such a magnitude would not be detected if it occurred immediately after a bioassay measurement and was eliminated from the body at such a rate that nothing was detected at

TABLE 6-1. SAMPLING FREQUENCY BASED ON AIR ACTIVITY

Results	Class D	Class W		Class Y		
	U	U	I	U	I	F
0 < Q < 1/10						
1/4 < M < 1	s	s	a	s	s	
1 < M < 10	q	q	q	q	q	
10 < M	m	m	q	m	q	
1/10 < Q < 1/4						
0 < M < 1	s	s	a	q	s	
1 < M < 10	q	q	s	m	q	
10 < M	m	m	q	b	q	
1/4 < Q < 1/2						
0 < M < 1	s	q	s	m	q	
1 < M < 10	q	m	q	b	q	
10 < M	m	b	m	w	m	a*
1/2 > Q						
0 < M < 10	m	b	m	b	m	a*
10 < M	b	w	m	w	m	a*

where: U = Urine I = In-Vivo F = Fecal

Q = Quarterly average air activity concentration (in fractions of DAC)

M = Maximum result for a single quarter (in DACs)

a = annual
s = semi-annual
q = quarterly
m = monthly
b = biweekly
w = weekly

* For highly enriched uranium.

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the next scheduled measurement. The dose resulting from such an intake would be the "missed dose" for that particular measurement technique and frequency.

This "missed dose" concept is illustrated in Figure 6-1 for In-Vivo monitoring of Class Y uranium. Here the lung retention function discussed in Section 6.5 has been combined with dose conversion factors to give a committed dose equivalent to the lung that could be missed as a function of In-Vivo counting interval. The lower limit of detection for the In-Vivo system is assumed to be 1 nci of deposited (total) uranium alpha activity.

In this example, a lung counting interval of 365 days could miss a committed dose equivalent to the lung of about 9.5 rem, or a committed effective dose equivalent of about 1.1 rem.

Similar estimates of missed dose for other organs of concern and other measurement techniques should be established. Retention functions specific to the chemical forms and particle size distributions found in the facility should be used.

The overall "exposure control" sensitivity of a bioassay program is, of course, supplemented by the air monitoring and contamination survey programs. It is the combination of all of these programs which results in a quality program to assure that intakes are kept to a minimum. The effective sensitivity of the prospective bioassay program (including all measurement techniques used) should allow the health physicist to identify intakes that could result in a committed effective dose equivalent of 100 mrem, considering all occupationally derived radionuclides to which the individual may be exposed during the year in the workplace.

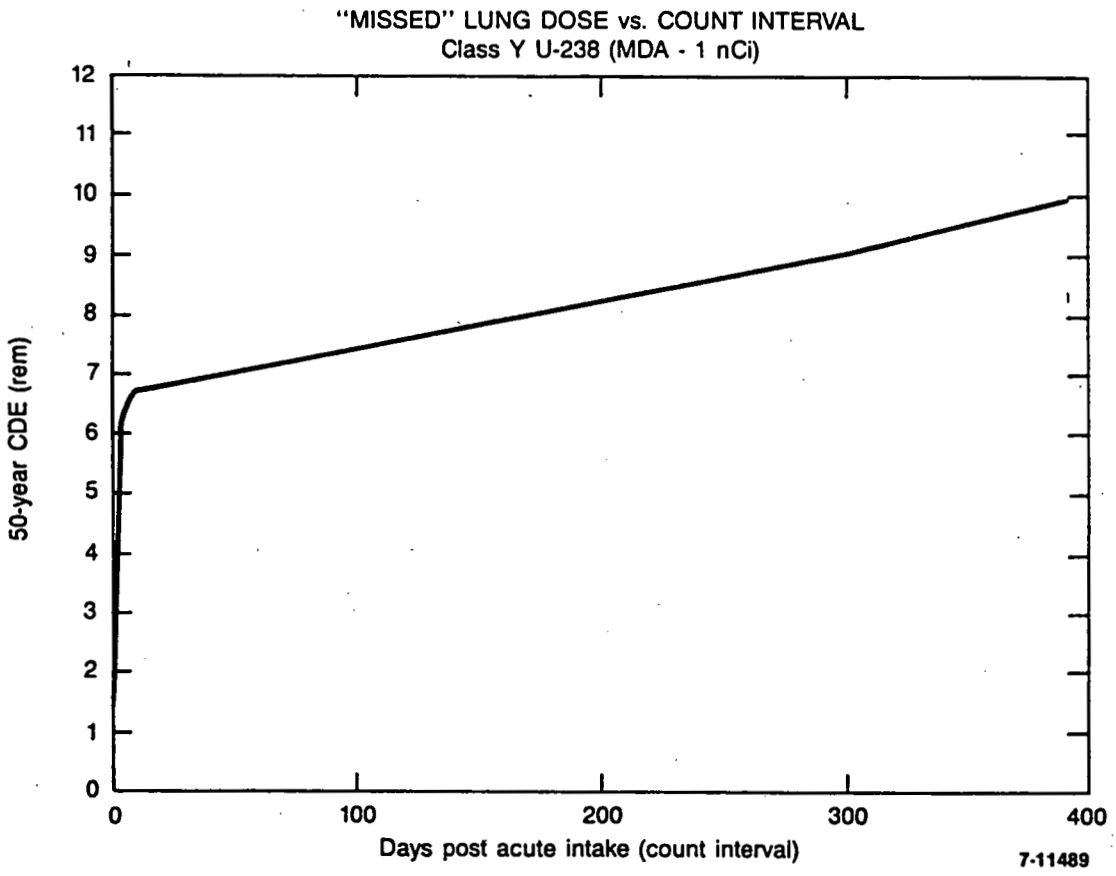

Figure 6-1. Missed lung dose concept for In-Vivo monitoring of Class Y uranium 238.

Finally, the frequency of bioassay measurements should normally not be reduced because of results near or not significantly different from zero. The bioassay program should be maintained in order to continue to act as the "base line" of the contamination control program, and to document the absence of significant intakes of uranium.

6.5 Administration of a Bioassay Program

Guidelines for administration of the routine (prospective) monitoring program must be established and should include specific procedures for scheduling, collection, and measurement. Action/reference levels should be established for appropriate responses to positive bioassay results. Procedures for managing and reporting bioassay data must be established. Dose and intake assessment methodology should be established and documented. Finally, a quality control program, covering sample collection through records storage, should be established.

6.5.1 In-Vivo Monitoring

In-vivo monitoring programs for uranium should use state-of-the-art detectors if possible. Detectors in common use include sodium iodide (NaI:Tl) phoswich (NaI and CsI Sandwich), and hyperpure germanium (HPGE). Greatest efficiency is achieved with NaI; superior spectral resolution is obtained with HPGE; phoswich detectors are intermediate in both efficiency and resolution. Table 6-2 lists uranium gamma and x-ray emissions used for in-vivo monitoring.

Facilities housing in-vivo detection systems should provide as low a background radiation contribution as possible. New facility sites should be selected with this goal in mind. Construction materials should be well-characterized in terms of their contribution to background. Facilities should be "sized" appropriately for the number of detection systems and anticipated scheduling load. The facility should be arranged so as to minimize foot traffic in the counting rooms. Adequate data acquisition, reduction, and management systems are of paramount importance.

TABLE 6-2. MAJOR URANIUM GAMMA EMISSIONS USED FOR IN-VIVO COUNTING

Radionuclide	Half-life	Energy (keV)	Yield %
U-234	2.445E5 y	none used	
U-235	7.038E8 y	144	10.5
		186	54.0
		200	~5.0
Th-231	25.5 h	26	2.0
		84	10.0
U-238	4.468E9 y	none	
Th-234	2.410E1 d	63	3.8
		93	~5.0
Pa-234m	1.17 m	765	0.2
		1000	0.6

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In-vivo counting systems should be calibrated using tissue equivalent torso phantoms such as the Lawrence Livermore National Laboratory (LLNL) phantom. Sources representative of the isotopic mixes encountered in the facility should be used if available. The background radiation environment of the facility/detection system must be well characterized.

The frequency of routine in-vivo counts should be established based primarily on the acceptable "missed dose" concept as discussed in Section 6.4.2. The count frequency associated with a particular missed dose is a function of detector sensitivity, retention of activity in the lung, and isotopic mix. Detection limits on the order of just less than 1 nci of U-238 or U-235 (detected directly or indirectly) should be available with state-of-the-art systems. Retention of activity in the lung will, of course, depend upon particle size and transportability/solubility considerations discussed in Section 6.6.1.

The lung retention functions used in conjunction with in-vivo counting should be "customized" as much as possible to reflect actual particle size distribution and transportability classes encountered in the facility. If a range of particle size and transportability classes is likely, retention functions which cover this range of possible retention behaviors should be available. Standard retention functions (e.g. Class Y, 1 micron AMAD) should be used for assessing significant intakes only when no additional information is available.

Counting and Reporting Practices

Since surface uranium contamination can severely distort in-vivo count results, workers should shower and shampoo their hair before being counted. Disposable coveralls or other disposable apparel should be worn during the count. Recycled, laundered cloth coveralls may be used provided that they are washed separately from other "plant" clothing. Slightly contaminated undergarments have caused false high count results. Consideration should be given to providing workers with alternative disposable undergarments if this problem appears to be likely.

In systems employing both "front" and "back" detectors, a front-to-back ratio can sometimes provide information about surface contamination. The use of high-resolution detectors can facilitate identification of surface contamination by allowing low-energy x-ray peak height comparisons.

Expressing results in terms of mass units (e.g. "100 mg Total Uranium," "20 mg U-235") is discouraged because such results must be coupled with an assumed degree of enrichment (and resultant isotopic mix) in order to be meaningful. It is recommended that results be expressed in units of activity of the particular radionuclide detected, along with the implied quantity of activity of other radionuclides of interest.

For example, if the 93 keV peak of Th-234 has been used to measure the deposition of natural uranium, the results would be expressed as:

measured: 1 nanocuries of Th-234,
which implies 1 nanocuries of U-238
and 1 nanocuries of U-234

Similarly, the practice of reporting results in terms of Maximum Permissible Body (or Lung) Burdens is also discouraged. These terms are not defined or used in DOE Order 5480.11.

Workers being counted should be informed about the purpose of the count and how the detection system works. Some facilities have information pamphlets available for this purpose. Workers should always be informed of the results of the count. Positive count results should be presented to the worker in terms of annual dose equivalents and predicted annual effective and committed effective dose equivalents, in addition to the quantity of activity observed.

6.5.2 Urine Sampling Program

Employees selected for participation in a routine bioassay program should be given information about the program including the rationale for the program, and the purpose of sampling as part of their training. Sample collection schedules and methods should also be presented. Questions should be encouraged and answered fully.

The selection of the day and time of sample collection should be tailored to the facility operation and type of uranium present and analysis of sample. Samples used for prospective monitoring should be collected within a few hours to a day after the period of likely intake. "Friday afternoon" sampling (for weekly sample intervals) will provide the most sensitive indication of whether intakes are occurring. Samples used for retrospective assessment should be timed to avoid the rapid early clearance period.

Collection of urine samples should be performed in a radiologically clean environment. Both internal and external contamination of the sample bottle must be avoided. Ideally, samples should be collected at home; however, collection and transport problems may rule out collection at home. Facility rest rooms may not be suitable collection sites, particularly if the workers can access the rest room directly from the work areas. Workers should be instructed to wash their hands before collecting the sample. Workers in contaminated areas should shower and change clothing prior to collecting the sample.

Sample collection stations should have an adequate supply of sample containers, with appropriate materials for labeling the samples. A sample log-in sheet should be used to track the samples. A "last day worked" entry on the sample label or log-in sheet may be useful. The sample collection station and sample containers should be designed and selected to avoid sanitation problems. Male and female workers should be kept in mind when selecting sample collection containers. Unbreakable collection

containers should be used if practicable. Care should be taken in selecting sample containers as some plastic containers can cause "plate out" of uranium, and some glass containers have small amounts of uranium in the glass. Sample labels should provide space for employee name, social security number (or other unique identifier), and date/time at the beginning and end of the sample collection period.

Samples should be collected and processed as soon as possible. Samples sent to outside laboratories must be packaged to withstand transport.

Newer analysis methods (such as laser phosphorimetry) enable detection of concentrations of uranium in urine which are on the order of those expected from excretion of uranium ingested in the normal diet. In such cases an expected range of "natural background" uranium concentrations must be established using samples from a nonexposed local population.

6.5.3 Fecal Sampling

A review of the ICRP Respiratory Tract Model makes it clear that a large fraction of inhaled Class Y and Class W material is cleared by ciliary action, swallowed and subsequently excreted in the feces. For example, fecal excretion rates from continuous intakes of Class Y uranium can be 2 orders of magnitude greater than urinary excretion rates. Factors of up to 5 can be expected for long periods after acute intakes. The metabolic path from inhalation to fecal excretion is fairly simple compared to that for uranium which goes to the bloodstream. On a purely theoretical basis, fecal sampling would appear to offer several advantages to urine sampling.

Practical limitations and problems have caused fecal sampling to play a very minimal role in most uranium facilities. Sample collection difficulties and aesthetics have resulted in routine programs which avoid fecal sampling. In addition, the "background" contribution of natural

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uranium from the average diet can mask low level intakes of uranium from the work place. This background contribution can vary greatly from person to person.

This background problem complicates routine (prospective) fecal sampling for natural or depleted uranium. In the case of highly enriched uranium, isotopic analysis must be performed on the samples to allow discrimination between "facility" and natural uranium.

There are, however, some circumstances (particularly in dealing with highly enriched Class Y uranium) where fecal sampling can provide much needed information. Fecal sampling should be considered where enrichments and clearance classes are such that urine sampling and in-vivo sampling do not provide the needed sensitivity. At the very least, the rapid early (1st week or so) fecal clearance rates of inhaled class Y and W materials can provide evidence that intakes are occurring.

Fecal sampling, in the event of accidental intakes of enriched nontransportable uranium, can provide an indication of the magnitude of the intake and of the rate of clearance from the pulmonary region of the lung.

Employees requested to give fecal samples should be given information regarding the necessity of this mode of sampling.

Fecal samples should be collected at home if at all possible, since minute amounts of uranium contamination can severely distort the results of the analysis. Sample collection should be made as easy and hygienic as possible for the employee. Fecal sample collection "kits" which mount on the toilet seat and permit relatively easy collection are available through medical supply companies.

6.6 Modeling the Behavior of Uranium in the Human Body

The key to the development of a sound internal dose monitoring program for uranium is an understanding of the behavior of uranium in the human body. Biokinetic models of uranium metabolism are the only link that the Health Physicist has between bioassay results and estimates of organ depositions or doses. Such models serve as the rationale for establishing appropriate bioassay frequencies, setting action levels, and assessing intakes or doses.

Remember that the models discussed in this section are very simple representations of the complex behavior of uranium taken into the body. Care should be used when dealing with the models--particularly when they are being applied to specific intake incident dose evaluation.

The behavior of uranium in the body may be best understood by breaking the body into a series of mathematical compartments between which activity is transferred. Inhaled activity is deposited in the respiratory tract where it is either retained or transferred into the bloodstream or GI tract. Activity deposited in the GI tract (through either ingestion or transfer from the lung) may be absorbed into the bloodstream or be excreted. Finally, activity deposited into the bloodstream (from the lung, GI tract, direct injection, or recycling) is either transferred to various organs or excreted via the urine.

The behavior (distribution and retention) of uranium in these three major compartments [respiratory tract, GI tract, and systemic (blood and internal organs) circulation] is a function of the chemical and physical form of the uranium. Different retention and excretion patterns are seen for the range of uranium compounds commonly encountered in uranium facilities. A knowledge of the chemical and physical forms present in a facility is critical in establishing appropriate internal dose monitoring systems.

6.6.1 Respiratory Tract

Although uranium may enter the body through inhalation, ingestion, absorption through the skin, or injection, inhalation is the most common route. Figure 6-2 illustrates the ICRP 30 respiratory tract model, a system of pathways by which inhaled uranium reaches the organs of interest, and is finally excreted via urine or feces.

The model may be broken into two functional parts: deposition or activity into the three lung regions and retention of that activity by each of the compartments of the lung.

Deposition

According to the model, the initial deposition of activity into either the naso-pharyngeal, tracheo-bronchial, or pulmonary region of the lung is solely a function of the particle size distribution of the inhaled material. This particle size distribution is represented by an Activity Median Aerodynamic Diameter (AMAD), expressed in microns. The respiratory tract model assumes that the activity inhaled is in the form of a log-normally distributed aerosol with an AMAD of 1 micron and a standard geometric deviation of less than 4.5 microns, unless better information is available. The IAEA, in its Technical Report Series publication No. 142, provides regional deposition estimates for monodisperse particulates. These can be applied when the aerosol is not log-normally distributed and the particle size distribution is known.

Figure 6-3 (ICRP 1979) shows the deposition in the three lung regions as a function of AMAD. Generally, the smaller the AMAD, the more activity is deposited in the pulmonary region and the less activity is deposited in the naso-pharyngeal region. The fraction of activity deposited in the tracheo-bronchial region is more or less constant.

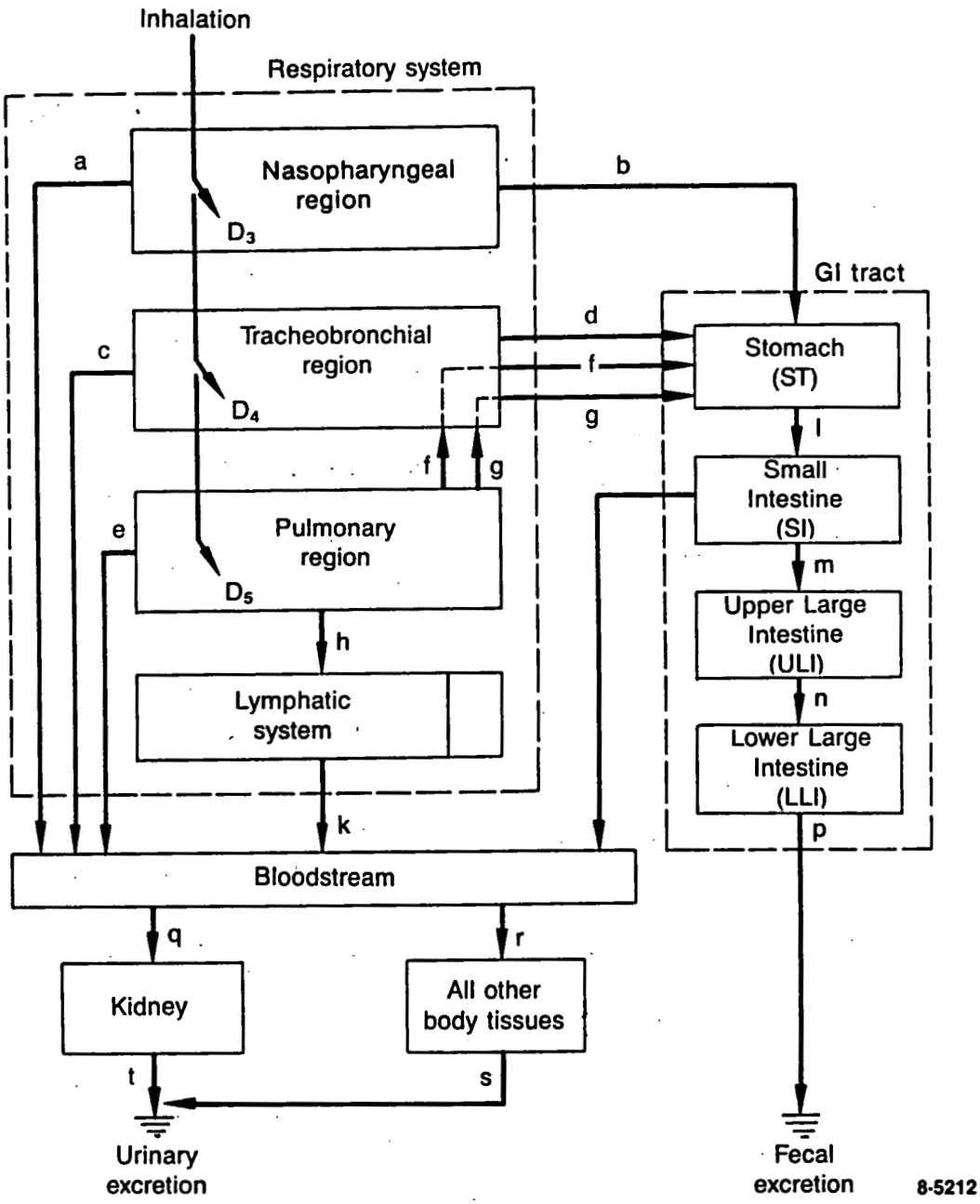

Figure 6-2. Schematic diagram of the metabolic model.

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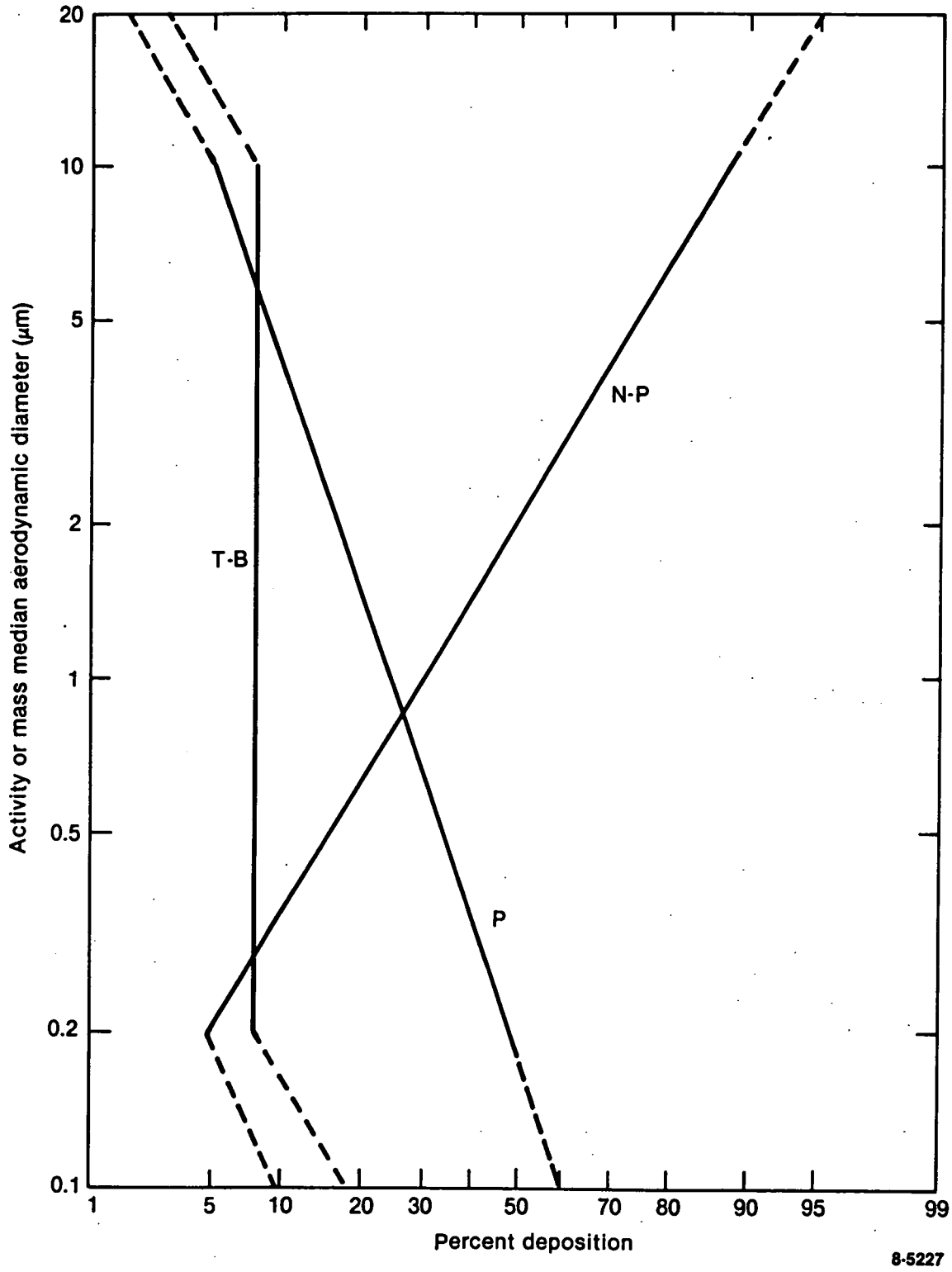

Figure 6-3. Deposition fractions.

Figure 6-4 illustrates how lung retention is a function of particle size distribution. It is clear that the retention and excretion of uranium are highly dependent upon the initial pattern of deposition. The interpretation of bioassay results (e.g., nasal swabs, early fecal samples, urine samples) is therefore also quite dependent upon particle size distribution.

Retention

Once deposited, the retention of inhaled uranium in the lung and the metabolic "fate" of that uranium is determined by the transportability (solubility) of the particular chemical/physical form of uranium. As seen in Figure 6-2, uranium deposited into the lung may be either absorbed into the bloodstream or transferred by various clearance mechanisms to the Gastro-Intestinal (GI) Tract. The fraction of activity from each lung compartment which goes in each "direction", and the biological half-time associated with this movement of activity are determined by the relative transportability of uranium.

The ICRP has established three "clearance classes" which describe the behavior of deposited activity in the lung. These classes are based on the biological clearance half-time from the pulmonary region of the lung as follows:

Clearance Class	Clearance Half-Time from Pulmonary Region (days)
Y (years)	> 100
W (weeks)	10-100
D (days)	< 10

As seen in Figure 6-5, each clearance class dictates compartmental fractions (F) and biological clearance half-times (T) for each of the paths by which activity is removed from the lung. Table 2.11, Section 2, (page 2-36) lists recommended clearance classes for common compounds of uranium.

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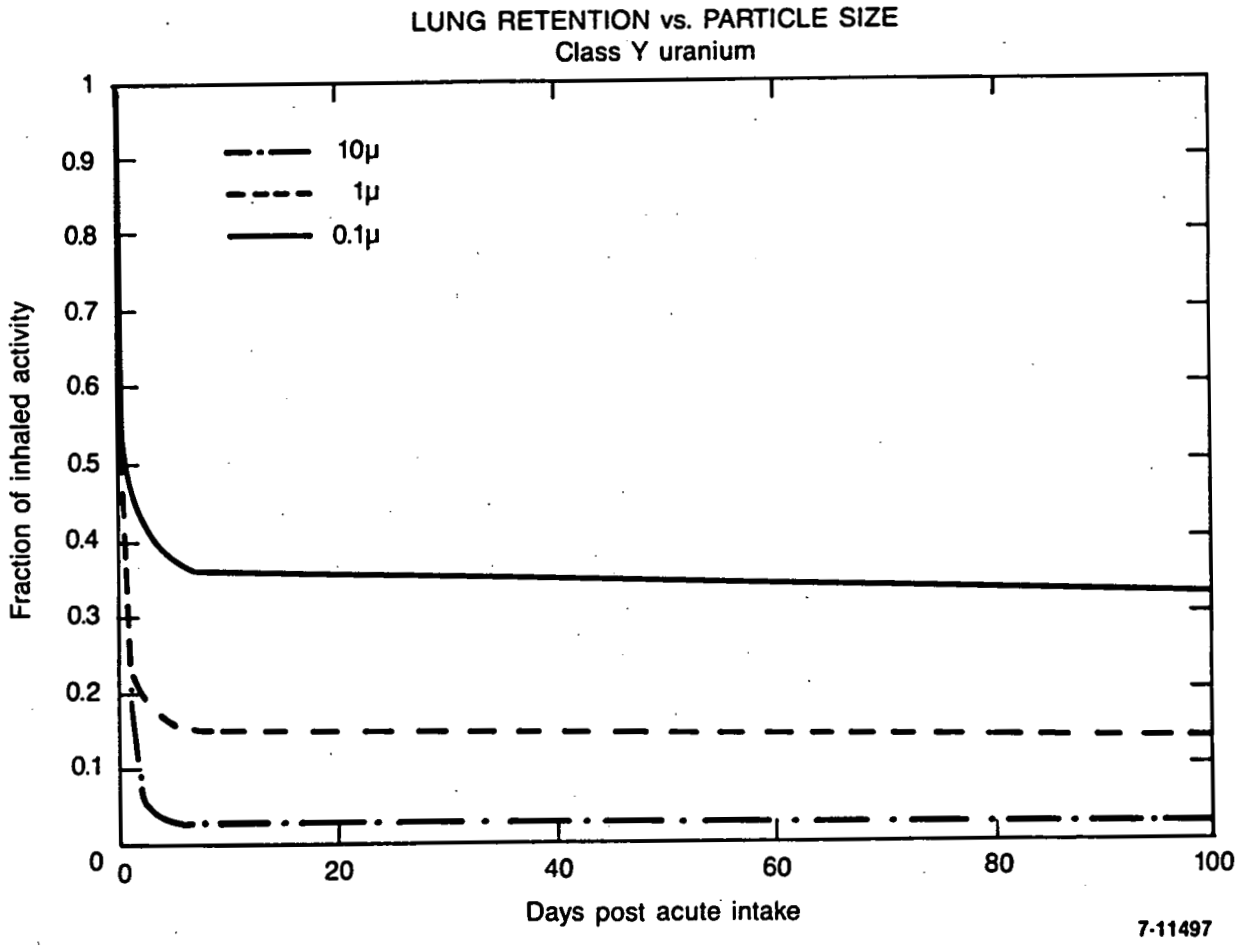
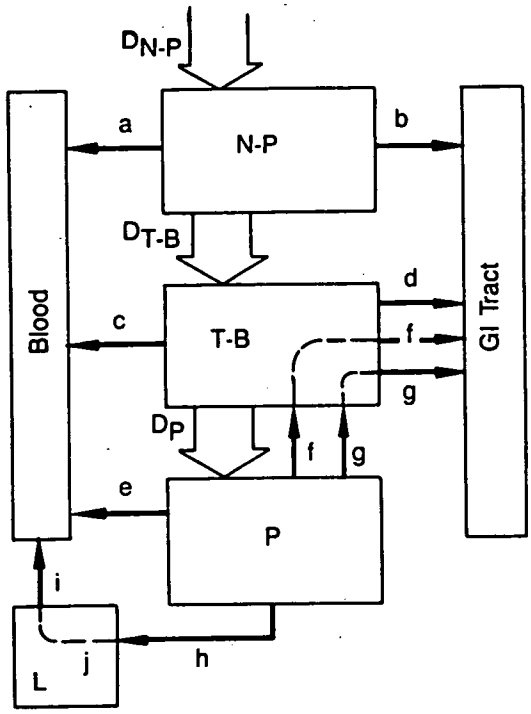


Figure 6-4. Lung retention vs particle size.



Region	Compartment	Class					
		D		W		Y	
		T	F	T	F	T	F
N-P (D_{N-P} =0.30)	a	0.01	0.5	0.01	0.1	0.01	0.01
	b	0.01	0.5	0.40	0.9	0.40	0.99
T-B (D_{T-B} =0.08)	c	0.01	0.95	0.01	0.5	0.01	0.01
	d	0.2	0.05	0.2	0.5	0.2	0.99
P (D_P =0.25)	e	0.5	0.8	50	0.15	500	0.05
	f	n.a.	n.a.	1.0	0.4	1.0	0.4
	g	n.a.	n.a.	50	0.4	500	0.4
	h	0.5	0.2	50	0.05	500	0.15
L	i	0.5	1.0	50	1.0	1000	0.9
	j	n.a.	n.a.	n.a.	n.a.	∞	0.1

ICRP RESPIRATORY MODEL 7-11465

Figure 6-5. ICRP respiratory model.

000237

Figure 6-6 illustrates how the retention of uranium activity in the lung varies with clearance class. The dramatic differences in clearance patterns have obvious implications for interpretation of in-vivo counting and other bioassay results. Note that the overall retention in the first day or so is controlled by the rapid clearance of activity from the NP and TB regions of the respiratory tract. This graph may be used for U-234, U-235, or U-238 since the clearance of activity from the lung is completely dominated by biological, not radiological half-times. Figure 6-7 illustrates how the clearance class of the material affects long-term buildup from chronic exposures to airborne contaminates. Note that these curves reflect only the activity in the pulmonary and lymph regions of the respiratory tract.

6.6.2 Gastro-Intestinal Tract

Figure 6-8 shows the GI Tract model recommended by the ICRP. Uranium activity entering the GI tract from either the respiratory tract or directly from ingestion is either excreted via the feces, or is absorbed into the bloodstream. The fraction of uranium activity absorbed from the small intestine to the bloodstream is a function of the clearance class (transportability) as follows:

Clearance Class	f1 Fraction of Activity Transferred to Bloodstream
D	0.05
W	0.05
Y	0.002

Absorption of uranium from the GI tract is minimal for Class D and W uranium and almost negligible for Class Y uranium.

Inspection of the respiratory and GI tract models reveals a number of predicted features which affect the management of an internal dose

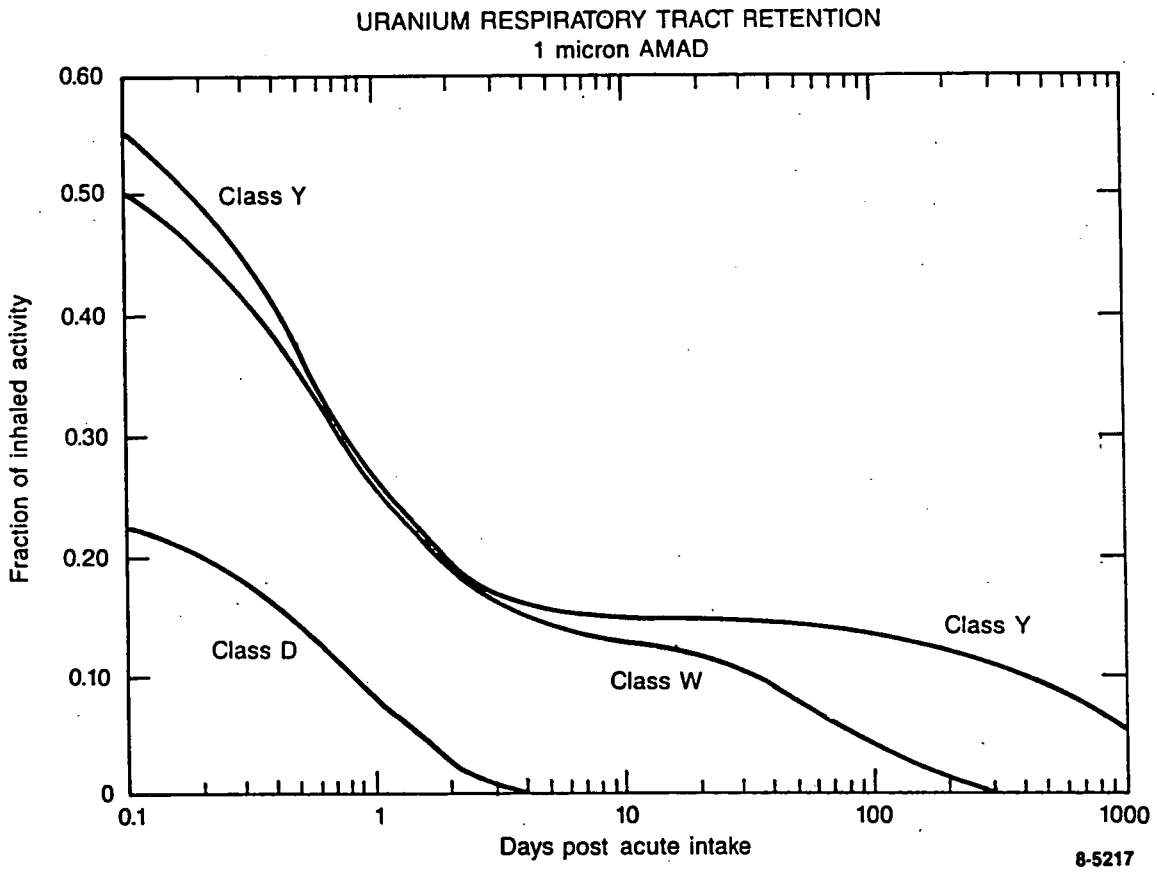


Figure 6-6. Uranium respiratory track retention.

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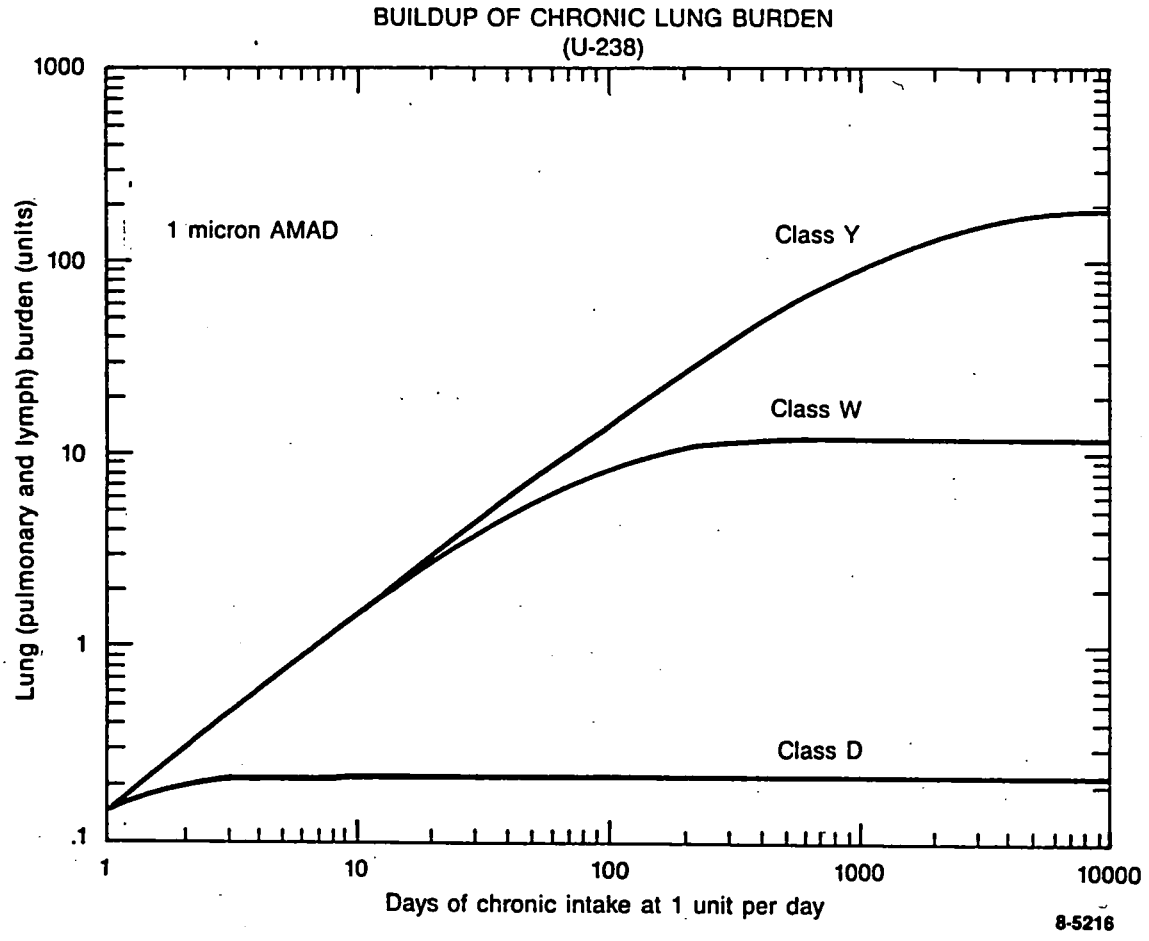
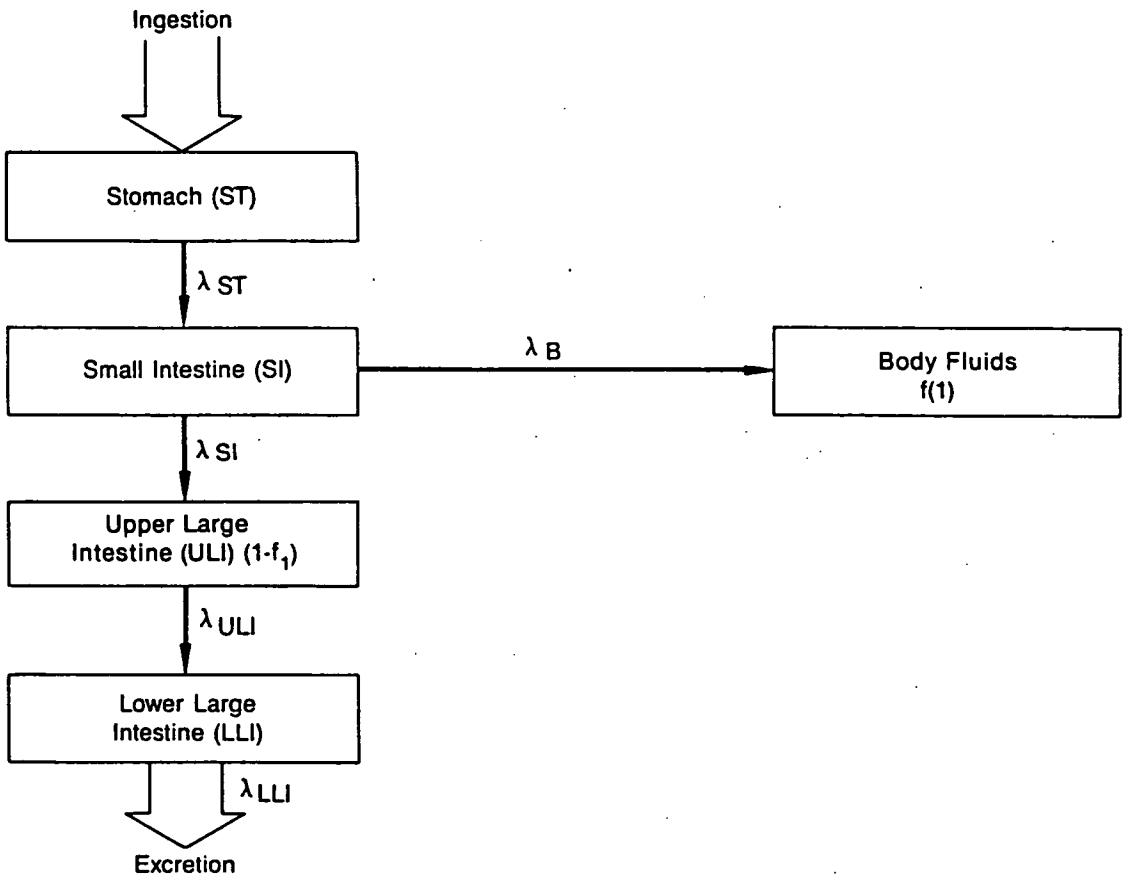


Figure 6-7. Buildup of chronic lung burden.



Section of GI tract	Mass of walls* (g)	Mass of contents* (g)	Mean residence time (day)	λ day⁻¹
Stomach (ST)	150	250	1/24	24
Small Intestine (SI)	640	400	4/24	6
Upper large Intestine (ULI)	210	220	13/24	1.85
Lower Large Intestine (LLI)	160	135	24/24	1

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Figure 6-8. ICRP-30 gastro-intestinal tract model.

monitoring program. It is worth noting that, for almost all clearance classes and particle sizes, a large fraction of inhaled activity will appear in the feces within the first week or so after intake.

By coupling this GI tract model to the previously discussed lung model, one can derive retention functions which predict the quantity of activity in the feces as a function of time after intake. Figures 6-9 and 6-10 show the predicted excretion patterns for acute and continuous intakes. Both of these figures may be used for either U-234, U-235, or U-238.

6.6.3 Systemic Retention of Uranium

Understanding and predicting the behavior of uranium deposited into systemic distribution (bloodstream and organs) play a critical role in assessing doses from intakes of uranium. Unfortunately, the behavior of uranium in the body, and subsequent excretion patterns are complex and are still the subject of on-going research.

A detailed review of models used to predict systemic uranium behavior is beyond the scope of this manual. Detailed discussions are available (Hursh 1973, Alexander 1974) and an excellent recent review of the history and development of systemic models may be found in USUR-05/HEHF-47 (Durbin 1984). A brief discussion of ICRP-10 and ICRP-30 metabolic models follows.

ICRP-10 Retention and Excretion Functions

ICRP Publication 10 (ICRP 1968) recommended the following systemic uptake retention function:

$$R(t) = 0.2\ t^{-0.5} \text{ for values of t greater than 1 day post intake.}$$

R(t) represents the fraction of the initial uptake to systemic distribution which is present in the system at any day t after intake. Since this is an uptake retention function, the value of the R(t) at t = 0 is, by definition, 1.0.

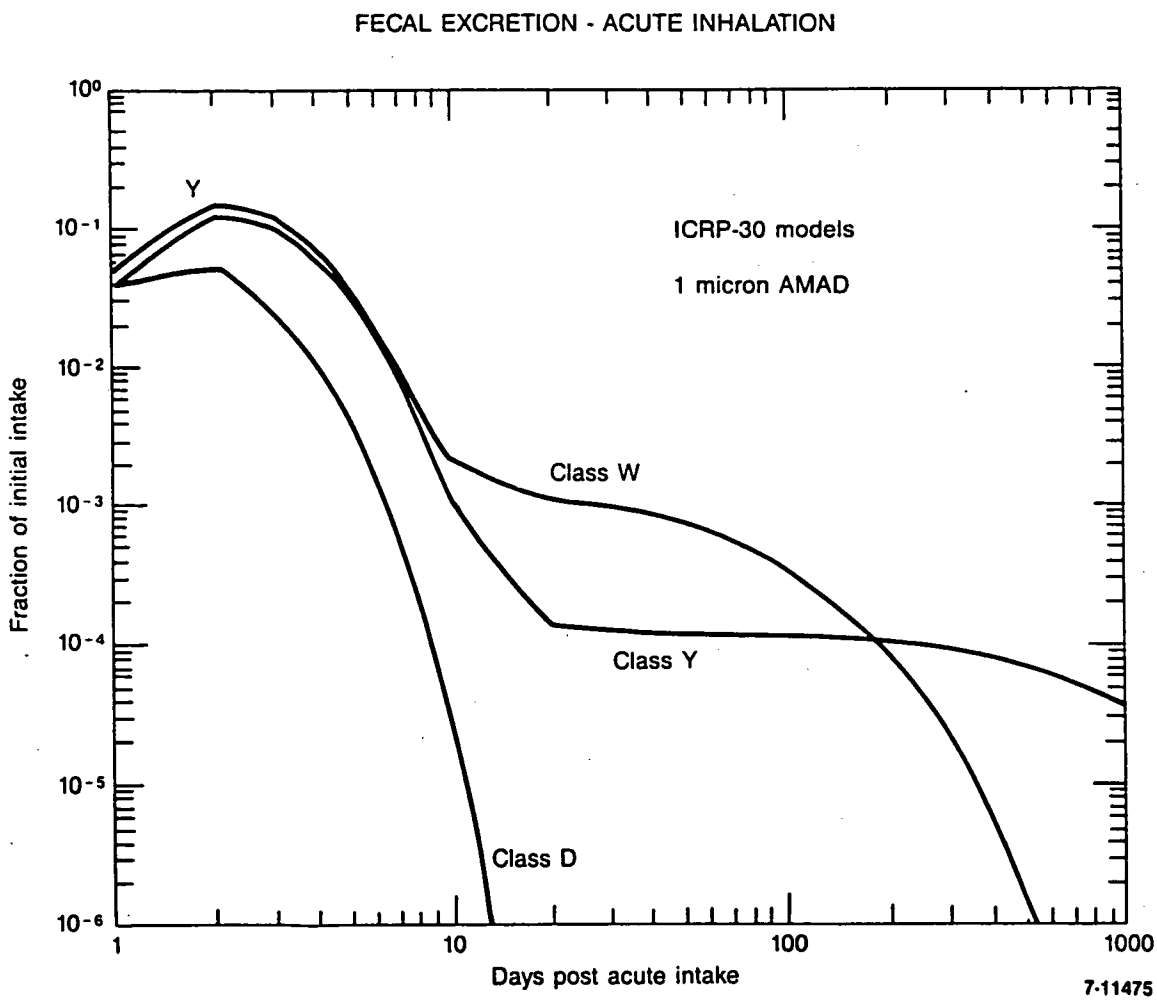

Figure 6-9. Fecal excretion--acute inhalation.

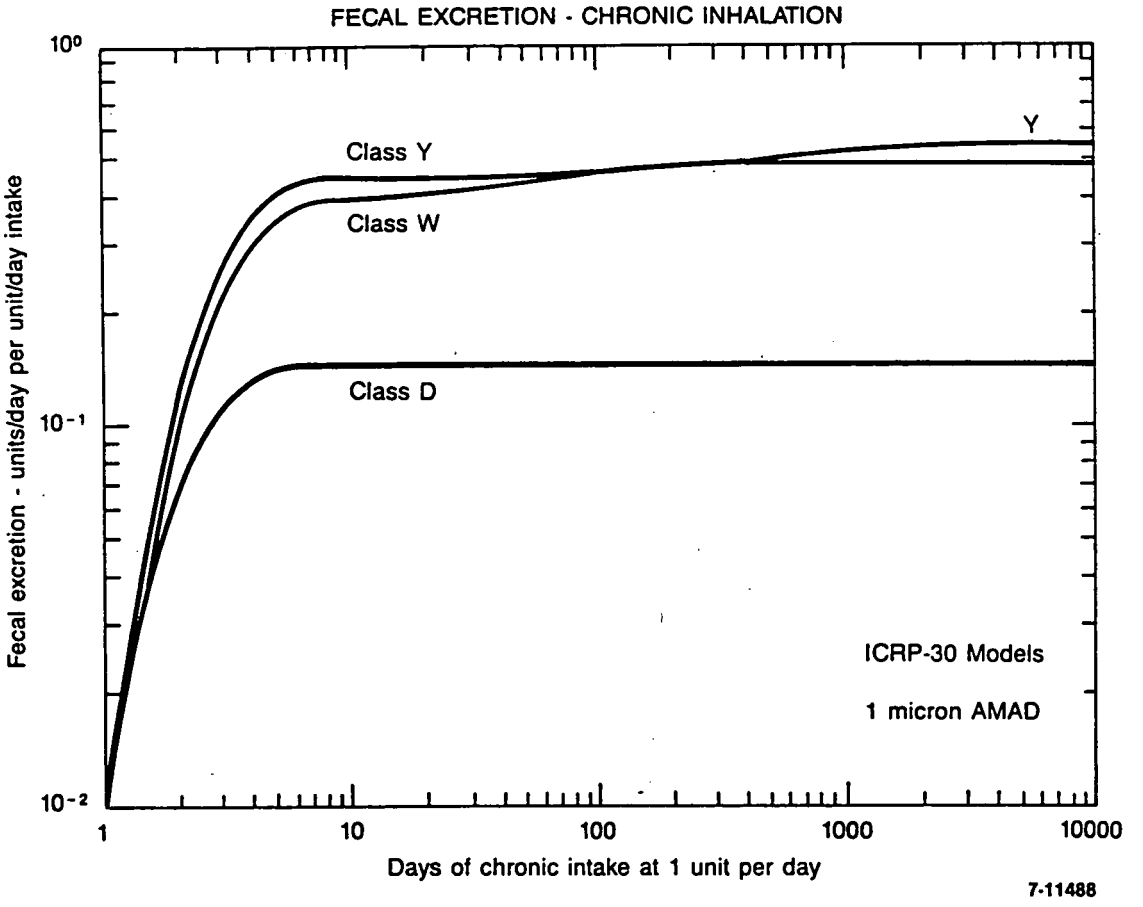

Figure 6-10. Fecal excretion--chronic inhalation.

A related fractional excretion function:

$$Y(t) = 0.8 \quad \text{for } t = 1 \text{ day}$$

$$Y(t) = 0.1\ t^{-1.5} \quad \text{for } t > 1 \text{ day}$$

is also presented. Since this fractional excretion function is simply the first derivative of the uptake retention function, it represents the instantaneous fraction of the initial uptake which is excreted at any time greater than 1 day.

Note that, with the possible exception of Class D materials, these functions must be coupled with those of the lung model in order to be used to interpret bioassay results from inhalation intakes.

ICRP-30 Systemic Retention Model

The ICRP has published newer metabolic models and retention functions to be used in establishing limits of intakes for workers. Figures 6-11 and 6-12 represent the formal metabolic model implied by the retention functions and statements of ICRP-30. According to this model, of uranium entering the bloodstream (or "transfer compartment") about 22% goes to bone, where it is assumed to be uniformly distributed throughout the volume of mineral bone. About 12% of the uranium goes to the kidneys, about 12% goes to "all other tissues of the body," and the remainder (about 54%) is assumed to go "directly to excretion."

The corresponding retention functions are as follows:

$$R(\text{bone}) = 0.2\ e^{-0.693\ t/20} + 0.023\ e^{-0.693\ t/5000}$$

$$R(\text{kidney}) = 0.12\ e^{-0.693\ t/6} + 5.2E\text{-}4\ e^{-0.693\ t/1500}$$

$$R(\text{other}) = 0.12\ e^{-0.693\ t/6} + 5.2E\text{-}4\ e^{-0.693\ t/1500}$$

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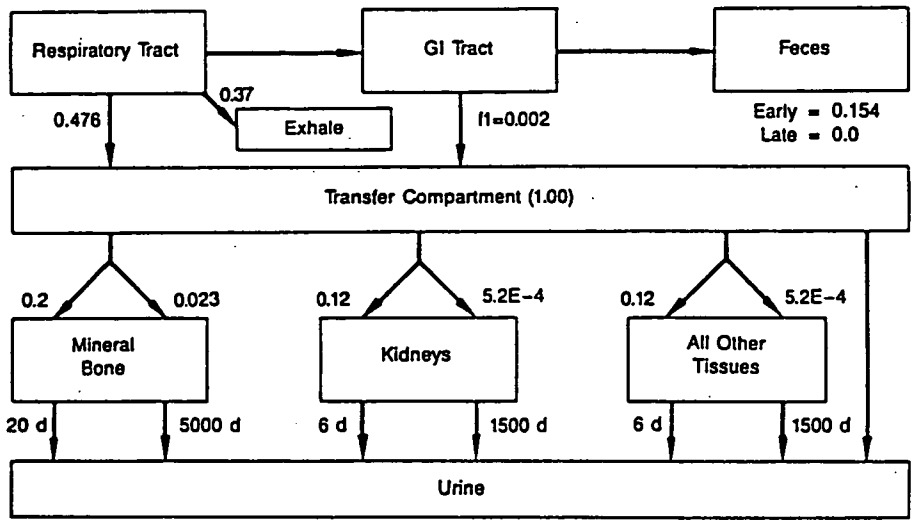
ICRP 30 METABOLIC MODEL - URANIUM

CLASS D

Chemical Form = UF_6, UO_2F_2, $UO_2(NO_3)_2$



CLASS W

Chemical Form = UO_3, UF_4, UCl_4

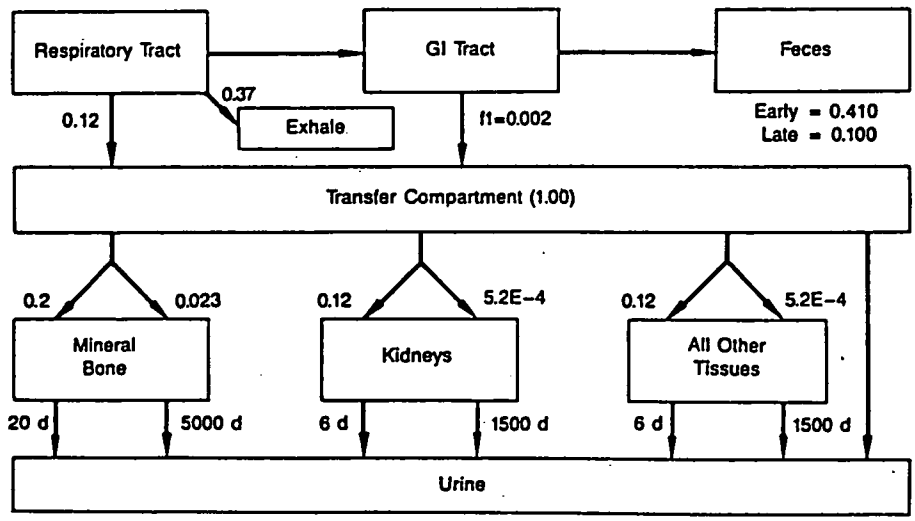


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Figure 6-11 ICRP-30 metabolic model - Class D and W.

ICRP 30 METABOLIC MODEL

Radionuclide = U-238 Class Y

Chemical Form = UO_2, U_3O_8

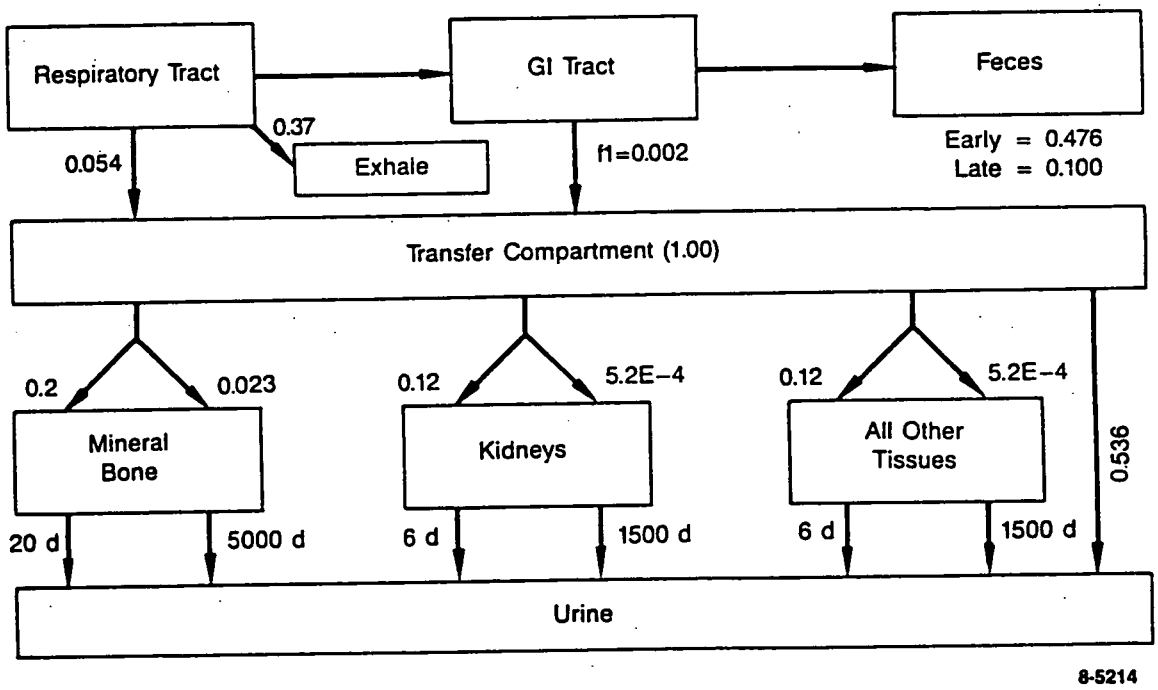

Figure 6-12. ICRP-30 metabolic model - Class Y.

This model is a very simplistic representation of the behavior of uranium in the body. It does not, for example, explicitly recognize any recycling of uranium from organs back to the transfer compartment. However, the retention functions of ICRP-30 are "effective" uptake retention functions, that is, they were developed to fit observed retention and excretion patterns. The behavior described by these functions must therefore include all of the complexities of distribution within the body, including recycling.

The uptake retention functions presented in ICRP-10 and ICRP-30 are compared in Figure 6-13. These curves represent the fraction of an acute uptake to the blood that is retained on any day post intake.

The corresponding uptake fractional excretion functions seen in Figure 6-14 represent the fraction of the acute uptake which is excreted at any day after intake. Note that these fractional excretion functions give "instantaneous" excretion rates, not incremental "24 hour" excretion rates.

Coupling the lung model with different systemic retention functions will of course, give different predicted excretion patterns and correspondingly different estimates of intake. All other things being equal, the ICRP-10 function will give higher estimates of intake (from about 3 to 100 days) because the predicted fractional excretion is lower than that of ICRP-30.

Prediction of Excretion Rates

The systemic retention functions are commonly coupled with the respiratory tract model and used to predict the rates of excretion from intakes. The Nuclear Regulatory Commission and others (Alexander 1974, King 1977, King 1979) have used the retention function of ICRP-10 coupled with the ICRP-30 lung model to predict excretion and derive reference levels and bioassay frequencies. More recent analyses (Lippman 1981, Wrenn 1985, Alexander 1986) have used models which differ somewhat from those of either ICRP-10 or ICRP-30.

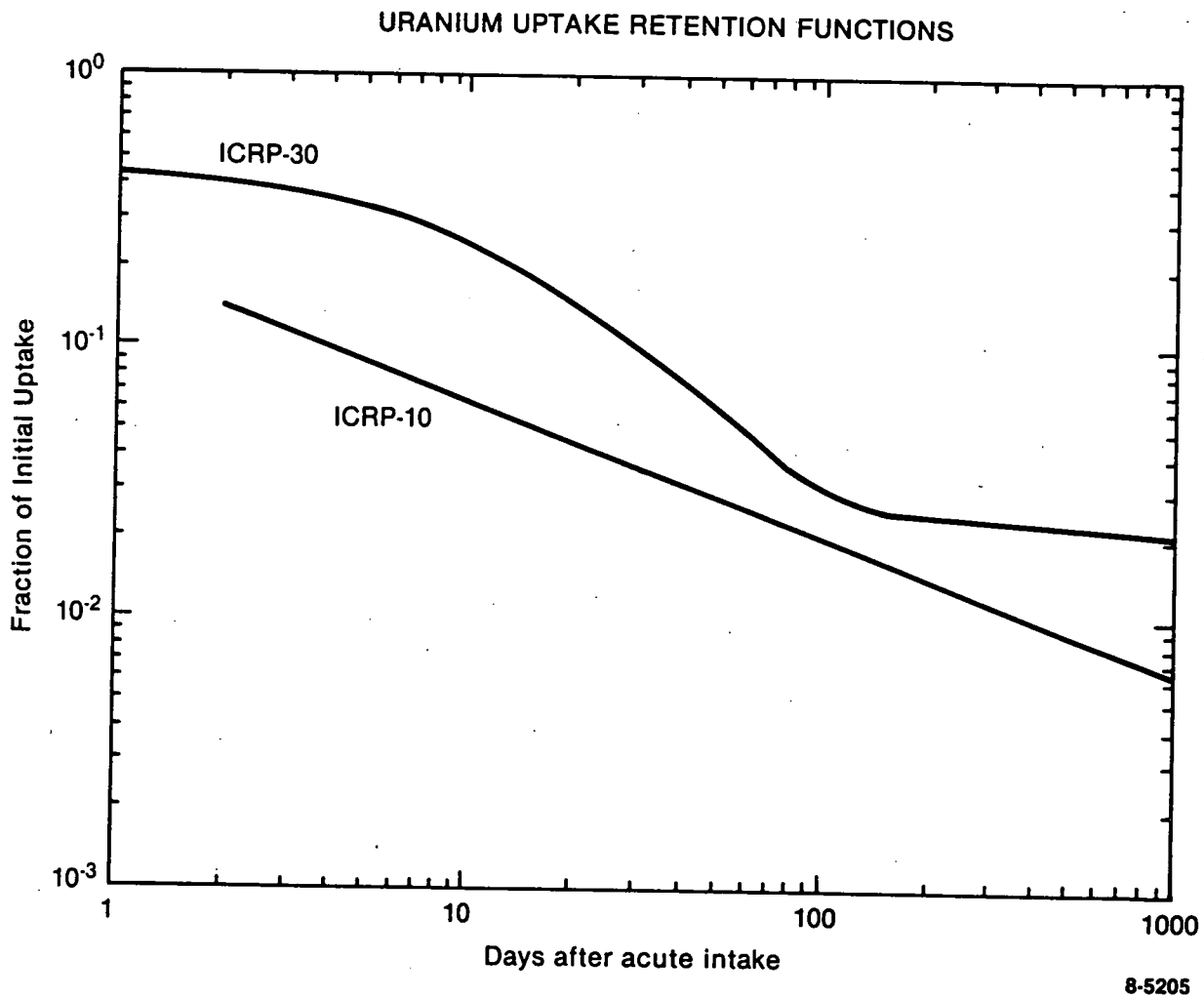

Figure 6-13. Uranium uptake retention functions.

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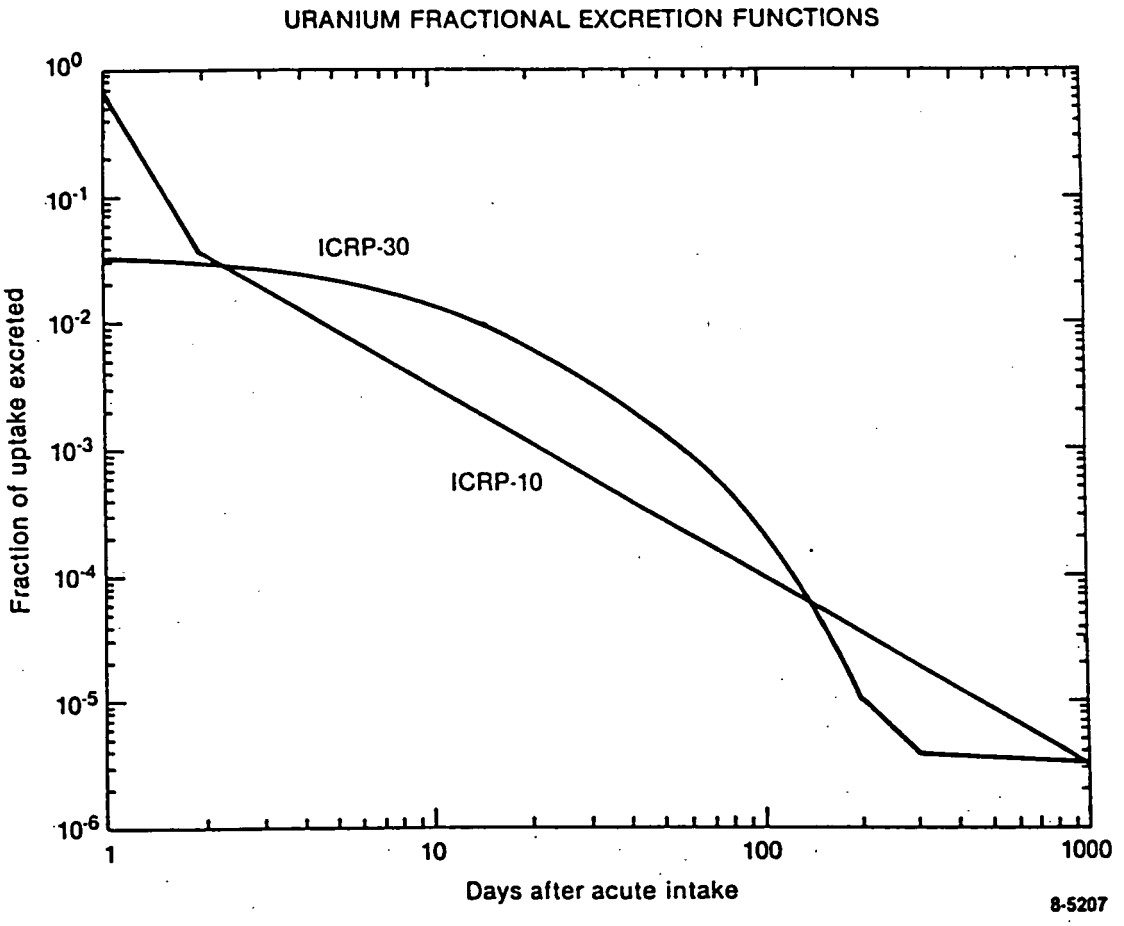

Figure 6-14. Fractional uranium urinary excretion functions.

This manual presents predictions of excretion based on the ICRP-30 lung model coupled to the ICRP-30 systemic retention model for uranium. In doing so, it is recognized that the systemic models of ICRP-30 were not necessarily intended to be used to predict excretion patterns or to predict intake from bioassay results. It is felt, however, that the use of such models produces estimates of intake which are sufficiently accurate to be used to establish reference levels and bioassay frequencies. This use is consistent with procedures used at several major uranium handling facilities.

Great care should be used in extending such use to estimates of intake and resulting dose for individuals. When practicable every attempt should be made to match the model and/or model parameters to the individual and actual excretion rates observed.

Figure 6-15 shows the predicted urine excretion rates for an acute intake of Class D, W, and Y, 1 micron AMAD uranium. The daily fractional excretion values have been divided by 1.4 to account for the average daily urine volume of male workers.

In a similar manner, Figure 6-16 shows the predicted urinary excretion produced by chronic intake of 1 unit (μg, d/s, etc.) per day. These values have also been adjusted to take into account the average daily urine volume. Thus, for example, a chronic intake of 1 μg/day of Class W uranium would produce a urine concentration of about 0.1 μg/liter at the end of 200 days of chronic intake. Since these curves include the rapid "direct" excretion component, they should be used only for truly chronic intakes and not intermittent (i.e., work week intakes).

Kidney Retention Functions.

The quantity of uranium present in the kidney is of interest because of chemical toxicity considerations. In the following figures, the ICRP-30 lung model has been coupled to the ICRP-30 retention function for the

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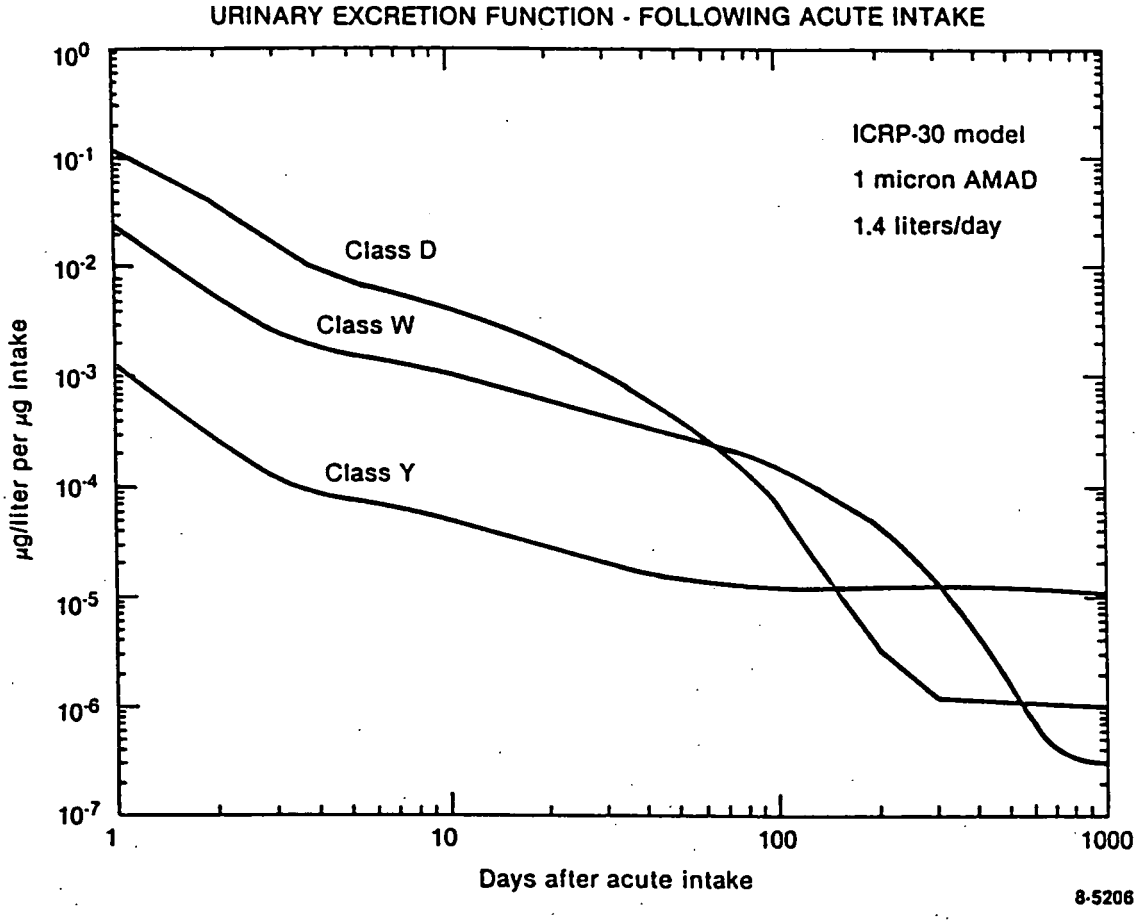
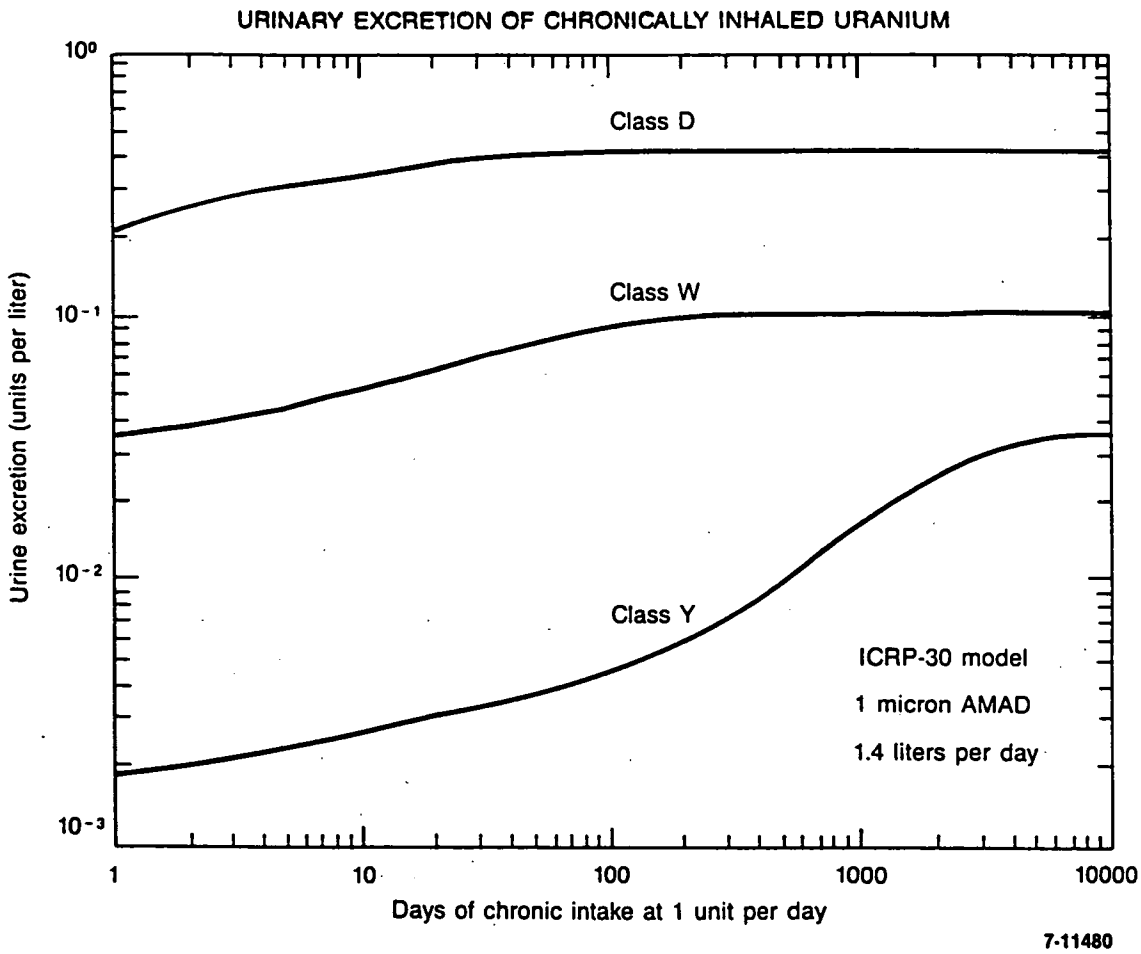


Figure 6-15. Urinary uranium excretion function - following acute intake.



Note that these curves include the rapid "direct" excretion component and are therefore only representative of truly chronic intakes.

Figure 6-16. Urinary excretion of chronically inhaled uranium.

kidneys to derive kidney intake retention functions. Although, as discussed earlier, the ICRP-30 models were not necessarily intended for such use, analysis of the model will show that it is similar to that of ICRP-10, and probably represents a reasonable estimate of the kidney burden.

Figures 6-17 and 6-18 show that the parameters which control the ICRP-30 kidney retention model are the fractional uptake from the blood (about 0.12) and the biological half-time in the kidney (essentially 6 days). In comparison, ICRP-10 uses a fractional uptake of 0.11, and a biological half-time in the kidney of 15 days. Thus, the only significant difference in the models is the biological half-time in the kidney.

A number of researchers believe that the 15-day half-time of ICRP-10 is more accurate than that of ICRP-30, and indeed, that even longer half-times may be appropriate. The ICRP-30 half-time has been retained in this discussion for the sake of consistency with the rest of the section. Although use of this 6-day half-time may not be conservative, it is adequate for prospective analyses. It is recommended that serious consideration be given to using the 15-day kidney half-time of ICRP-10 in deriving specific retention functions for retrospective dose assessments.

The use of the ICRP-30 metabolic model is felt to be sufficiently accurate to establish reference levels and bioassay frequencies. Once again, the extension of use of the ICRP-30 model to individuals should be done with great care.

Figure 6-17 shows the predicted kidney burden as a function of time after a single acute intake of 1 micron, Class D, W, or Y material.

Figure 6-18 shows the predicted kidney burden as a function of days of chronic intake at 1 unit per day.

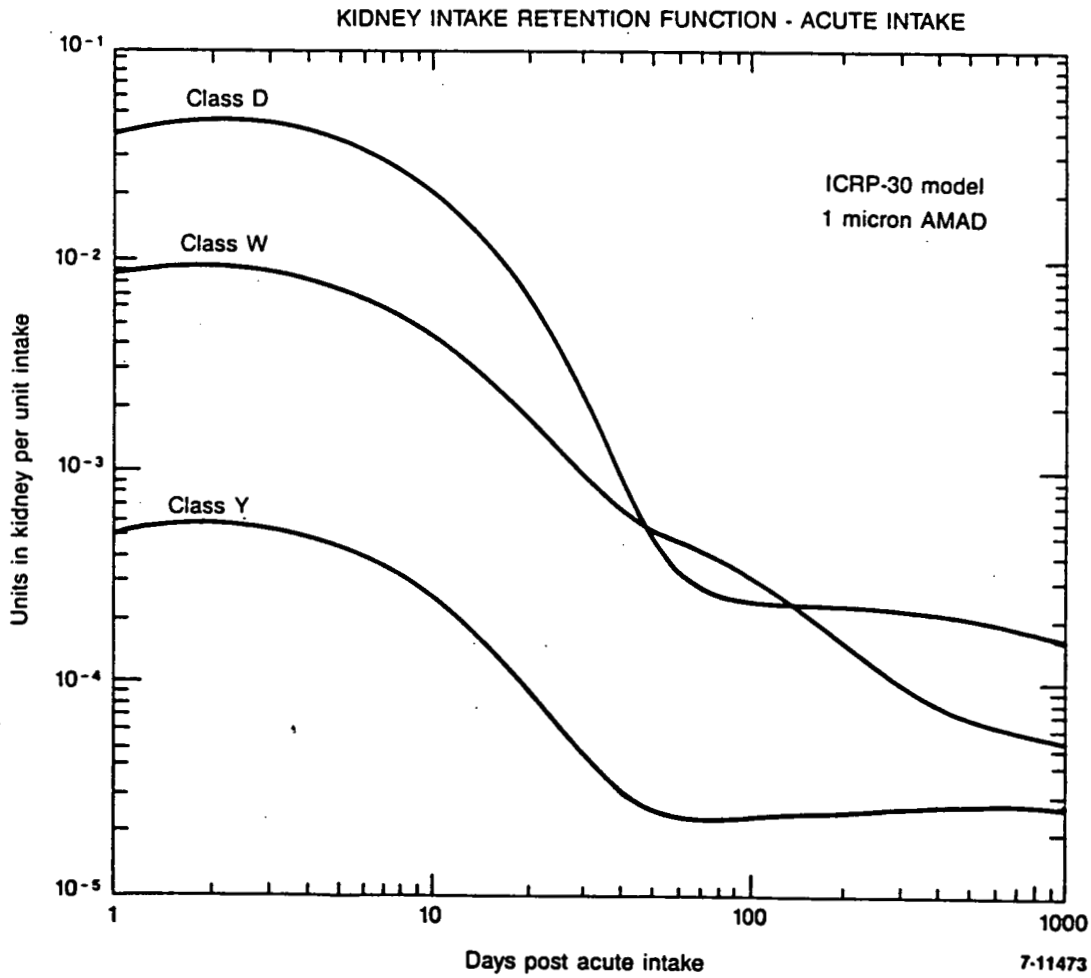


Figure 6-17. Kidney intake retention function - acute intake.

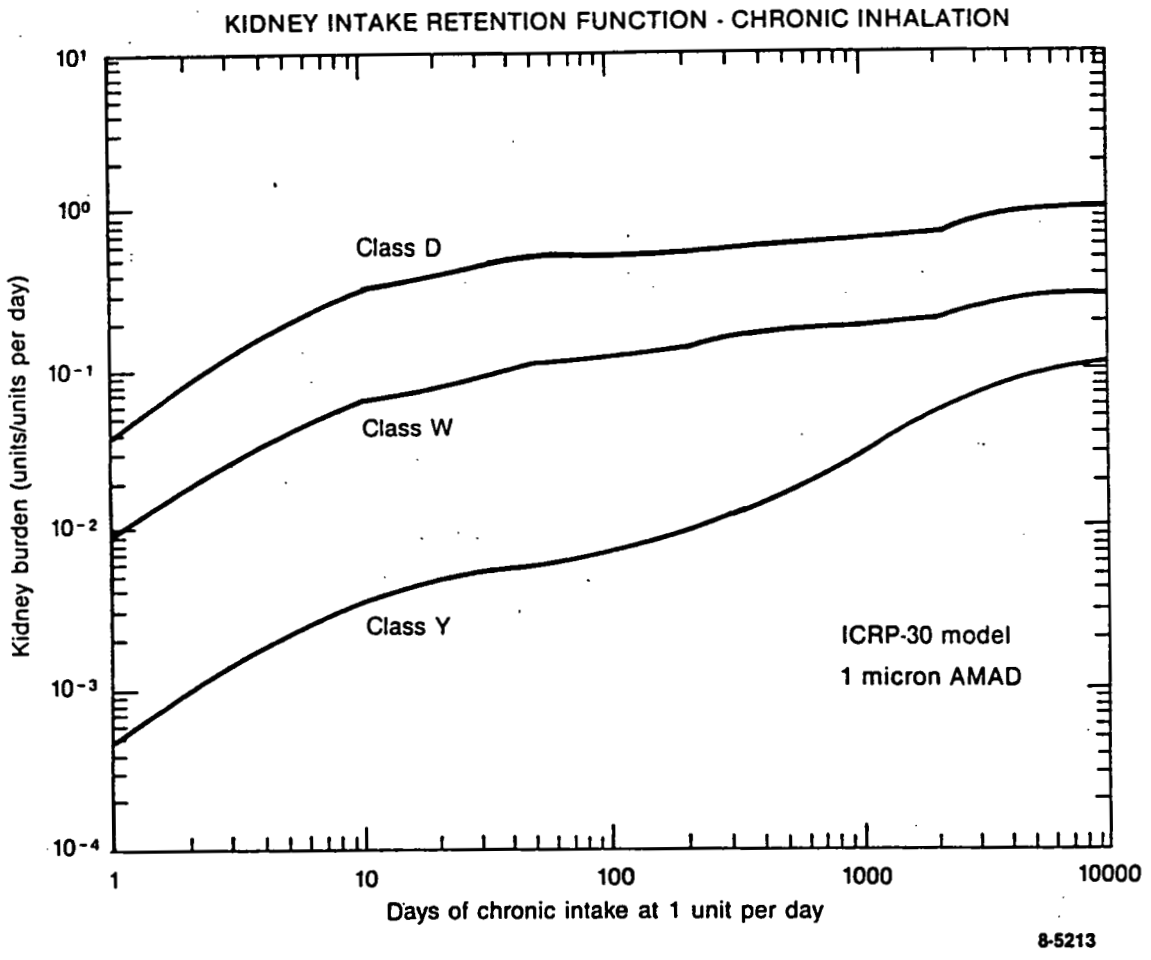


Figure 6-18. Kidney intake retention function - chronic inhalation.

6.6.4 Metabolic Model for Uranium in Bone

According to the ICRP-30 metabolic model, 0.223 of the uranium reaching the transfer compartment is assumed to go to mineral bone, where 0.20 is retained with a 20-day half-time, and 0.023 is retained with a 5000-day half-time. All of the long-lived isotopes of uranium (U-232, U-233, U-234, U-235, U-236, and U-238) are assumed to be uniformly deposited throughout the volume of mineral bone. Since this uranium is distributed evenly within the mass of trabecular and cortical bone (1 and 4 kg, respectively), it is assumed that 20% of the activity goes to trabecular bone and 80% goes to cortical bone. Figure 6-19 reflects this assumed distribution of uranium in the bone.

6.6.5 Natural Uranium Balance in Man

Uranium is present in trace quantities throughout the environment. As a result, man ingests about 2 μg of natural uranium each day in food and fluids. A similar quantity is excreted each day in the feces and urine. The uranium balance for reference man is as follows: (ICRP-23)

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

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.

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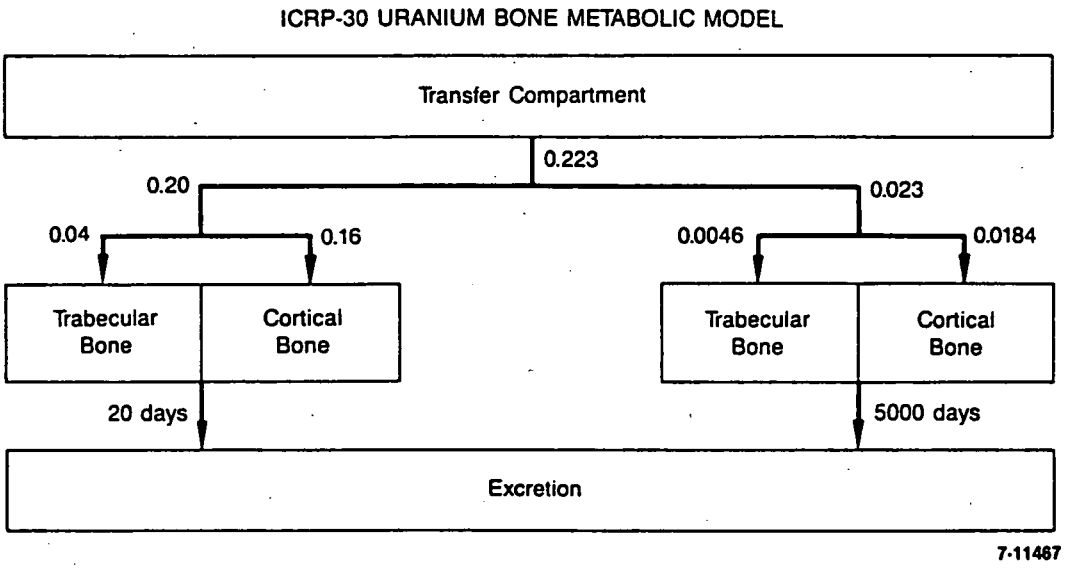

Figure 6-19. ICRP-30 uranium bone metabolic model.

6.7 Interpretation of Bioassay Results

6.7.1 In-Vivo Count Results

The essence of interpretation of In-Vivo counting results is to use an assumed or observed retention function to relate the amount of activity observed in the lungs to the amount initially deposited or inhaled. Standard models (i.e., the ICRP-30 Respiratory Tract Model) may be used initially to set action levels and to provide guidance regarding follow-up actions. "Customized" retention functions derived from observed clearance patterns should be used whenever possible for interpretation of significant results for individuals. For example, if the activity deposited in the respiratory tract appears to be clearing with a 100 day half-time, that observed half-time should be incorporated into the retention functions used to estimate intake and dose.

Figures 6-20 and 6-21 are examples of intake estimators which may be derived from respiratory tract retention functions discussed in Section 6.6. Figure 6-20 provides an estimate of the quantity of activity initially inhaled per nanocurie of activity detected in the respiratory tract as a function of days post-acute intake.

Figure 6-21 allows one to estimate the daily intake rate which would produce an observed lung count result after so many days of chronic intake. For example, assume that 1 nci of uranium is observed in the respiratory tract after 100 days of what is assumed to be chronic intake of Class Y material. The Class Y curve shows that a daily intake of about 0.07 nci would produce such a result.

Care must be taken to assure that the assumptions underlying such figures are understood. For example, the curves in Figure 6-20 are derived from the respiratory tract retention functions of Figure 6-6. These retention functions include activity deposited in the NP and TB regions of the lung. Use of these functions implies that the In-Vivo detection system

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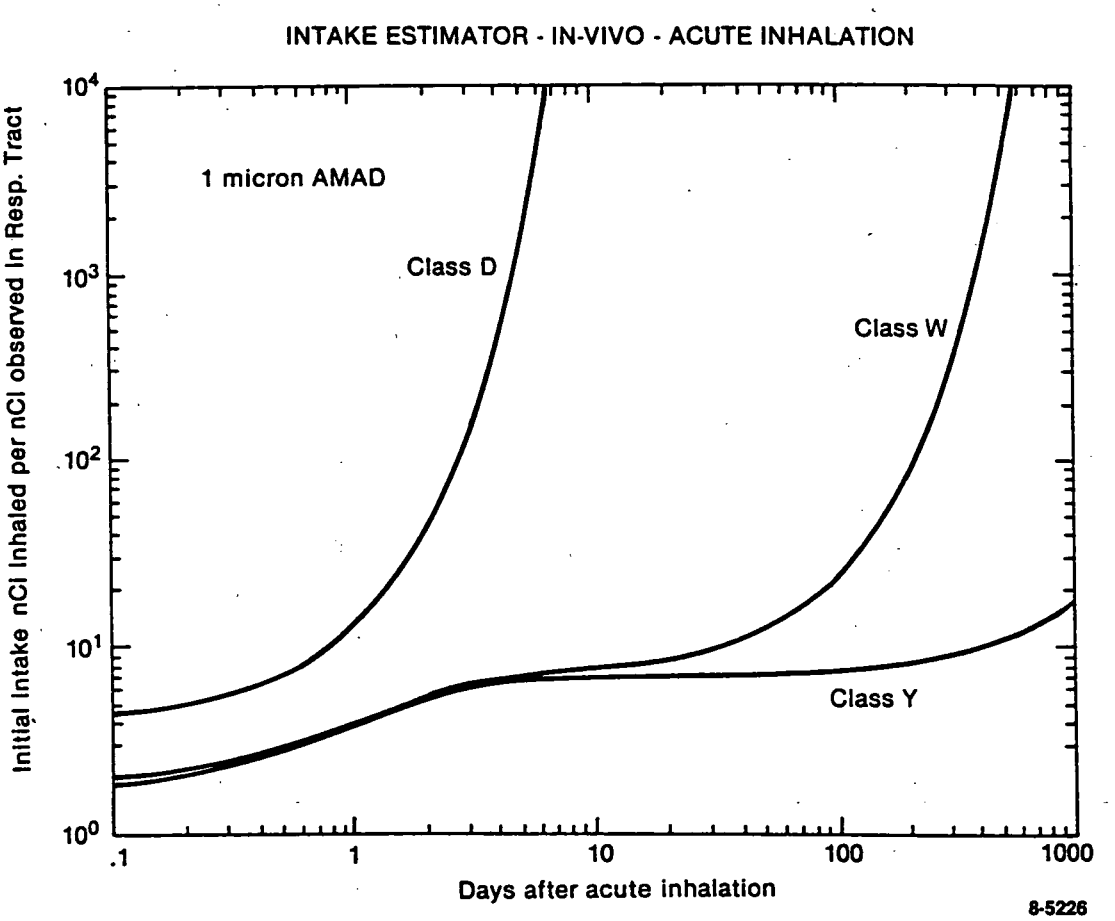

Figure 6-20. Uranium in-vivo intake estimator.

URANIUM IN-VIVO INTAKE ESTIMATOR - Example for Use

1. The equivalent of 2 nanocuries of total uranium activity is "seen" in a lung count taken about 100 days after an acute intake of suspected 1 micron Class W uranium. What was the initial quantity of activity inhaled?

   From Figure 6-20 at 100 days post intake, the initial intake per nanocurie detected in the respiratory tract is about 25 nci. Thus, the initial intake must have been about 50 nci.

2. For 1 micron Class Y uranium, what intake could be "missed" if routine lung counts are performed annually, and the detection limit (for total uranium alpha activity deposited in the lung) is 1.5 nanocuries?

   At 365 days, the corresponding initial intake for Class Y uranium is about 10 nci per nanocurie observed. If 1.5 nanocuries could be "just missed", then the corresponding intake is 15 nci.

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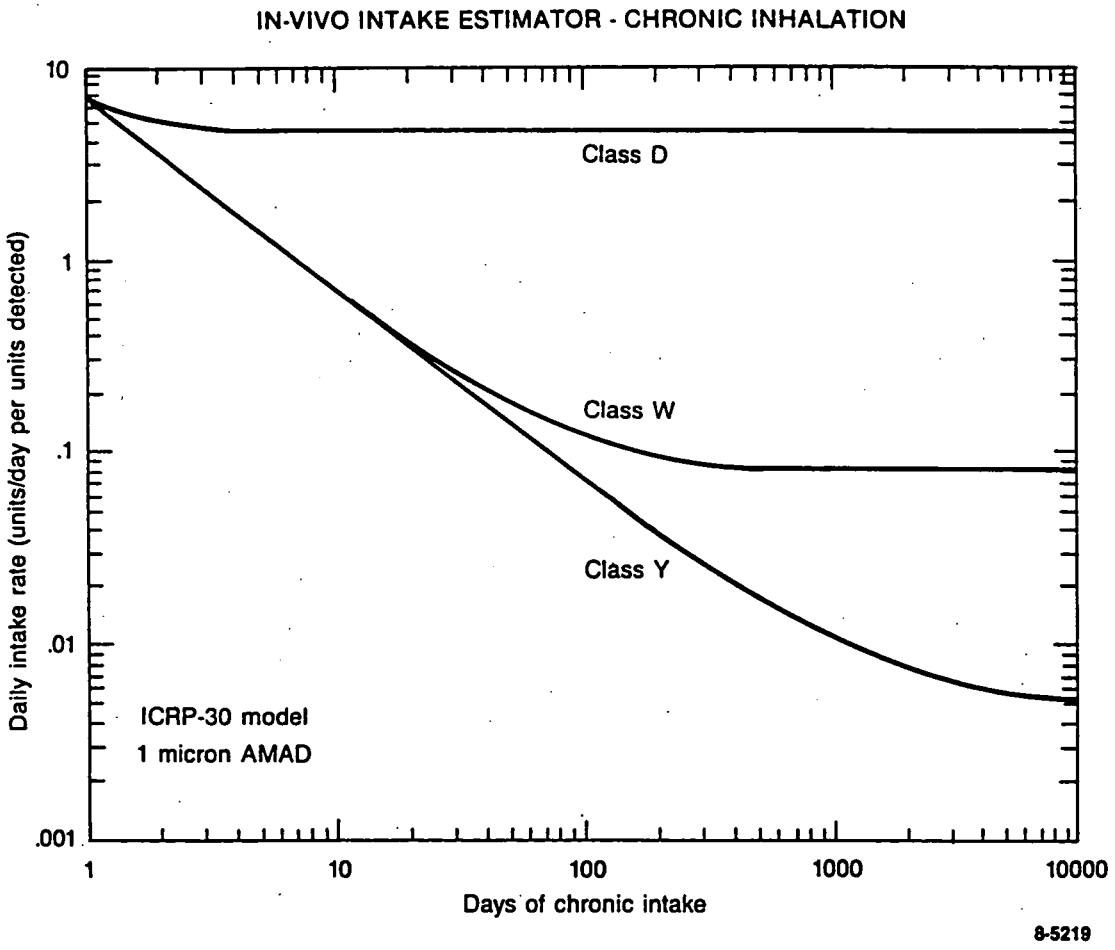

Figure 6-21. In-vivo intake estimator - chronic intakes.

can detect activity deposited in the NP and TB regions. Whether or not this is the case depends upon the nature and geometry of the counting system. (Since the NP and TB regions will clear within a day or so, this particular problem only exists in the first day or so after intake.)

6.7.2 Urine Sample Results

Interpretation of urine sample results can be more of an art than a science. All of the factors discussed previously (route and mode of intake, particle size and transportability) can affect the rate at which uranium is excreted in the urine. Add to these factors the uncertainties in the metabolic models, coupled with individual differences in metabolism, (not to mention uncertainties associated with collection and sample analysis) and the difficulties become obvious. Notwithstanding such difficulties, the significance of urine sample results must be determined, and estimates of intake and dose must be made for dose assessment purposes.

Interpretation for Prospective Monitoring

For general planning purposes (such as establishing sampling frequencies and setting action levels) the use of retention/excretion functions derived from standard metabolic models is appropriate. Any site specific information available (e.g., typical particle size, or mix of transportability classes) should be incorporated in the models to the extent feasible.

The interpretation of results of samples taken for prospective monitoring purposes is generally limited to the following concerns:

1. Is there statistically significant activity in the sample?

2. Is the activity present statistically greater than what would be expected due to excretion of naturally occurring uranium? (This question will assume a more important role as more sensitive urinalysis techniques are developed.)

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3. If the answers to both (1) and (2) are "yes," then what further actions are warranted by these results? (Action levels.)

Most urine samples collected for prospective monitoring purposes will be "spot samples." Typically, the concentration of uranium observed is related to the assumed "Reference Man" daily output of 1.4 liters per day. Corrections for sex (1.0 vs. 1.4 l/day for females and males respectively) and for daily variations (creatinine clearance) are usually not warranted.

In a similar manner, action levels for interpretation of such exposure control samples are typically set in a very general manner, using standard metabolic models applied to "Reference Man" parameters. Such a general application of the models is usually appropriate in these circumstances, provided that site-specific information (if available) is incorporated into the models. An excellent example of this site/hazard specificity can be found in NUREG-0874 where both models and parameters specific to uranium milling operations were used to establish action levels.

Interpretation of Samples for Retrospective Monitoring

The purpose of retrospective or follow-up sampling is to either confirm a suspected intake, or to provide data for assessing intake, deposition, or dose. Accordingly, care must be exercised when applying standard retention/excretion functions to diagnostic situations where a specific individual is involved. In such instances, a 24-hour or "simulated 24-hour" sample should be used in conjunction with incremental excretion functions. "Simulated 24-hr samples" typically involve collecting the first and last voidings for two consecutive days. Creatinine clearance ratio methods may be used to normalize sample results. Sex-specific average daily output volumes should be used. Any available information about particle size or transportability should be incorporated into the interpretation process.

Application of Retention/Excretion Functions

The incremental urine excretion functions of Section 6.6.3 may be used to establish estimates of intake and dose, sampling frequencies, and reference levels. Each of the following intake estimators follows directly from the excretion functions of Section 6.6.3.

Figure 6-22 shows the estimated intake as a function of days post acute intake for a given urine sample result. The estimate of intake is derived as follows:

$$I = \frac{E \times 1.4}{IRF}$$

where

I = the initial quantity of material or activity inhaled (units)

E = the concentration of material or activity observed in a urine sample collected on day t post intake (units/liter).

IRF = the incremental intake retention function, i.e., the fraction of the intake which would be expected to be excreted during day t post intake

1.4 = reference man excretion rate - ℓ/day.

In this case, E is set to 1 unit/liter, and the IRF is taken directly from Figure 6-15.

In a similar manner, Figure 6-23 was derived from Figure 6-16. This figure shows the estimated intake rate as a function of days of chronic intake. Although the Y axis of Figure 6-22 (and similar figures) is expressed as "μg of intake per μg/liter", noted that the curves are in fact unitless and could be expressed, for example, as "dpm of intake per dpm/ℓ".

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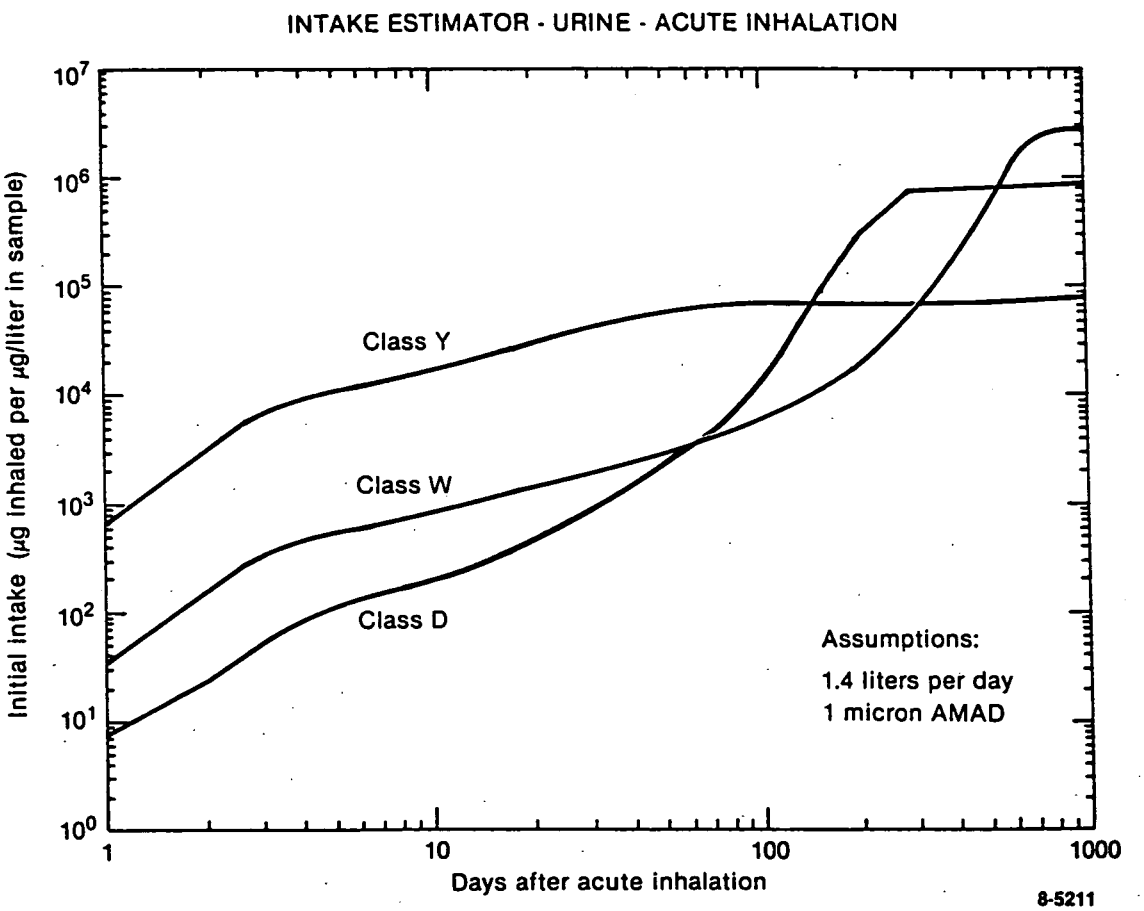

Figure 6-22. Uranium urine intake estimator - acute inhalation.

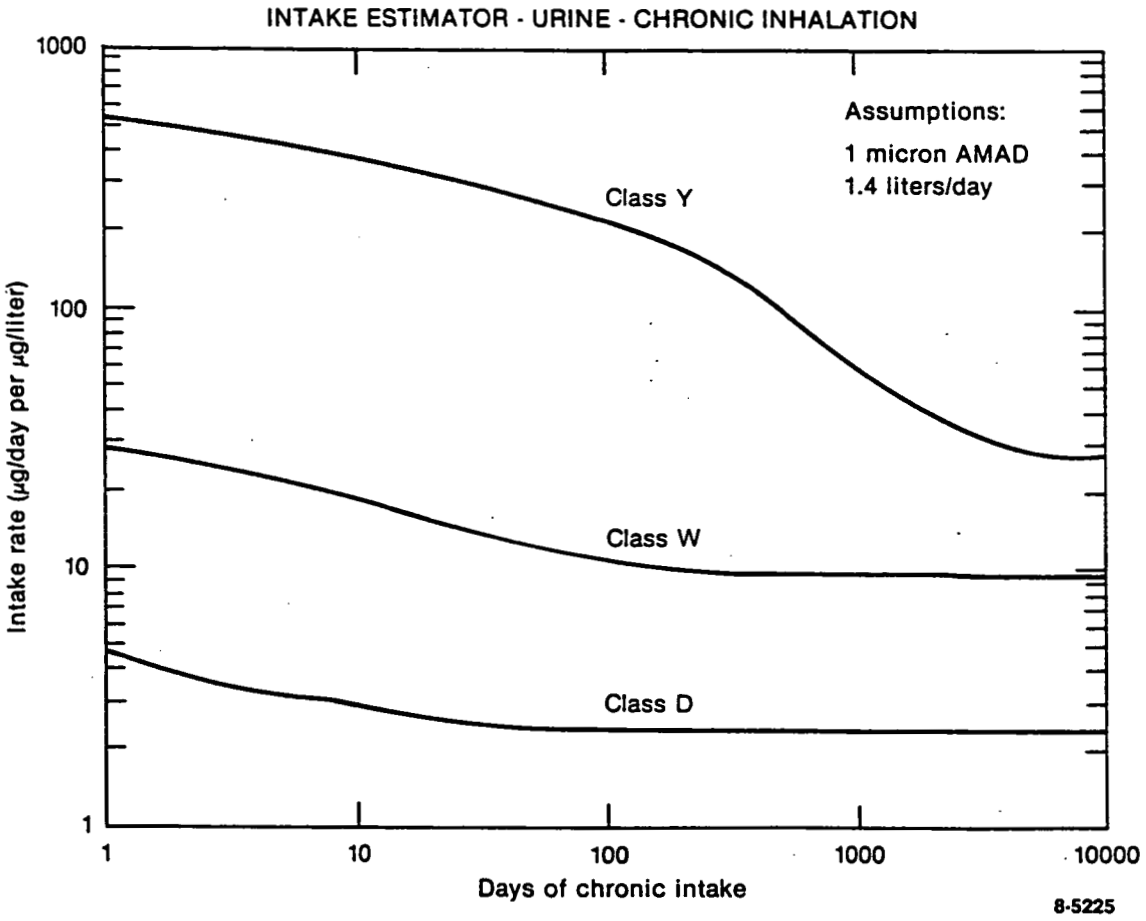

Figure 6-23. Urine intake estimator - chronic intake.

Examples of Use - Figure 6-23*

1. After 30 days of presumed chronic inhalation of Class Y uranium, a concentration of 1 μg per liter is observed in the urine of a male worker. What uniform intake rate would produce such a result?

   The curves on the graph are already normalized to the average daily urine excretion rate of 1.4 liters per day for males. At 30 days of chronic inhalation, the estimated normalized intake rate is about 300 μg per day per μg/liter. The implied intake rate is therefore 300 μg per day.

2. A "Friday afternoon" sample from a male worker shows a urine concentration of 25 μg/ℓ of presumed Class D uranium. (This is the worker's first week working in the plant.) What chronic daily inhalation rate would produce such a result?

   At 5 days of chronic inhalation, the estimated normalized intake rate is about 3.2 μg/day per μg/liter. The implied intake rate is 80 μg/day.

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* Recall that these curves include the rapid, direct excretion component and are only applicable to continuous intakes.

Chemical Toxicity

One can relate the quantity of uranium observed in the urine to the quantity of uranium presumed to be deposited in the kidney. This "kidney burden" may then be compared to the "nephrotoxic limit" of 3 micrograms of uranium per gram of kidney, and the "no effect" threshold burden of 1.1 microgram uranium per gram of kidney tissue.

Figure 6-24 shows the ratio of the mass of uranium in the kidneys to the mass concentration observed in the urine as a function of days post acute intake. The curves have been normalized to an assumed daily output rate of 1.4 liters per day. This figure has been derived from the curves of Figure 6-15 in a manner similar to that discussed for interpretation of urine samples. The curves, for all practical purposes, are identical from about day three to roughly 100 days post intake.

In some circumstances it may be useful to estimate the fraction of inhaled uranium which relatively quickly reaches the kidneys. This quantity may be estimated as outlined below:

Class D:

Path/Compartment

a	(0.30 * 0.50)	= 0.15	(0.01 days)
c	(0.08 * 0.95)	= 0.076	(0.01 days)
e	(0.25 * 0.80)	= 0.20	(0.50 days)
h → i	(0.25 * 0.20)	= 0.05	(0.50 → 0.50 days)
b → GI	(0.30 * 0.50 * 0.05)	= 0.0075	(0.01 → GI)
d → GI	(0.08 * 0.05 * 0.05)	= 0.0002	(0.20 → GI)
		0.4837	

The fraction of the initial intake reaching the kidney is thus:

[0.4837 * (0.12 + 0.00052)] = 0.05830

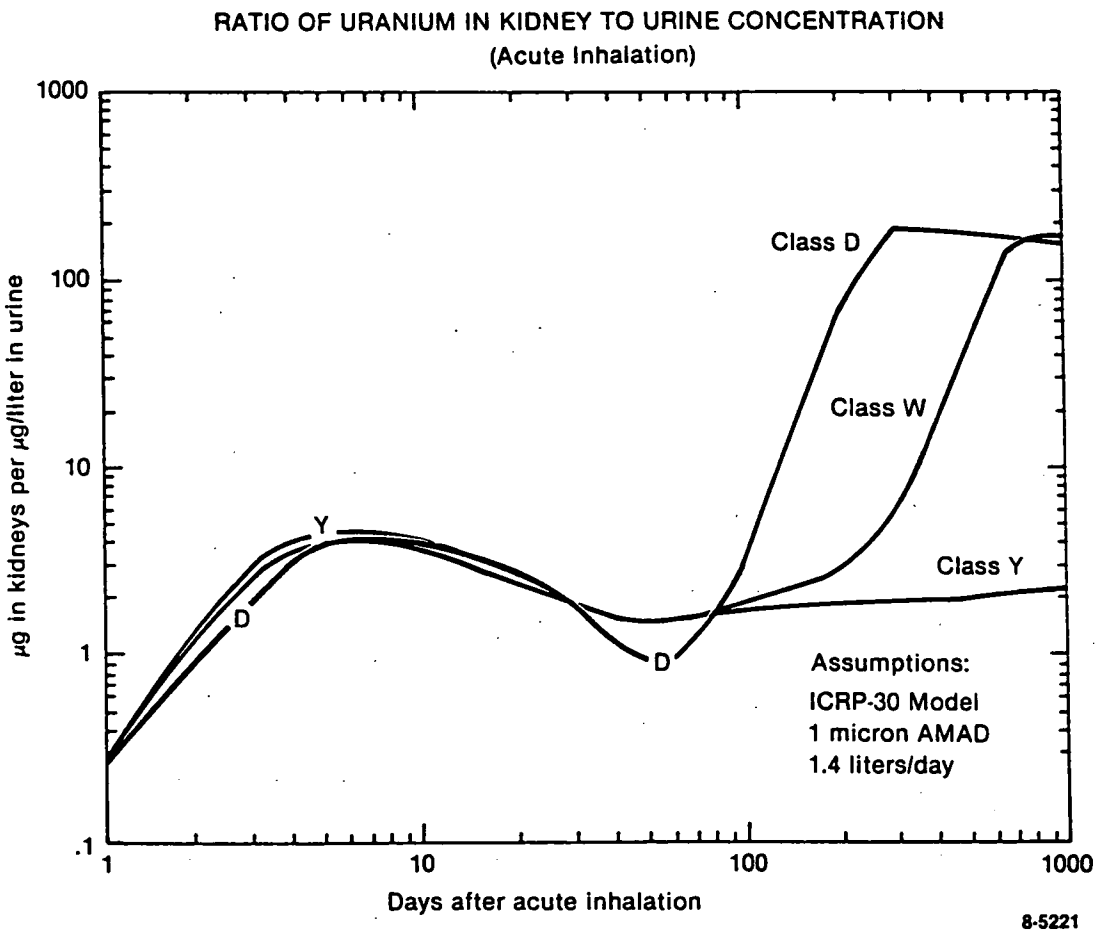

Figure 6-24. Ratio of uranium in kidney to urine concentration - acute inhalation.

Class W: (Only rapidly transferred activity has been considered. Activity from 50-day half-time compartments has been ignored.)

Path/Compartment

a	(0.30 * 0.1)	= 0.03	(0.01 days)
b → GI	(0.30 * 0.9 * 0.05)	= 0.0135	(0.04 days → GI)
c	(0.08 * 0.5)	= 0.04	(0.01 days)
d → GI	(0.08 * 0.5 * 0.05)	= 0.002	(0.2 → GI)
f → d → GI	(0.25 * 0.4 * 0.05)	= 0.005	(1 → 0.2 days → GI)
		0.0905	

The fraction of the initial intake quickly reaching the kidney is:

[0.0905 * (0.12 + 0.0052)] = 0.01091

Class Y:

Path/Compartment

a	(0.30 * 0.01)	= 0.003	(0.01 days)
b → GI	(0.30 * 0.99 * 0.002)	= 0.00059	(0.40 days → GI)
c	(0.08 * 0.01)	= 0.0008	(0.01 days)
d → GI	(0.08 * 0.99 * 0.002)	= 0.00016	(0.20 days → GI)
f → d → GI	(0.25 * 0.40 * 0.002)	= 0.0002	(1 → 0.2 days → GI)
		0.00475	

The fraction of the initial intake quickly reaching the kidney is:

[0.00475 * (0.12 + 0.00052)] = 0.00057

Figure 6-25 shows the ratio of uranium in the kidney to uranium in the urine for chronic intakes. The Y-axis shows the kidney burden in micrograms per microgram per liter observed in the urine as a function of days of chronic intake. These curves were derived from those of Figure 6-16.

Given the existing uncertainties in applying such models, the curves of ratios for chronic intakes are essentially identical.

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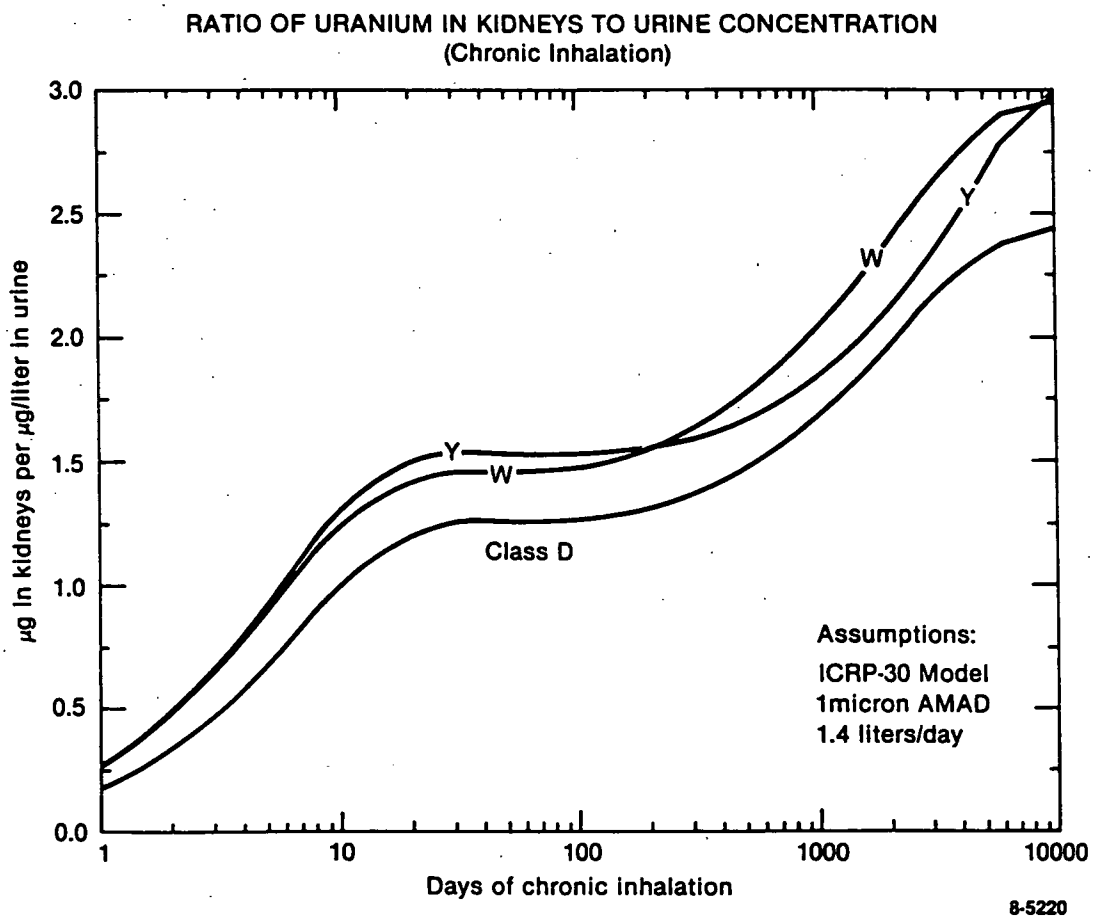

Figure 6-25. Ratio of uranium in kidney to urine concentration - chronic inhalation.

6.7.3 Fecal Sample Results

Routine fecal sampling for uranium is of limited use except for facilities which deal with highly enriched, relatively nontransportable uranium. However, in such facilities, such samples can provide valuable exposure control and dosimetric information.

Figures 6-26 and 6-27 have been derived from the incremental fecal excretion curves of Figures 6-9 and 6-10, respectively. Figure 6-26 shows the estimated intake as a function of daily fecal uranium excretion. No normalization has been performed. It is assumed that the quantity of uranium observed in the fecal sample is representative of the quantity excreted in a 24-hr period.

Figure 6-27 shows the estimated intake rate which would result in an excretion rate of 1 unit per day as a function of days of chronic intake. Predicted intake rates for Class W and Class Y uranium are essentially identical.

6.8 Dose Assessment

The first step of dose assessment is to estimate the quantity and nature of the intake. Once these factors have been estimated, organ dose equivalents and effective dose equivalents may be calculated for the time periods of interest.

Standard intake assumptions and metabolic models are appropriate for use in estimating intakes and doses for reference levels and planning purposes. However, for individual dose assessments, any individual or facility specific information available (e.g., observed clearance half-times, assumed particle size distribution) should always be incorporated into the models used for estimation of intake and subsequent dose. Models and parameters used in assessments should be thoroughly documented.

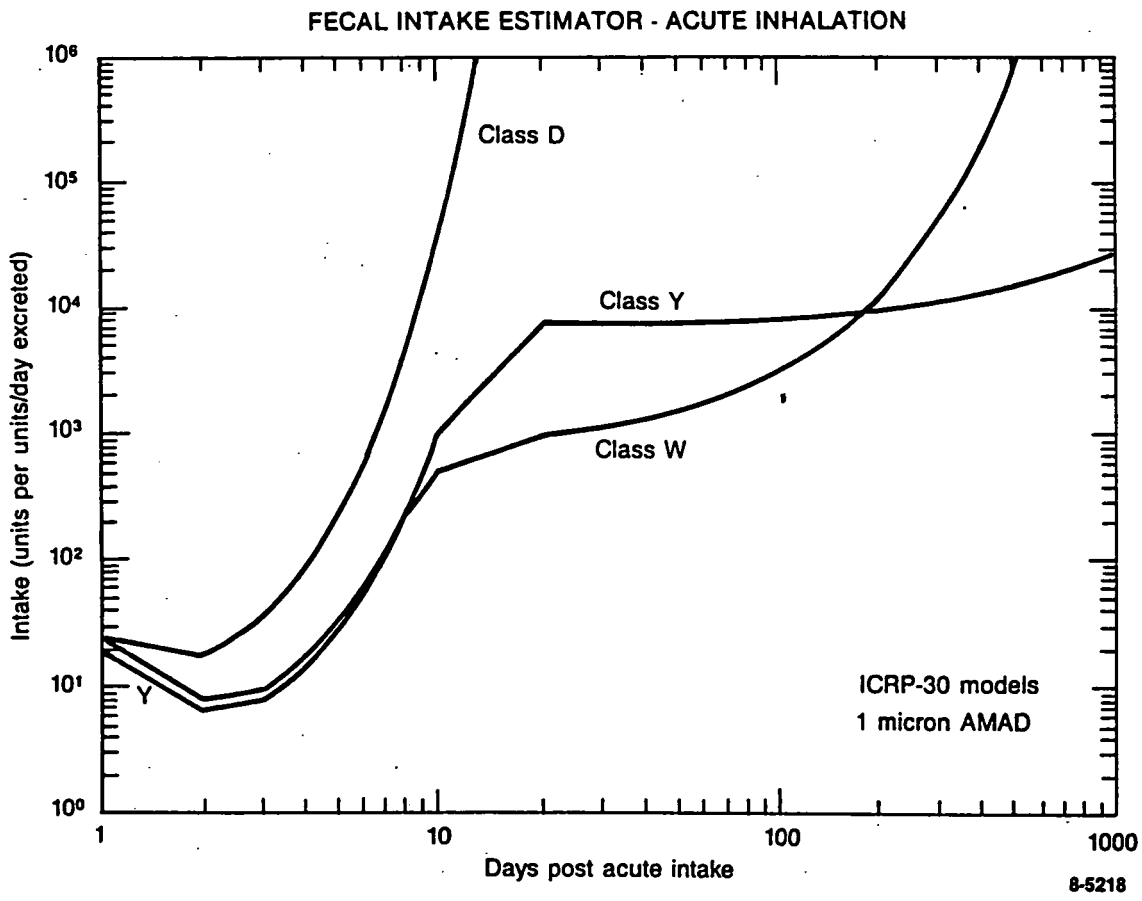

Figure 6-26. Fecal intake estimator - acute inhalation.

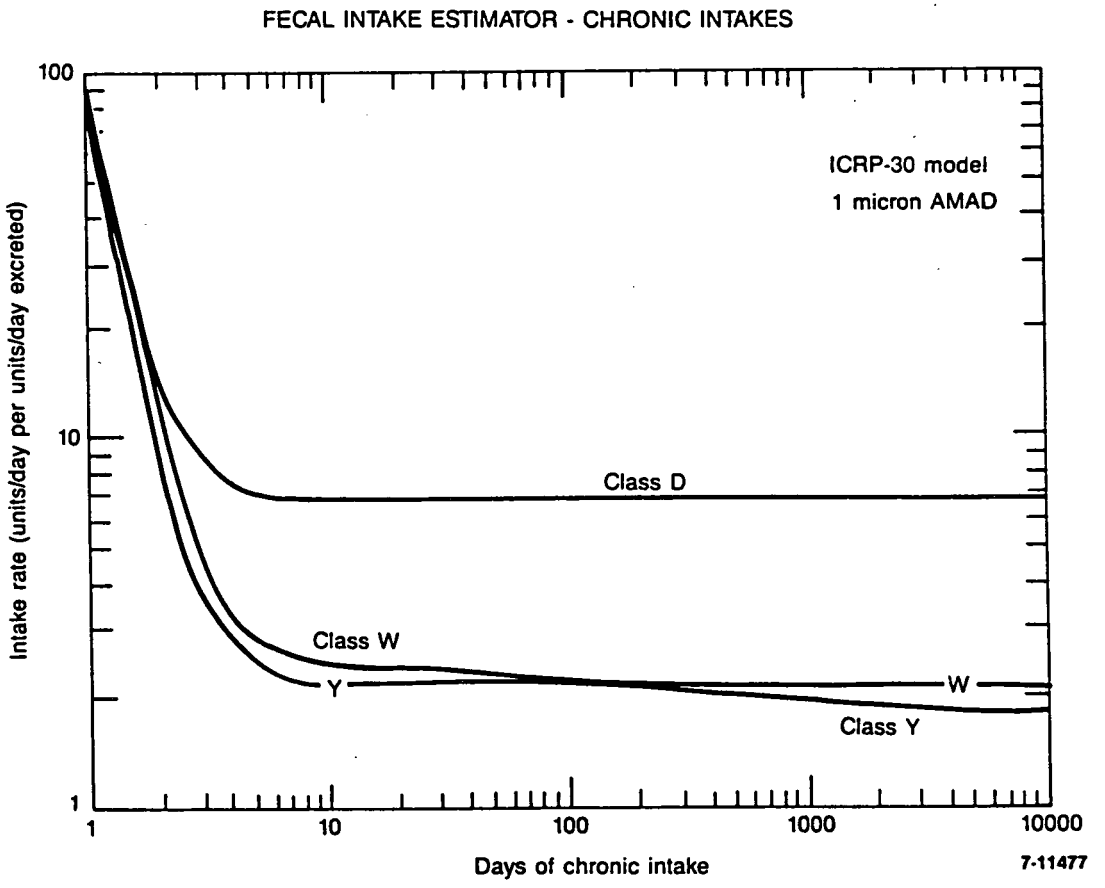

Figure 6-27. Fecal intake estimator - chronic intake.

It is recommended that the models and guidelines of ICRP-30 be used to assess doses from intakes of uranium. Dose conversion factors for first year and committed dose equivalents are presented in Table 6-3.

A variety of dose estimators can be established by applying such dose conversion factors to the intake estimators of Section 6.6. Figures 6-28 through 6-31 are presented as examples of such dose estimators. The dose conversion factors used in these figures were derived for natural uranium, where the activity is about 50% U-234 and about 50% U-238. Dose conversion factors for other mixes of uranium will be slightly, but not significantly, different.

TABLE 6-3. DOSE CONVERSION FACTORS FOR URANIUM ISOTOPES
(REM PER d/s INTAKE)

Radio-nuclide	First Year Lung	First Year Bone Surfaces	First Year Kidney	First Year Red Bone Marrow	First Year H(E,1)	H(E,2)	Committed Lung	Committed Bone Surfaces	Committed Kidney	Committed Red Bone Marrow	Committed H(E,50)
Class D											
U-234	3.1E-05	9.6E-05	2.5E-04	6.0E-06	2.2E-05	3.9E-06	3.2E-05	1.1E-03	4.5E-04	7.0E-05	7.2E-05
U-235	2.8E-05	8.8E-05	2.4E-04	5.5E-06	2.1E-05	3.6E-06	2.9E-05	1.0E-03	4.2E-04	6.6E-05	6.7E-05
U-238	2.7E-05	8.5E-05	2.3E-04	5.6E-06	2.0E-05	3.5E-06	2.8E-05	9.8E-04	4.0E-04	6.6E-05	6.5E-05
Class W											
U-234	1.6E-03	2.7E-05	7.4E-05	1.7E-06	2.0E-04	2.8E-06	1.6E-03	3.3E-04	1.4E-04	2.0E-05	2.1E-04
U-235	1.5E-03	2.5E-05	7.0E-05	1.6E-06	1.8E-04	2.6E-06	1.5E-03	3.0E-04	1.3E-04	1.9E-05	2.0E-04
U-238	1.4E-03	2.4E-05	6.8E-05	1.6E-06	1.7E-04	2.5E-06	1.4E-03	2.9E-04	1.2E-04	1.9E-05	1.9E-04
Class Y											
U-234	6.2E-03	1.9E-06	5.7E-06	1.1E-07	7.4E-04	5.1E-04	3.0E-02	1.1E-04	4.8E-05	7.0E-06	3.6E-03
U-235	5.7E-03	1.7E-06	5.3E-06	1.1E-07	6.8E-04	4.7E-04	2.8E-02	1.0E-04	4.5E-05	6.5E-06	3.4E-03
U-238	5.4E-03	1.6E-06	5.2E-06	1.1E-07	6.5E-04	4.5E-04	2.7E-02	1.0E-04	4.3E-05	6.6E-06	3.2E-03
Injection											
U-234	n/a	2.0E-04	5.2E-04	1.2E-05	7.0E-05	8.0E-06	n/a	2.3E-03	9.3E-04	1.4E-04	2.0E-04
U-235	n/a	1.8E-04	4.9E-04	1.1E-05	6.5E-05	7.4E-06	n/a	2.1E-03	8.7E-04	1.3E-04	1.8E-04
U-238	n/a	1.8E-04	4.7E-04	1.2E-05	6.4E-05	7.2E-06	n/a	2.0E-03	8.5E-04	1.3E-04	1.8E-04

Notes:

1. Dose conversion factors for Class D, Class W, and Class Y are for acute inhalation.
2. H(E,1) = Effective Dose Equivalent received during first 365 days after intake.
3. H(E,2) = Effective Dose Equivalent received during second 365 days after intake.

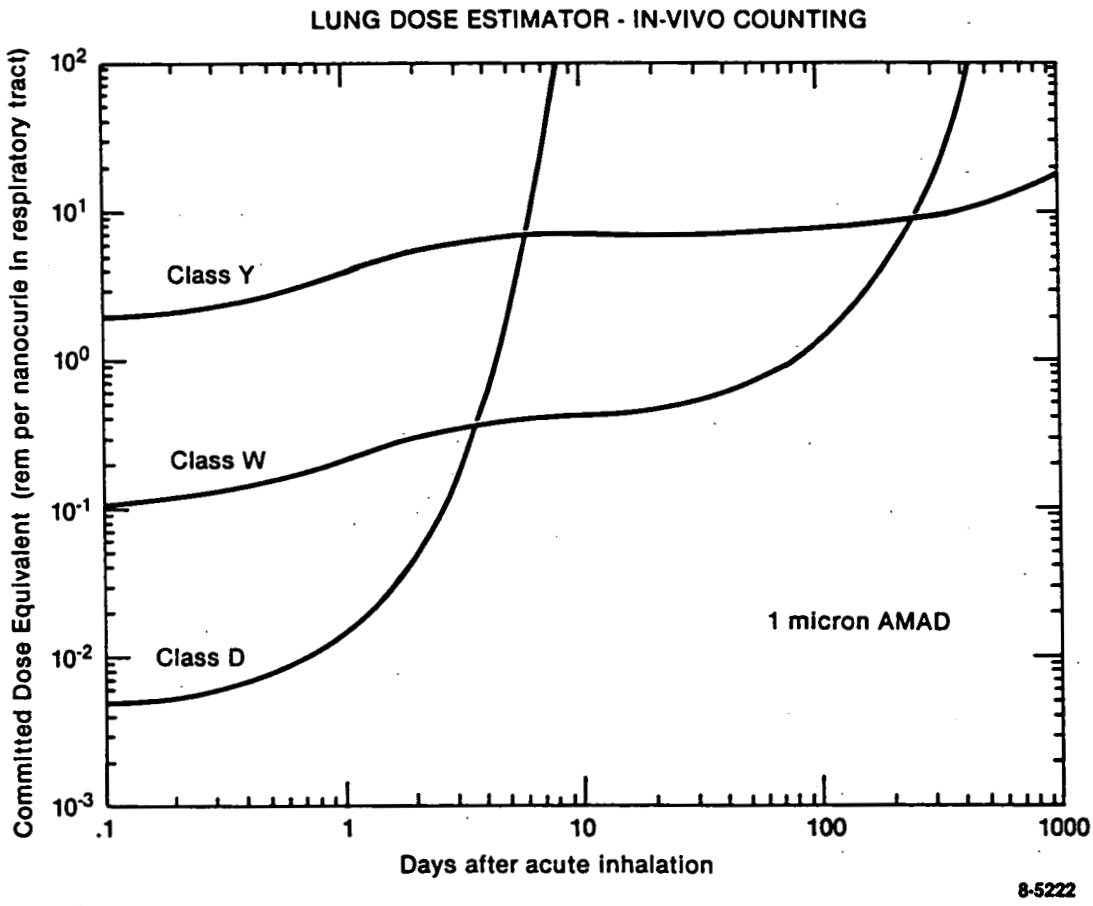

Figure 6-28. Uranium in-vivo lung dose estimator.

URANIUM IN-VIVO LUNG DOSE ESTIMATOR - Example of Use

1. A facility is currently running quarterly lung counts on a group of employees who work with Class W uranium. (The AMAD is assumed to be 1 micron.) Management has requested that this schedule be changed to annual lung counts. What loss of sensitivity (ratio of "missed doses") would be expected from this change? Assume that your in-vivo detection limit is about 1 nci of total deposited uranium activity.

   At 90 days, the "missed dose" per nci is about 1.2 rem (committed dose equivalent). At 365 days, the "missed dose" would be about 40 rem. Thus changing from quarterly to annual counts for this material would result in a loss of sensitivity (as measured by "missed dose") of a factor of 35.

2. A second group of employees (working with Class Y uranium) is also being considered for the same schedule change, i.e., quarterly to annual. What will be the impact of this change?

   At 90 days, the "missed dose" for 1 micron Class Y uranium is about 8 rem. At 365 days, the "missed dose" is about 10 rem. Thus, the difference in sensitivity is minimal.

3. A worker has been exposed to what is assumed to be 1 micron Class Y uranium dust. A lung count performed about 24 hours after the incident shows about 5 nci of total uranium activity deposited in the lungs. What is the estimated committed dose equivalent to the lungs from this intake?

   At 1 day post acute intake, the dose per nanocurie observed in the total respiratory tract (including the NP and TB regions) is about 4 rem. Thus, the observed 5 nci implies a committed lung dose on the order of 20 rem.

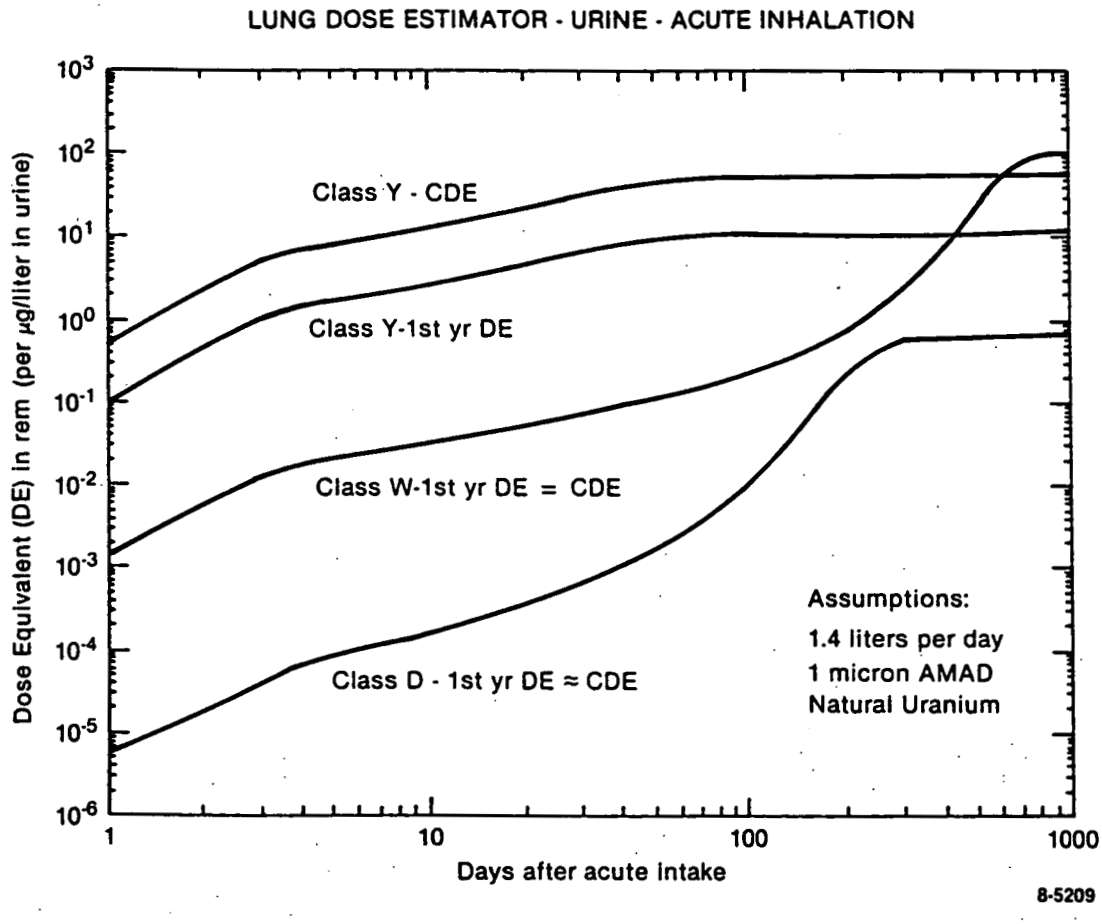

Figure 6-29. Estimation of lung dose from urine sample results - natural uranium.

URANIUM LUNG DOSE ESTIMATOR - Example of Use

1. Monthly "spot" urine samples are collected from people who work with Class Y (1 micron) natural uranium. Your detection limit for uranium in urine is about 0.8 micrograms per liter. What is the corresponding dose detection limit for this group of people? In other words, what lung dose could just be missed?

   Assume that an individual collects the sample 30 days after an acute intake. The implied lung dose (committed dose equivalent) per microgram of natural uranium in urine is about 30 rem. An analytical detection limit of 0.8 μg/ℓ lowers this dose to about 24 rem.

2. 5 micrograms of uranium per liter is detected in a urine sample collected 2 days after a suspected exposure to 1 micron, Class W or Class Y, 5% AVLIS enriched uranium. What range of committed lung doses are possible?

   For natural uranium (at 2 days post acute intake), the corresponding committed dose equivalent (per μg/ℓ in the urine) for Class W and Class Y uranium are about 5×10^{-3} and 2 rem, respectively. Thus, for the observed 5 μg/ℓ, the corresponding doses are 2.5×10^{-2} and 10 rem, respectively. Figure 2.4, Section 2, shows that the difference in specific activity between natural and 5% AVLIS enriched uranium is minimal.

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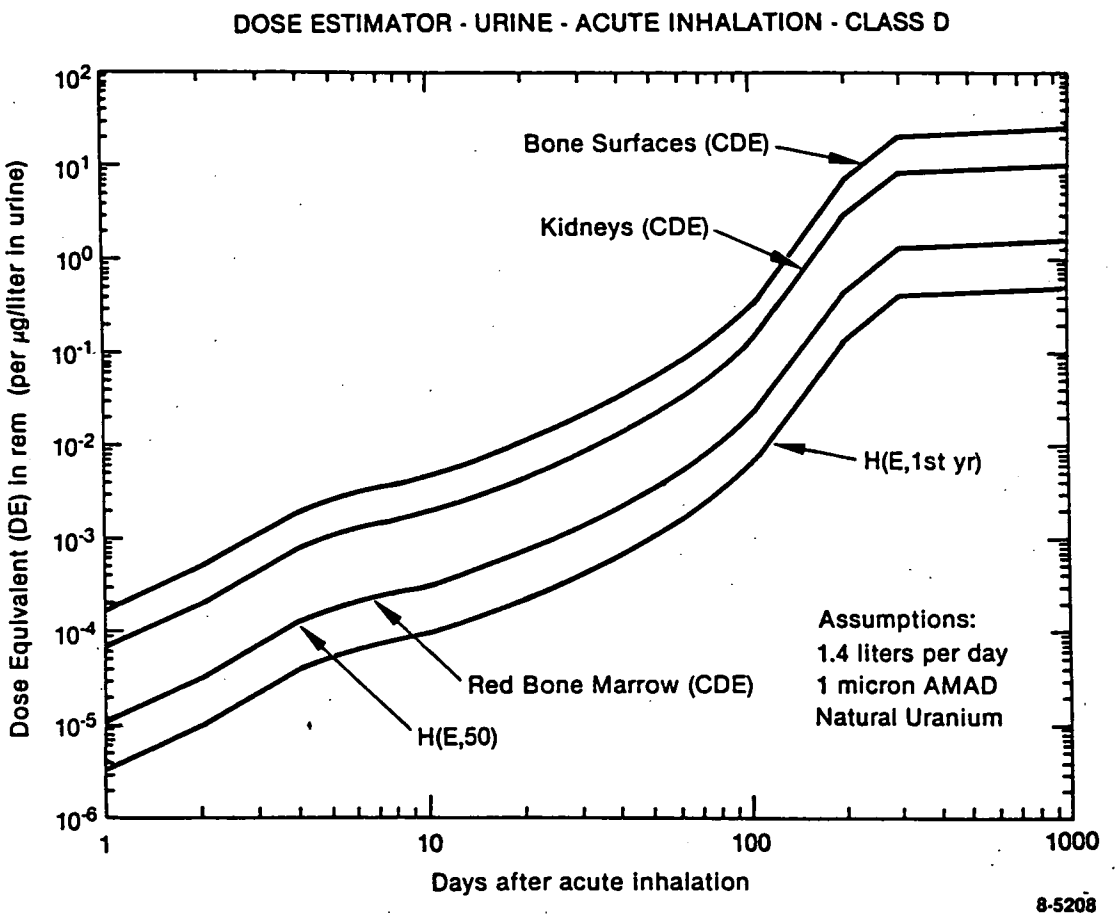

Figure 6-30. Estimation of first year and committed doses from urine sample results - Class D natural uranium.

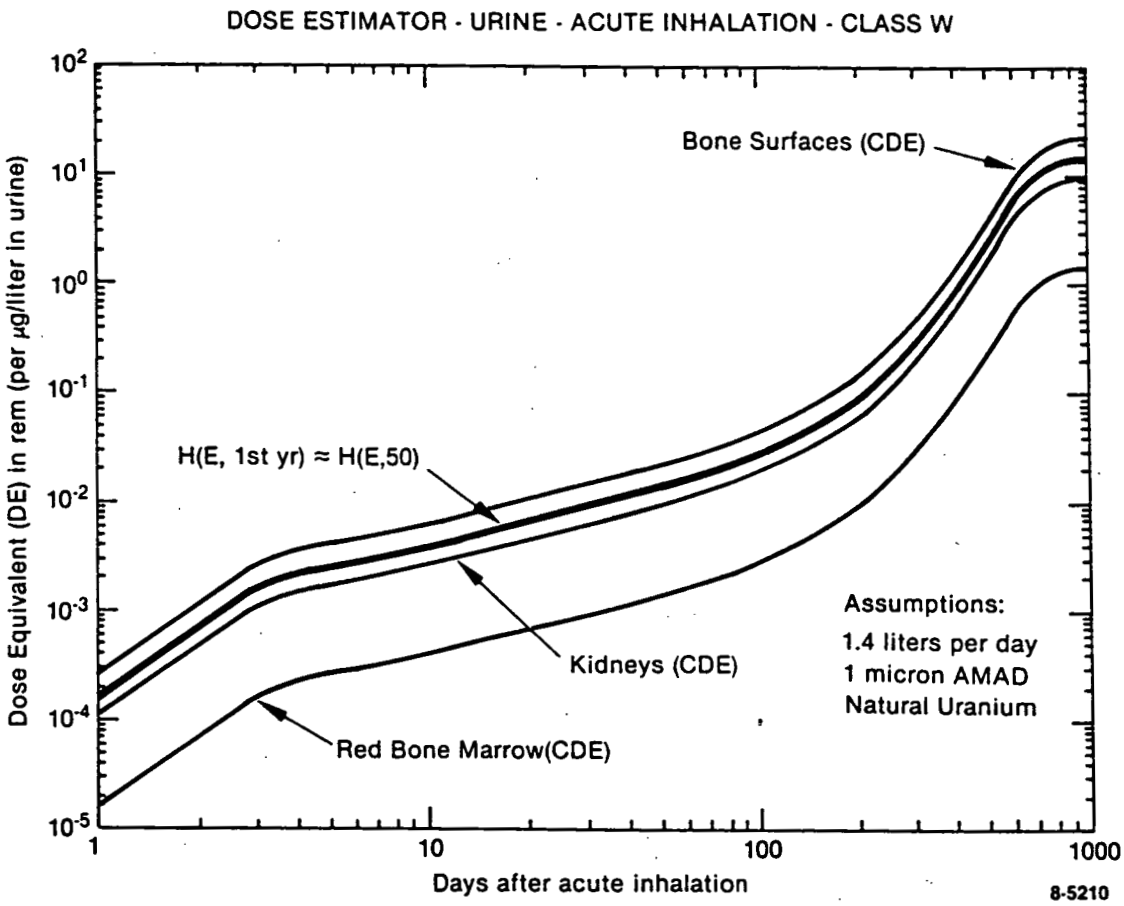

Figure 6-31. Estimation of first year and committed doses from urine sample results - Class W natural uranium..

PRELIMINARY URANIUM DOSE ESTIMATOR - Example of Use

1. Weekly "spot" urine samples are collected from people who work with Class D (1 micron) natural uranium. Your detection limit for uranium in urine is about 1.0 micrograms per liter. What is the corresponding dose detection limit for this group of people. In other words, what committed effective dose equivalent (CEDE) could just be missed?

   Assume that an individual collects the sample 7 days after an acute inhalation. The implied CEDE per microgram of natural uranium in the urine is about 3.5×10^{-4} rem.

2. 5 micrograms of uranium per liter is detected in a urine sample collected 2 days after a suspected exposure to 1 micron, Class W, 93% conventionally enriched uranium. What bone surface and kidney doses can be expected?

   Referring to Figure 6-31, for natural uranium, (at 2 days post acute inhalation) the corresponding committed organ dose equivalent (per μg/ℓ) for the bone surfaces and the kidneys are about 1×10^{-3} and 5×10^{-4} rem, respectively. Figure 2.1, Section 2, shows that the specific activity of 93% conventionally enriched uranium is about 100 times that of natural uranium. Thus, for the 5 μg/ℓ sample, the implied doses for the bone surfaces and kidneys are 0.5 and 0.25 rem, respectively.

6.9 Reference and Action Levels

Actions based on bioassay results should be appropriate and uniformly applied. These actions may range from merely requesting a re-sample to confirm a positive result, to an extensive follow-up sampling schedule, investigation and dose assessment.

For each applicable clearance class of uranium, range of particle size, anticipated mode of intake (usually inhalation), and bioassay procedure the following general categories of reference levels should be established:

1. Minimum Detection Level

   Activity (and corresponding intake, deposition, or dose equivalent) which can reliably be expected to be detected by the system of measurement.

2. Investigation Level

   Activity (and corresponding dose, etc.) at which some sort of follow-up actions become warranted. Such follow-up actions would typically involve re-sampling, information gathering, i.e., requests for follow-up and/or other types of bioassays, investigations of working conditions, air monitoring results, etc.

3. Action Level

   Activity (and corresponding dose, etc.) at which significant action is warranted based on the need to limit further exposures to the worker and/or co-workers, or to eliminate the likelihood of further intakes to assist in diagnostic efforts.

A recommended scheme which encompasses the above considerations is presented in NCRP Report No. 87, and is (in essence) presented in Table 6-4. A similar scheme is presented in WASH-1251. This scheme suggests actions to be taken depending upon the fraction of the "basic protection criterion" (i.e., dose limit) implied by the bioassay result. It is suggested that the Annual Limit of Intake (ALI) be used for the "basic protection criterion" for ingestion and inhalation intakes.

Different groups of actions are recommended depending upon the type (baseline, prospective, retrospective) of bioassay sample yielding the results. For baseline (on-hire) bioassays, it would be prudent to investigate the cause of any statistically positive result which is higher than background levels. Normally, a review of the worker's previous exposure history and any previous bioassay results will be sufficient.

Guidelines for setting reference/action levels for prospective bioassays are somewhat more complicated, and depend to a certain extent on the nature of the facility. Newer facilities should be designed so as to prevent any significant intakes of uranium. In such a facility, any positive result (above background) may be cause for investigation. In contrast, some operations in older facilities routinely result in small intakes. In this case, one has to distinguish between this expected "normal" background, and a result which indicates an unusually high or unexpected intake.

6.10 Response to Suspected Intakes

Incidents involving uranium are not usually serious or life-threatening, with the exception of criticality accidents and accidents involving the release of UF_6. Although the latter case does involve the intake of activity, it is the chemical toxicity of the UF_6 and reaction products that are the primary concern.

TABLE 6-4. EXAMPLES OF CORRECTIVE ACTIONS BASED ON BIOASSAY MEASUREMENT RESULTS[a]

Fraction of Protection Criterion	Interpretation	Actions
Below 1/10 to 1/4	Confinement and air sampling capabilities confirmed	It is good practice to investigate a positive bioassay result when none were being observed previously.
Between 1/10 to 1/4 and 1/2	Confinement and/or air sampling capabilities marginal.[a]	1. Confirm result (repeat measurement). 2. Identify cause and initiate additional control measures. 3. Determine whether other workers could have been exposed and perform bioassay measurements for them. 4. If exposure (indicated by excreta analysis) could have been to Class (Y) material that can be measured using in-vivo techniques, consider such measurements.[b]
Between 1/2 and 1	Confinement and/or air sampling capabilities unreliable.[a]	1. Take the actions listed above. 2. If exposure (indicated by excreta analysis) could have been to Class (Y) material that can be measured using in-vivo techniques, assure that such measurements are performed.[b] 3. Determine why air samples were not representative and did not warn of excessive airborne radioactivity. Make necessary corrections. 4. Perform additional bioassay measurements as necessary to make a preliminary estimate of the critical organ burden; consider work assignment limitations that will permit natural elimination and/or radioactive decay and assure that the protection criterion is not exceeded.

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TABLE 6-4. (Continued)

Fraction of Protection Criterion	Interpretation	Actions
Between 1/2 and 1 (continued)		5. If exposure could have been to Class (Y) material, continue operations only if it is virtually certain that the protection criterion will not be exceeded by any worker.[b]
Greater than 1	Containment and/or air sampling capabilities unacceptable.[a]	1. Take all actions listed above. 2. Continue operations only if it is virtually certain that the protection criterion will not be exceeded by any other worker, regardless of radionuclide classification. 3. Establish work assignment limitations as necessary for affected workers. 4. Perform individual metabolic studies for affected workers.

a. Table reproduced directly from NCRP Report No. 87, page 48.

b. Unless the result was anticipated and caused by conditions already corrected.

c. Material classification per ICRP Publication 30 (ICRP, 1979).

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Generally, appropriate response to an accident includes the following actions, in order of importance.

a. Render first aid to victims as necessary
b. Stabilize the situation
c. Address radiological concerns
d. Take action to mitigate further consequences
e. Notify and document.

This section will address specific responses to incidents where inhalation or injection/absorption of uranium is known or suspected. Although some of the responses are similar, each route of intake will be treated separately.

6.10.1 Planning

The first step in incident response is to prepare for incidents before they occur. Part of the characterization of hazards discussed earlier in this manual should be a characterization of the types of accidental intakes which may occur. Response guidelines, based on the chemical forms and particle size distributions which might be expected, should be prepared and available to the emergency response staff. Such guidelines should clearly present step-by-step actions to be taken.

6.10.2 Estimate of Intake

The health physicist's primary concern is to make an estimate of the magnitude of the intake or deposition. Some or all of the following early indicators should be used.

a. Observed contamination levels on:

- the body
- the head and face
- around the mouth and nose
- on working surfaces
- on equipment being used.

b. Air activity information:

- CAM alarms
- CAM strip charts
- air sampler filter analyses.

6.10.3 Nasal Contamination

Nasal swab results (if carefully and quickly obtained) can provide an estimate of the magnitude of intake in the event of accidental intakes. Nasal swabs should be collected in all incidents where inhalation of significant quantities of uranium is suspected. The following procedure is recommended:

1. Swabs should be collected as quickly as possible after an incident. Typically, moistened cotton swabs or filter paper wrapped around swabs are used. Collection procedures described in NCRP-65 should be used.

2. Swabs must be collected before the individual showers, washes their face or blows their nose.

3. The swab should be collected by the individual (under direct guidance of the health physicist or technician) if possible. Great care must be exercised to avoid contaminating the swab with the hands or on the lip or face of the individual.

4. Individual swabs should be taken from each nostril. Swabs should be individually bagged (or otherwise isolated) and labeled with the individual's name, collection date and time, and right or left nostril.

5. Swabs can be counted immediately for beta activity, but must be thoroughly dried before being counted for alpha activity unless liquid scintillation counting is used.

6. Geometry factors for swab counting should be established for use during incidents. Geometry factors of about 0.25 have been observed for laboratory counters in the detection of uranium alphas on whole cotton swabs.

7. Swabs should always be saved for re-counting or radiochemical analysis if necessary.

Interpretation of Nasal Swab Results

Nasal swab results may be used to estimate the magnitude of activity initially deposited (or inhaled) by relating the amount of activity collected on the swabs to the activity assumed to be deposited in the NP region of the respiratory tract. This activity may then be related to the total quantity of activity deposited or inhaled.

The fraction of NP activity collected on nasal swabs (or on nose blows) has been observed to range from 8 to 15%. Obviously, this fraction will be highly dependent upon a number of factors such as collection technique and time of collection (after intake, etc.). A value of 10% would seem reasonable and somewhat conservative.

The ratio of activity initially deposited in the NP region to total intake (or to activity deposited in the pulmonary region) is, of course, a function of particle size. At 1 micron AMAD, the ratio of NP deposition to intake is 0.3. Combining this fraction with an assumed 10% collection

efficiency gives an estimated intake that is about 30 times the activity observed on the sum of both swabs. Figure 6-32 may be used to estimate the magnitude of an intake as a function of particle size distribution. The peak which appears at 0.2 microns is an artifact of the "turn-around" in deposition fractions assumed by the deposition model.

Although nasal swab results can be useful in estimating the magnitude of intake, they can also be misleading in certain circumstances. Results of nose swabs are very sensitive to many factors. Intakes of small particle size material may not yield positive nasal swab results and insignificant intakes of large particle size material may mimic serious depositions.

A comparison of the right and left nostril results can provide some information regarding the "validity" of the intake estimate. Although normal variations in the nasal orifices will produce some differences, vastly different right and left nostril results strongly suggest external contamination, rather than inhalation.

Finally, a nasal swab results can be meaningless if the individual is a "mouth-breather", or if the person had a cold or sinus problems.

6.10.4 Contaminated Wounds

Contaminated wounds are not uncommon in facilities where workers handle large quantities of uranium. Such wounds are rarely serious (from a radiological standpoint) due to the low specific activity of uranium.

Figure 6-33 shows that for a single compartment wound, the 50-year committed dose equivalent to the bone surfaces is on the order of 85 mrem per nanocurie initially deposited in the wound. The clearance half-time from the wound does not significantly affect the committed dose for wound clearance half-times up to several years.

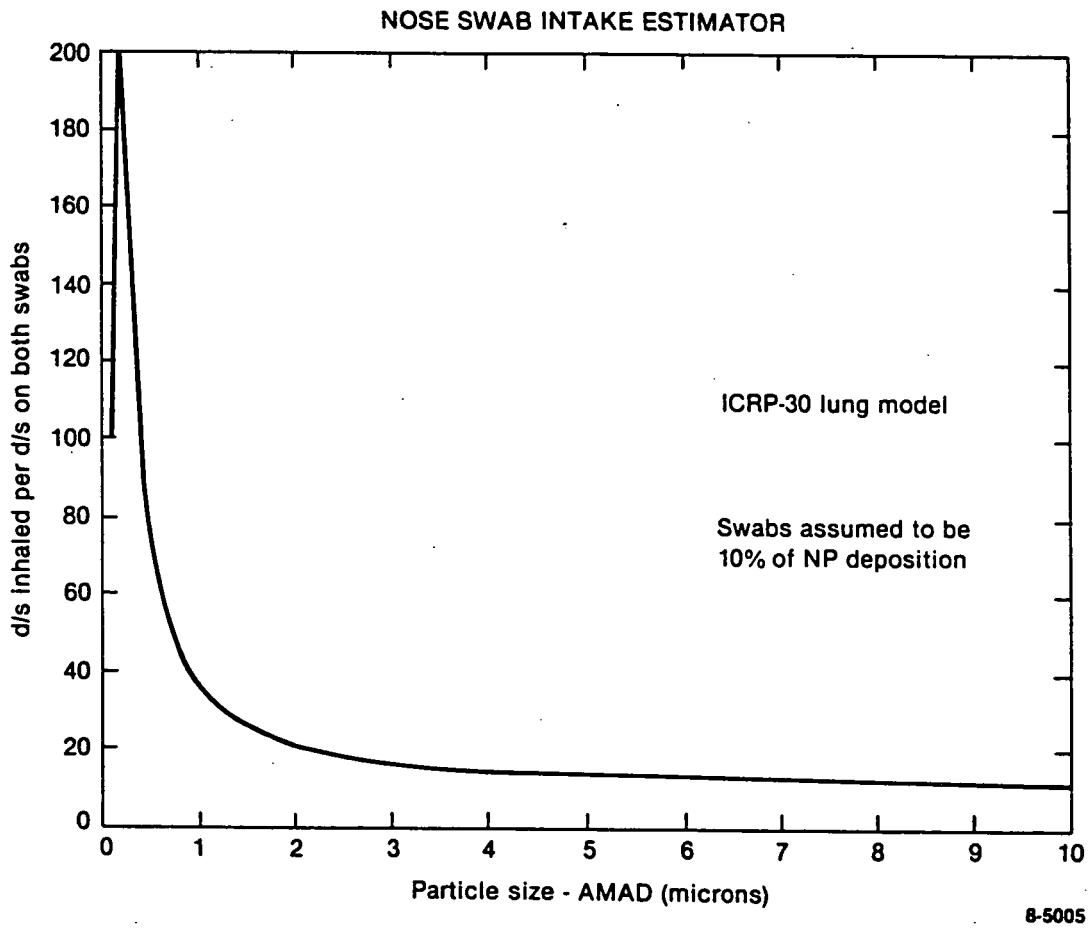

Figure 6-32. Nose swab intake estimator.

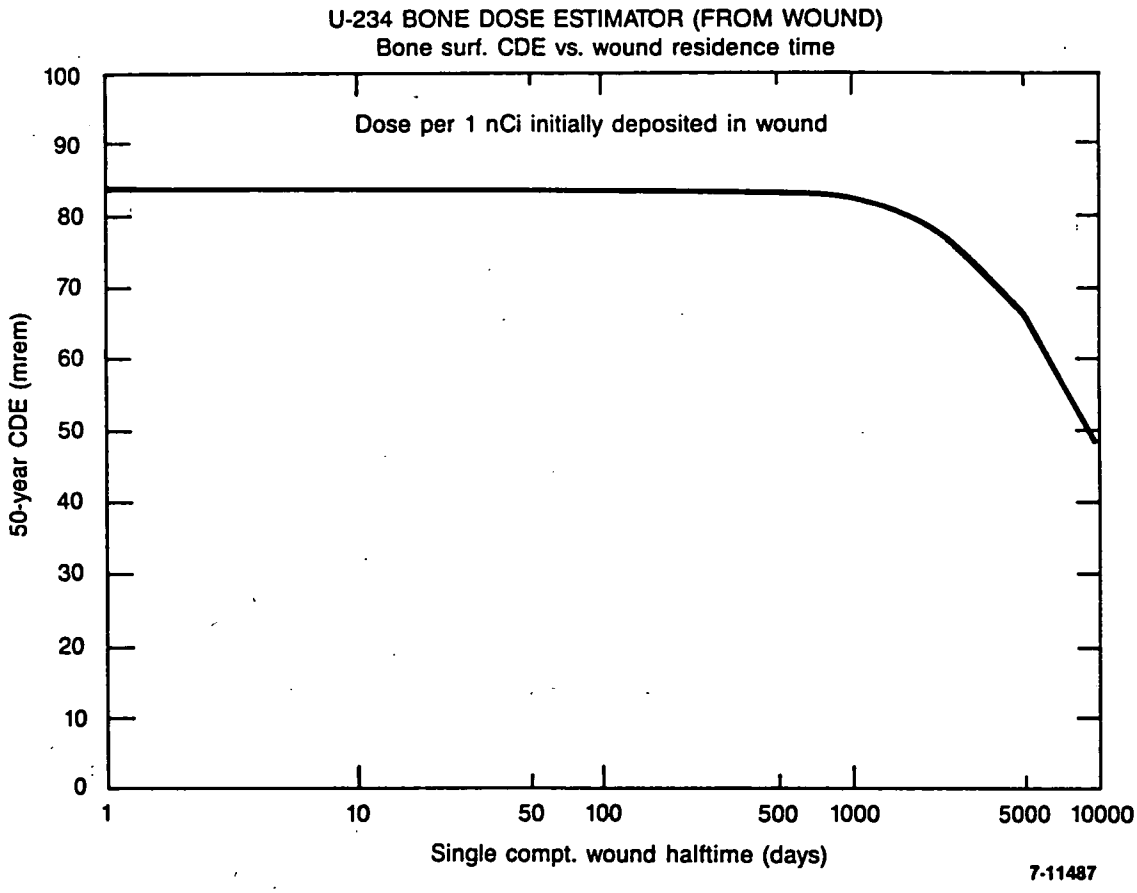

Figure 6-33. U-234 bone dose estimator from wound.

After appropriate first aid is administered, contaminated or potentially contaminated wounds should be cleaned as much as possible. Facilities in which contaminated wounds are likely should have a wound counter available to assess the quantity of uranium deposited in a wound.

If a significant quantity of activity is observed in the wound, it would be prudent to obtain urine samples and to "track" the activity in the wound with subsequent counts. This will allow the health physicist to accurately assess the quantity and pattern of absorption of uranium from the wound into the bloodstream.

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SECTION 7

EXTERNAL DOSE CONTROL

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SECTION 7

EXTERNAL DOSE CONTROL

The primary purpose of an external dose equivalent (DE) control program is to protect the individual radiation worker by minimizing external dose equivalent and preventing exposure above the limits specified in DOE Order 5480.11 (see Table 7-1). A secondary but important purpose is to minimize the total plant collective DE as determined by summing all the individual personnel doses. These purposes are accomplished by providing means to detect, evaluate, and control radiation fields in the work place and the exposure of the worker to these fields. The elements of the external DE control program are: detection and characterization of the beta, gamma, and neutron radiation fields, measurement/quantification of these fields, measurement of personnel DE, and determination of DE control practices.

7.1 Dose Equivalent Limits

7.1.1 DOE Limits

DOE DE limits of interest in control of external DE from uranium are those for the whole body, lens of the eye, "unlimited areas of the skin," forearms, and hands and feet. DOE Order 5480.11 as well as the International Commission on Radiological Protection (ICRP) and National Council on Radiation Protection and Measurement (NCRP) standards specifies the following depths in tissue at which these doses are to be measured:

	mg/cm^2
Deep (penetrating)	1000
Lens of the eye	300
Shallow (skin)	5-10 (ICRP)
	7 (NCRP and DOE Order)

TABLE 7-1. RADIATION DOSE LIMITS

Exposure Type	
Occupational	
Stochastic Effects	5 rem (annual effective dose equivalent)
Non-Stochastic Effects	
Lens of Eye	15 rem (dose equivalent)
Extremity	50 rem (dose equivalent)
Skin of the Whole Body	50 rem (dose equivalent)
Organ or Tissue	50 rem (dose equivalent)
Unborn Child of a Worker	
Gestation Period	0.5 rem (annual effective dose equivalent)
Planned Special Exposure	
Event Plus Annual Occupational Exposure	10 rem (annual effective dose equivalent)
Minors	One-tenth of occupational radiation protection standards
Student	0.1 rem (annual effective dose equivalent)

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7.1.2 Beta Doses

Beta doses to the skin, extremities, and (sometimes) the lens of the eye can be limiting in facilities which process unshielded depleted, natural, or low-enrichment uranium. Processes which separate and sometimes concentrate beta-emitting uranium daughters are not uncommon in DOE uranium facilities (see Section 7.2). Surface beta dose rates on the order of 1 to 20 rads per hour have been observed in such circumstances. Control of exposure is complicated by the fact that considerable contact work takes place in facilities which process uranium metal.

7.1.3 Gamma Doses

Although beta radiation fields from unshielded uranium tend to present the most intense radiation problem, storage of large quantities of uranium can create widespread, low-level (< 5 mrem/hr) gamma radiation fields. Such fields can create ALARA problems--particularly when significant numbers of people must work in adjacent areas.

7.1.4 Neutron Doses

In uranium processes which create fluoride compounds (UF_4, UF_6, etc.) the α-n reaction with this light nuclide can result in neutron radiation fields, which are a function of the compound, mixing, storage configuration, and enrichment. As indicated in Section 2 (2.1.1), low (< 5%) enriched UF_6 in large storage containers can result in neutron radiation in the 0.2 mrem/hr range while high enriched (> 97%) UF_6 can read in the 4 mrem/hr range. At high enrichments the neutron fields can be up to a factor of 2 higher than the gamma fields and be the limiting source of whole body exposure. Neutron radiation from U metals and low enriched compounds is considerably lower than the gamma component and consequently not limiting.

7.2 Radiation in Uranium Facilities

7.2.1 Characterization

The design of an external DE control program, including instrument and dosimeter selection, is dependent upon the type and intensity of the radiation fields to which the workers will be exposed. Many factors can affect the radiation field such as:

1. Enrichment (mix of uranium isotopes)
2. Emissions from parent radionuclide(s)
3. Emissions from daughter radionuclide(s)
4. Emissions from impurity radionuclide(s)
5. Type of radiation emitted (beta, gamma, etc.)
6. Energies of emitted radiation
7. Specific activity of the source material
8. Self-shielding of source material
9. Shielding provided by process equipment
10. Shielding provided by protective clothing
11. Distance and geometry factors.

7.2.2 Radiation From Isotopes Other Than Uranium

Radiation fields from uranium are frequently dominated by contributions from daughter product or impurity radionuclides. For example, essentially all of the beta radiation field from depleted uranium comes from the daughter radionuclide Pa-234m, and to a lesser extent from Th-234. During melting and casting operations, these daughter elements may concentrate on the surface of the castings and equipment, producing beta radiation fields up to 20 rad per hour (see Figures 7-1, 7-2, and 7-3).

7.2.3 Radiation Types and Energies

The ratio of uranium isotopes in a specific process (a function of enrichment) will determine the source term by which the radiated fields can

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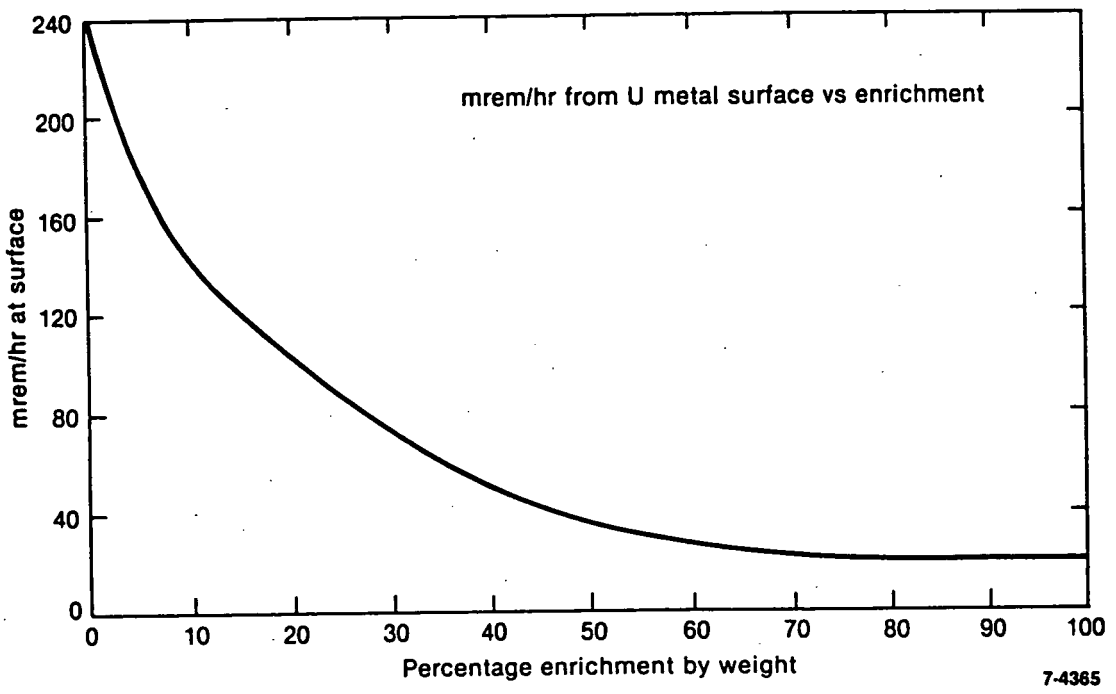

Figure 7-1. Radiation readings at surface of uranium metal vs percentage enrichment by weight.

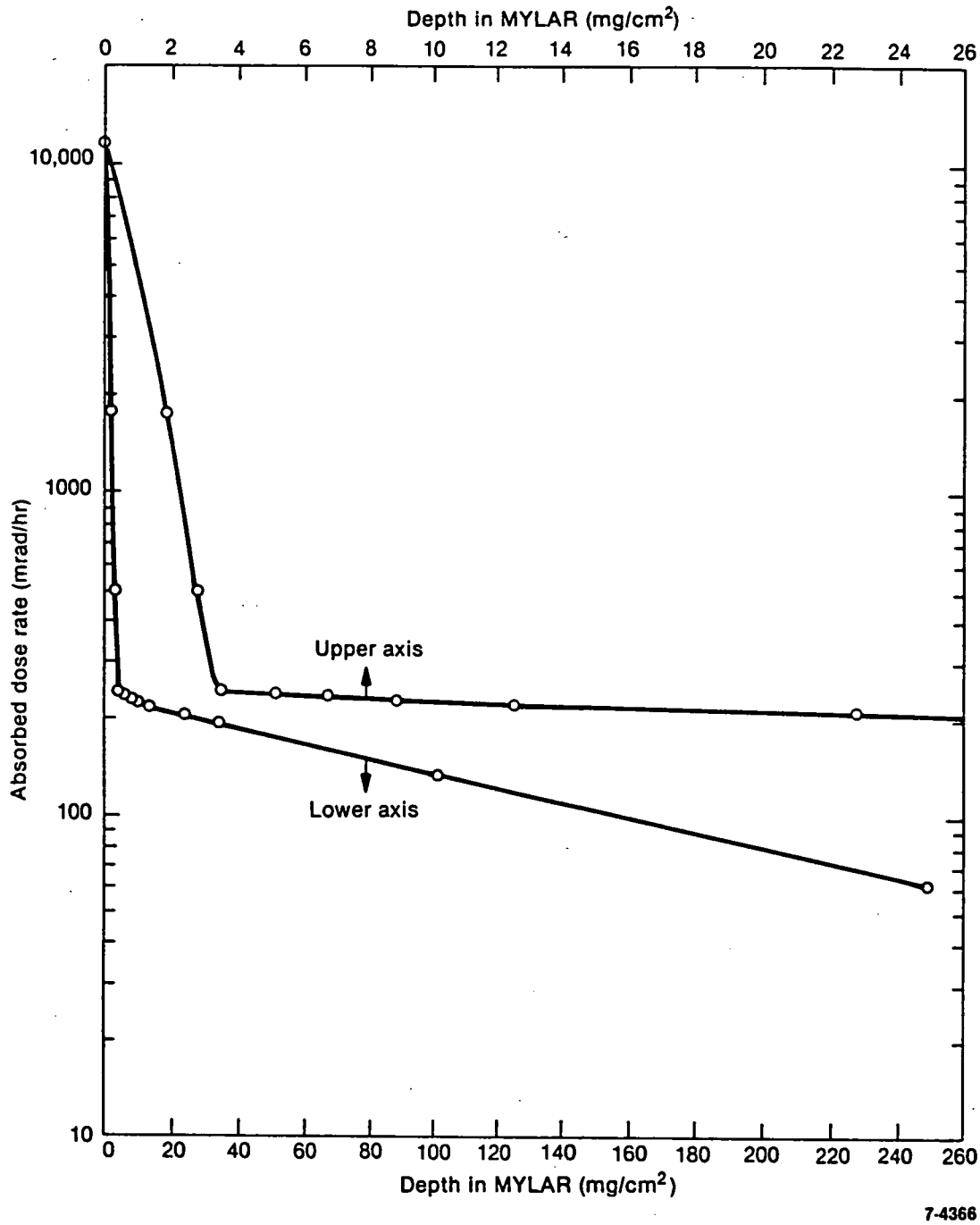

Figure 7-2. Absorbed dose rate as a function of depth in Mylar.

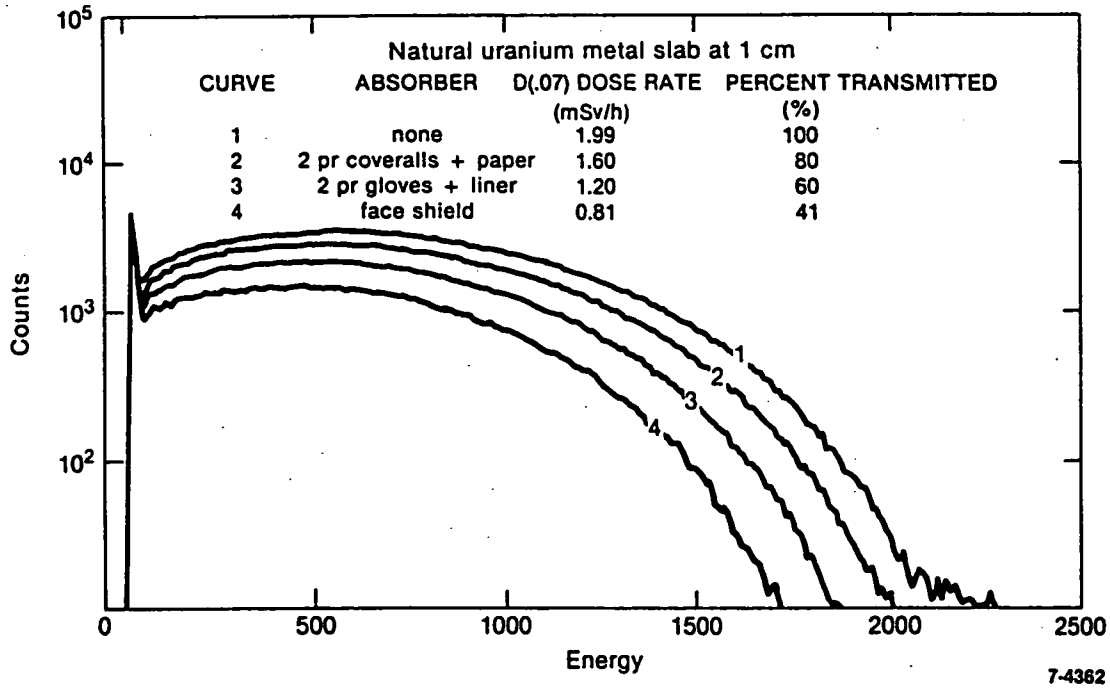

Figure 7-3. Changes in beta energy spectra and shallow dose rate from a natural uranium metal slab source caused by protective apparel. Note the bremsstrahlung peak in the low energies range.

be predicted. This mix of uranium isotopes and daughter radionuclides may be estimated by using an equation developed to predict specific activity as a function of enrichment. Figure 2-2 (Section 2) shows the estimated activities of the uranium isotopes as a function of enrichment as predicted by the reference equation.

Beta radiation fields are usually the dominant external radiation hazard in facilities requiring contact work with unshielded forms of uranium. Table 7-2 shows the contribution of beta emissions from U-238, U-233, U-234, and their daughter products. Most of the beta dose rate from U-238 comes from the 2.29 MeV beta emitted by protactinium-234m.

The information in Tables 7-2 and 7-3 can be utilized to provide the values in Figure 7-1 which gives the estimated beta dose rates from a semi-infinite slab of uranium metal of various enrichments. One can see from this figure that for uranium of enrichments up to 30%, the beta radiation field is dominated by contributions from U-238 decay products. Thus, for uranium of these enrichments, one is dealing essentially with 2.29 MeV maximum betas.

Gamma radiation from uranium is normally not the controlling problem. For example, the contact beta radiation field from depleted uranium is approximately 240 millirem per hour, while the contact gamma radiation field is less than 10 millirem per hour. Although gamma radiation fields from uranium are not usually the dominant problem, significant gamma fields can exist in areas where large quantities of uranium are stored. Neutron fields from enriched uranium fluoride compounds can also add to this area of concern. Care should be taken to assure that dose-equivalents from such fields are kept to levels which are ALARA. Table 7-4 lists the major gamma emissions from uranium isotopes.

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TABLE 7-2. MAJOR URANIUM BETA AND GAMMA EMISSIONS (MEV)

Radionuclide	Beta (Max)	Gamma
U-238	None	None
Th-234	0.103 (21%) 0.193 (79%)	0.063 (3.5%) 0.093 (4%)
Pa-234m	2.29 (98%)	0.765 (0.3%) 1.00 (0.6%)
U-235	None	0.144 (11%) 0.186 (54%) 0.205 (5%)
Th-231	0.140 (45%) 0.220 (15%) 0.305 (40%)	0.026 (2%) 0.084 (10%)
U-234	None	0.053 (0.2%)

TABLE 7-3. BETA EMISSIONS FROM URANIUM AND DAUGHTER RADIONUCLIDES

Radionuclide	Maximum Beta Energy (MeV)	Decay Fraction	MeV per Transformation	% MeV per Transformation
U-238	None	0.00	0.00	0.0
Th-234	0.103	0.19	0.020	0.8
Th-234	0.193	0.73	0.141	5.9
Pa-234m	2.29	0.98	2.244	93.3
Total MeV/Transformation = 2.405				
U-235	None	0.00	0.00	0.0
Th-231	0.14	0.45	0.063	28.9
Th-231	0.22	0.15	0.033	15.1
Th-231	0.305	0.40	0.122	56.0
Total MeV/Transformation = 0.218				
U-234	None - no significant beta contributions			

Assumptions:

1. The mix of radionuclides seen above is that which might be expected from the "pure" uranium isotope that has had time for the short lived daughters to grow in.

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TABLE 7-4. MAJOR GAMMA EMISSIONS FROM URANIUM AND DAUGHTER RADIONUCLIDES

Radio-nuclide	Gamma Energy (MeV)	Branch Fraction	Decay Fraction	Effective Yield	MeV per Trans-formation	% MeV per Trans-formation
U-238	None	1.00	0.00	0.00	0.00	0.0
Th-234	0.063	1.00	0.04	0.04	0.0025	27.6
Th-234	0.093	1.00	0.05	0.05	0.0047	51.9
Pa-234m	0.766	1.00	0.0002	0.0002	0.00015	1.66
Pa-234m	1.001	1.00	0.0006	0.0006	0.0006	6.63
Pa-234	0.100	0.0013	0.5	0.0006	0.00006	0.66
Pa-234	0.700	0.0013	0.24	0.0003	0.00022	2.43
Pa-234	0.90	0.0013	0.70	0.00091	0.00082	9.06
U-235	0.143	1.00	0.11	0.11	0.0157	11.66
U-235	0.185	1.00	0.54	0.54	0.100	74.29
U-235	0.204	1.00	0.05	0.05	0.010	7.43
Th-231	0.026	1.00	0.02	0.02	0.0005	0.37
Th-231	0.084	1.00	0.10	0.10	0.0084	6.24
U-234	0.053	1.00	0.002	0.002	0.0001	100.0

Assumptions:

1. The mix of radionuclides seen above is that which might be expected from the "pure" uranium isotope that has had time for the short lived daughters to grow in.

7.2.4 Dose Equivalent Rates Vs Depth in Tissue

Skin DE rates from uranium occur primarily from the relatively energetic Pa-234m betas at tissue depths of 4 mg/cm^2 and greater. Dose rates were measured by Plato with an extrapolation chamber at various depth in a tissue equivalent medium (mylar). See Figure 7-2. The dose values at less than 4 mg/cm^2 result from alpha particles and are of no concern from an external radiation exposure standpoint.

The data in Figure 7-3 was obtained with a tissue equivalent plastic scintillation detector and demonstrates the spectral changes and the resultant dose rates under typical protective clothing. It can be seen from Figures 7-2 and 7-3 that significant fractions of the uranium beta radiation will penetrate typical protective clothing worn in facilities which process uranium.

7.3 Radiation Detection and Evaluation

7.3.1 Portable Survey Instruments Beta Radiation Response

The primary exposures of concern when handling bare uranium materials come from the beta radiation. The accuracy and precision of survey instruments used for measurement of beta radiation fields depend upon some or all of the following factors:

a. beta energy response
b. angular response of instrument
c. source-detector geometry factors
d. detector construction (window thickness, etc.).

Energy Dependence

Most commercial radiation survey instruments currently available under-respond to beta radiation fields from uranium. Figure 7-4 shows the energy response of a group of typical survey instruments as measured by

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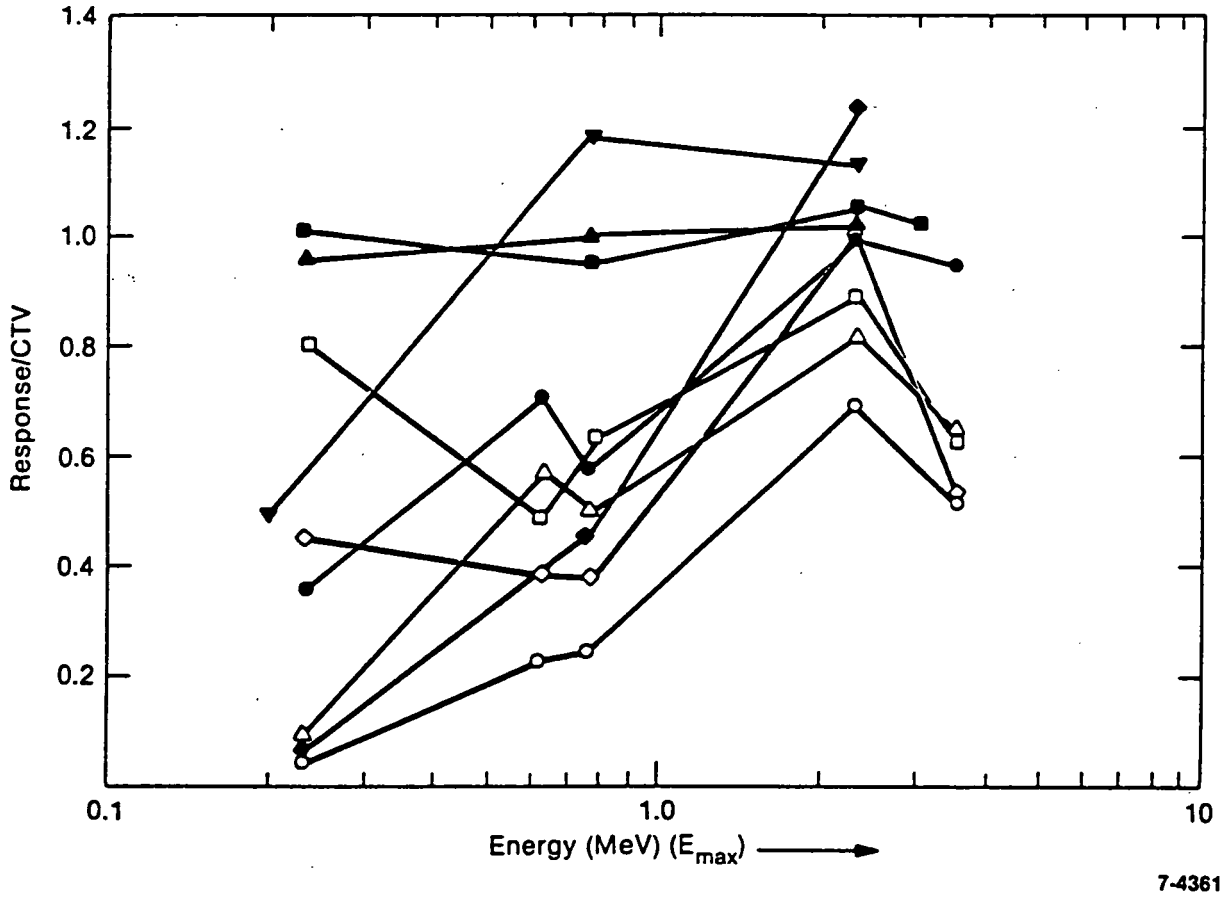

Figure 7-4. Energy dependences of several beta survey meters as determined by the Battelle Pacific Northwest Laboratory.

Battelle Pacific NW Laboratory. Figures 7-5 and 7-6 show the beta and gamma spectra measured with a Tissue Equivalent plastic scintillation detector and compares dose rate measurements with four different detectors. The extrapolation chamber values are considered standard/correct. Table 7-5 presents typical survey instrument response to uranium fields specifically. At best, typical "beta correction factors" (true dose rate/indicated dose rate) are on the order of 1.5 to 2. This under-response is due primarily to (a) the angular response of the detector and (b) attenuation of the dose-rate by the detector window and the sensitive volume of the detector.

Currently, skin dose measurements are related to the dose at a depth of 5 to 10 mg/cm^2 in tissue. Window thicknesses of commonly available survey instruments range from on the order of 7 mg/cm^2 to several hundred mg/cm^2. An ideal survey instrument would have a 5 mg/cm^2 tissue equivalent (TE) detector under a 5 mg/cm^2 window thickness.

Even if the window provides only minimal attenuation, the attenuation of the beta dose-rate through the sensitive volume of large detectors remains a problem. The detector indicates the <u>average</u> dose-rate throughout the sensitive volume. The "true" dose-rate is that which occurs in the first 7 mg/cm^2 of detector (assuming the detector is tissue equivalent). The instrument will under-respond by the ratio of this average dose-rate to the "7 mg/cm^2" dose-rate. This sensitive volume under-response is a function of the beta energy distribution and of the size and shape of the sensitive volume.

<u>Angular Response</u>

The construction of most survey instruments (e.g., "cutie pie") leads to a severe angular dependence on beta radiation fields. This angular response is due to the attenuation of the beta dose-rate by the walls of the detector as the window is moved away from the source.

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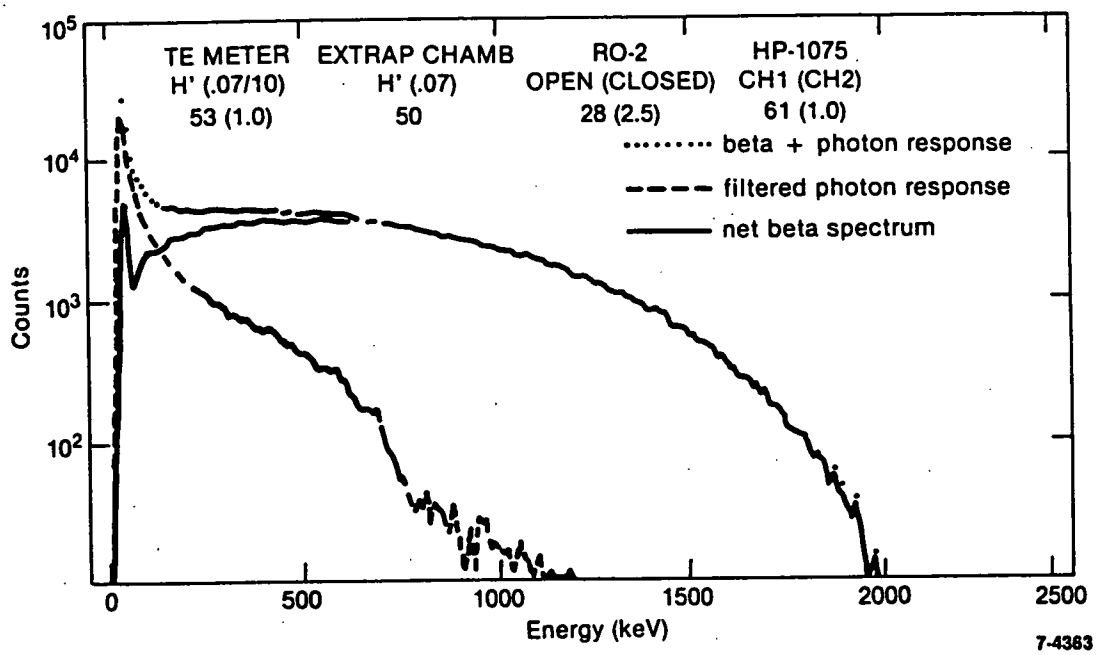

Figure 7-5. Comparison of meter readings in mrem/hr for a depleted uranium ingot.

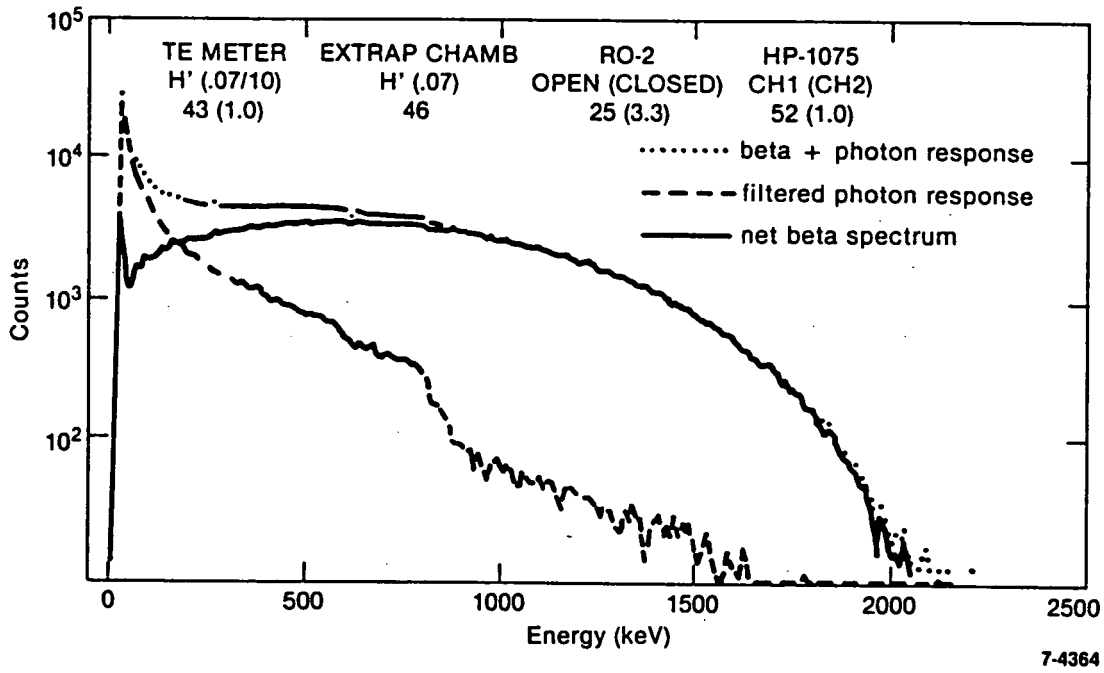

Figure 7-6. Comparison of meter readings in mrem/hr for an open drum of UF_4 (green salt).

TABLE 7-5. INSTRUMENT RESPONSE TO URANIUM BETA FIELDS

Instrument	Window (mg/cm^2)	Beta * Correction Factor	Exposure Geometry
Victoreen 471	1.1	1.4	30 cm from U foils
Eberline RO-2	7	2.0	30 cm from U foils
Eberline RO-2A	7	4.0	Contact with DU slab
Al-walled GM	30	1.7	30 cm from U foils
Victoreen Radector III	34	14	Contact with DU slab
HPI-1075	7	1.8	Contact with DU slab
"Teletector"	30 (low range)	50	Contact with DU slab
Eberline PIC-6A	30	40	Contact with DU slab
British BNL-3	7	1.3	1.5 cm from 100 cm^2 DU

* True reading/measured value.

Figure 7-7 demonstrates the response of a tissue equivalent ($5\ mg/cm^2$ TE detector under a $5\ mg/cm^2$ window and mounted in a TE phantom) response to off axis Sr 90/Y 90 betas (similar energies to those from uranium). Skin tissue dose response is greater to off axis betas and survey instruments which effectively shield these high angle particles will under-respond compared to skin tissue.

Source-Detector Geometry

Measurements taken close to small beta sources may be inaccurate due to non-uniform irradiation of the sensitive volume of the detector. Since uranium in most DOE facilities tends to present wide-area source of beta radiation, significant non-uniform irradiation would not normally be encountered.

Detector Construction and Use

Characteristics of instrument construction may significantly affect their response and use. For example, many survey instruments have "beta windows" which are intended to "discriminate" between beta and gamma radiation. Obviously, measurements of beta dose-rate must be made with the "beta window" open. It should be noted, however, that a number of instruments have "beta windows" which are only a few hundred mg/cm^2 thick. Such windows can transmit a significant fraction of the dose-rate from high energy beta emitters (e.g., Pa-234m). Thus, up to 10 or 20% of the "gamma only" reading may be due to the higher energy betas penetrating the so-called "beta window."

Occasionally, survey instruments are placed in plastic bags or covered in order to protect them from becoming contaminated. Bagging the instrument places additional absorber between the radiation field and sensitive volume of the detector. Calibration of the instrument (or application of a correction factor) should take this additional shielding into account.

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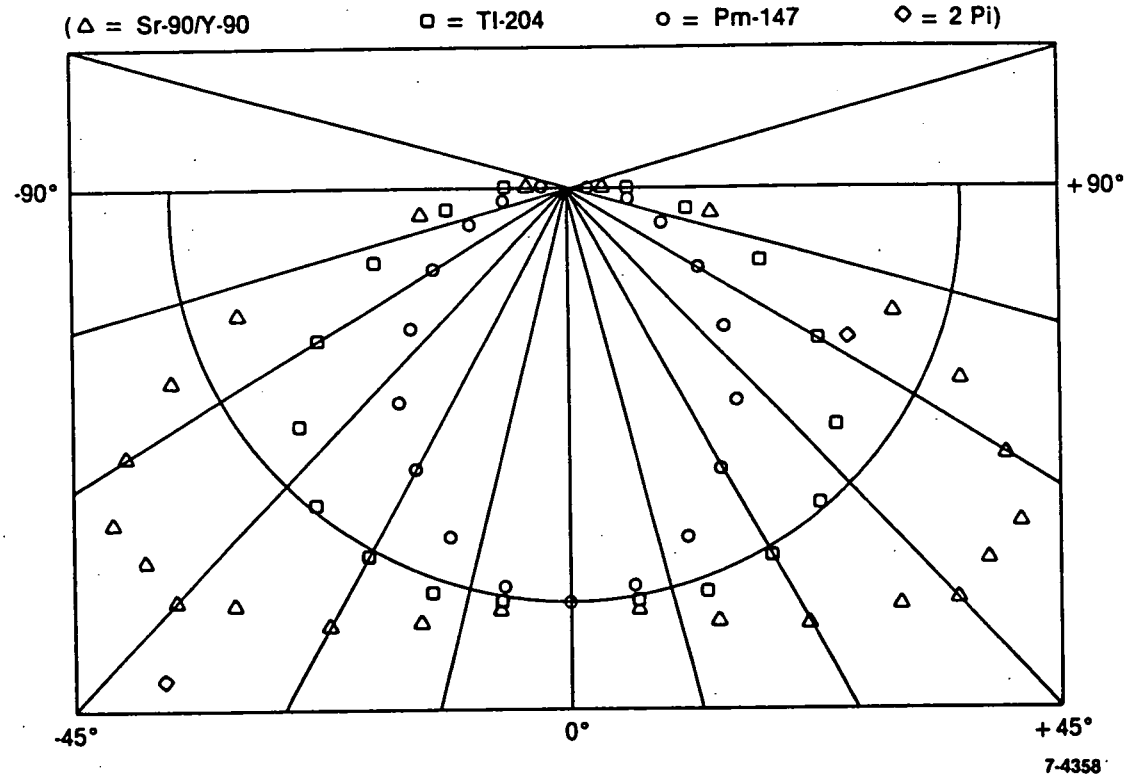

Figure 7-7. Measured angular response of the INEL TE survey meter to parallel beams of beta particles from three standard beta sources.

Gamma Radiation Response

As previously indicated, the external dose which results from gamma and x-ray radiation from bare uranium is a small fraction of the total. However, it represents the "penetrating" or whole body dose source and is the only source of radiation from contained (glove boxes, etc.) facilities. Survey instruments are typically calibrated with Cs-137 (0.663 MeV) photons. Typical portable survey instruments demonstrate a fairly "flat" energy response above 250 keV, while the response below 250 keV can be variable to a greater or lesser degree depending upon the instrument design. Figures 7-8 and 7-9 show average response of a group of commercial survey instruments. Figure 7-10 shows a typical gamma spectra from a uranium oxide source while Table 7-6 demonstrates the wide variation that can occur in the photon spectra that can occur at various locations in a single plant. This demonstrates the desirability of using ion chambers or compensated beta instruments for dose-rate measurements. It also indicates the need to know the energy response of the instrument used and the value or at least qualitative knowledge of the photon spectra at the various work stations.

7.3.2 Personnel Dosimetry

Personnel dosimeters produce the data which becomes the formal or "legal" record of personnel exposure. However, these detectors experience many of the same energy dependence and angular response problems encountered by survey instruments. The most difficult problem is relating badge results to the shallow or skin dose.

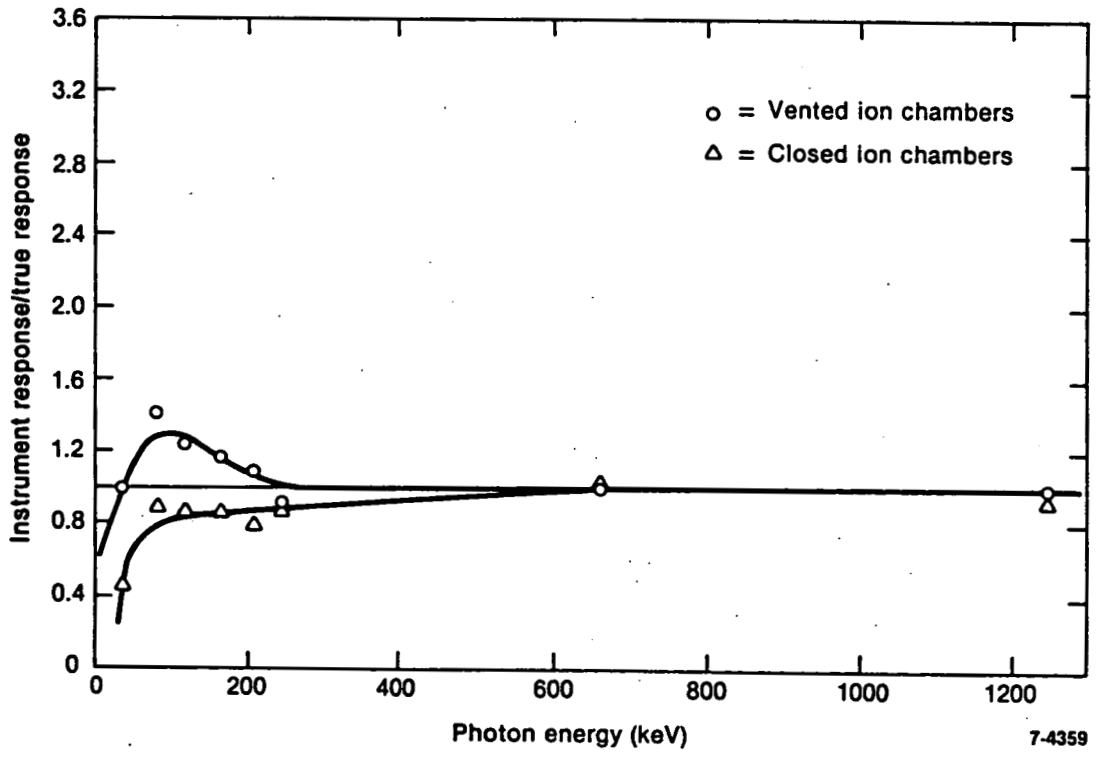

Figure 7-8. Average ion chamber survey meter response by group to X or gamma photon radiation.

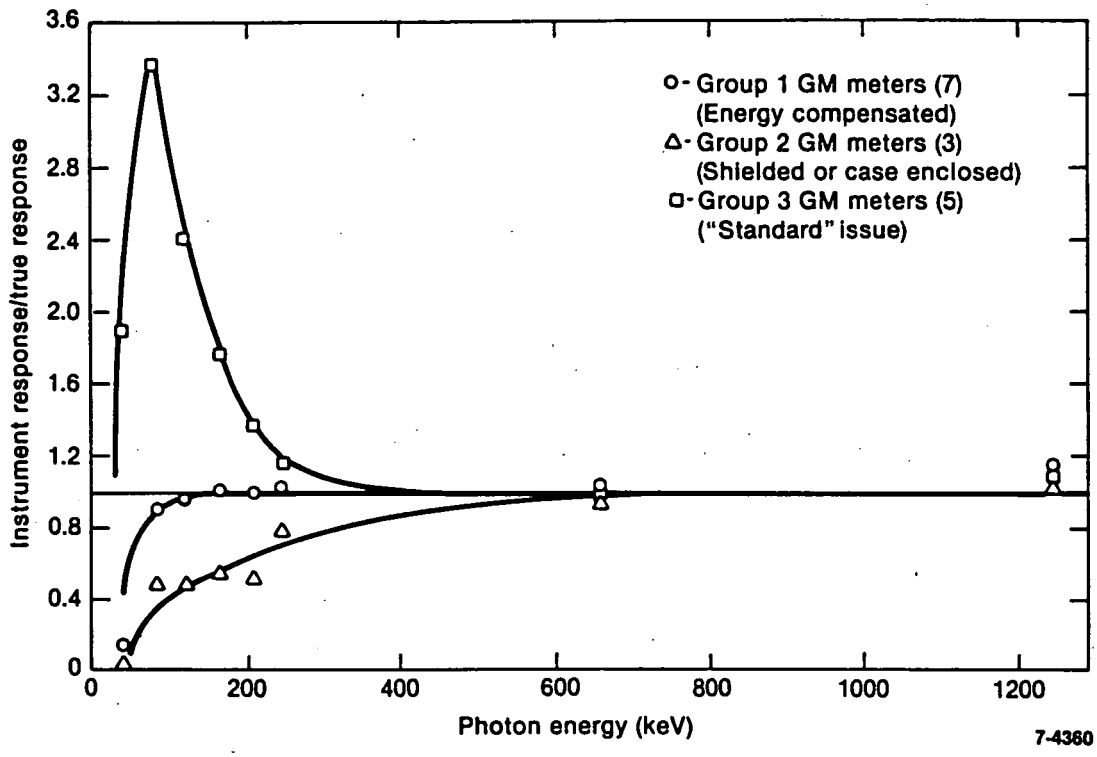

Figure 7-9. Average GM survey meter photon energy response by group.

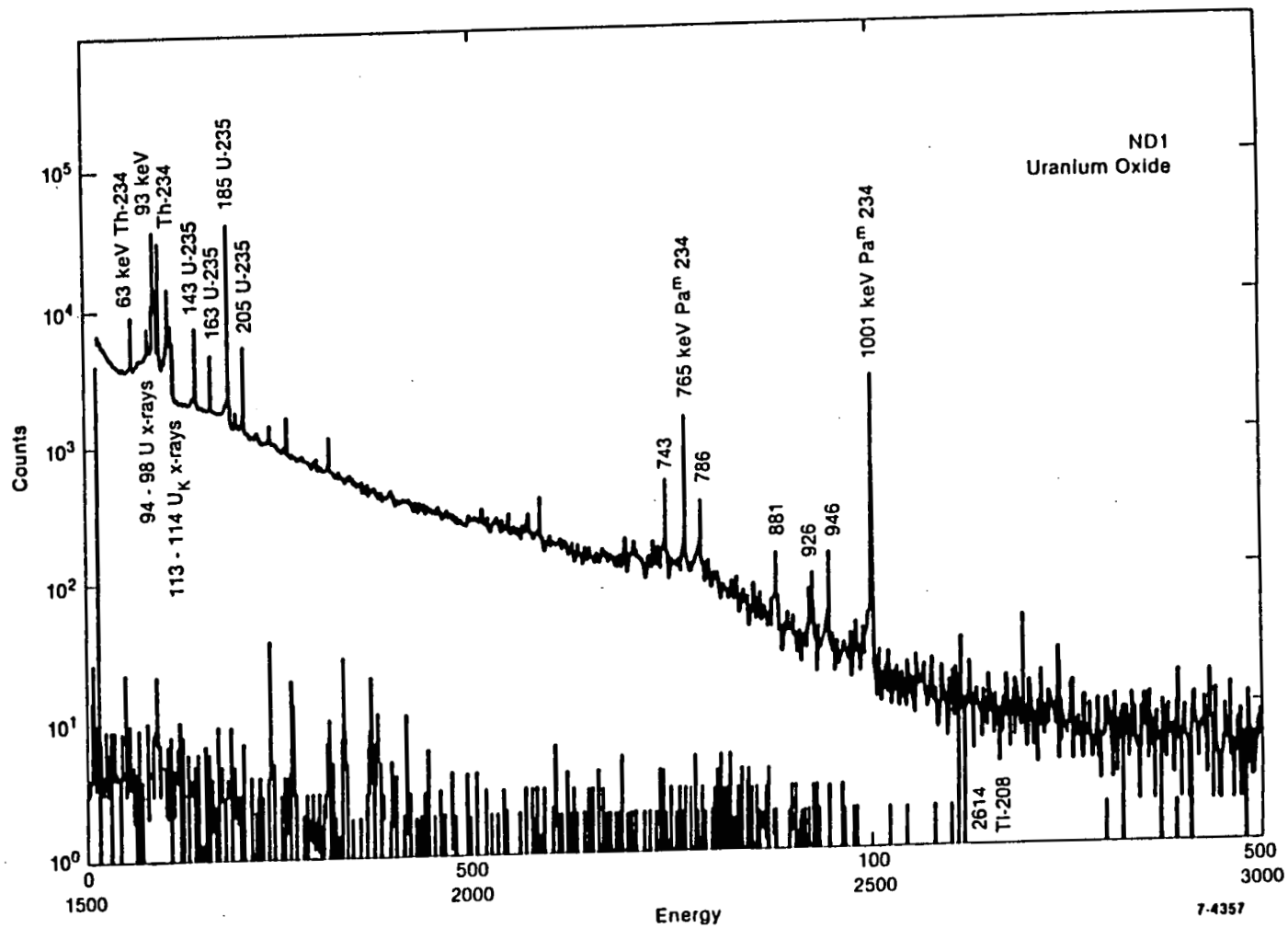

Figure 7-10. High resolution gamma spectrum of slightly enriched uranium oxide (1% U-235) recorded with Ge(Li) detector.

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TABLE 7-6. GAMMA FLUX AND RATIOS AT VARIOUS LOCATIONS AND SOURCES AT FERNALD PLANT

Source Description or Location	Integrated Gamma Flux (photons/cm^2/sec) 30 to 225 keV	675 to 1050 keV	Ratio
Crucible load station - 55 gal drum	990	348	2.8
Beside UO_3 barrel	538	159	3.4
Open UO_3 barrel	919	232	4.0
Tube cutting work station - metal	253	58	4.4
Outside plant 9 south entrance by exhaust fan	776	165	4.7
Box of black top crop at 25 cm	848	154	5.5
Lathe work station	424	76	5.6
Background outside Building 3045	35	5	7.0
Near "thorium" hopper	424	58	7.3
Plant 9 west wing - SW hot area	708	72	9.8
Crucible burnout station	776	69	11.2
Plant 9 HP change room	5	< 0.4	12.5
Background 75 ft from Bldg. 3045	25	2	12.5
Graphite crucible (G-8010) 30 cm	183	11	16.6
Graphite crucible (3898) 30 cm	310	18	17.2

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Thermoluminescent dosimeters (TLDs), currently the dosimeter of choice in most DOE uranium facilities, provide the most accurate and precise means of measuring doses received by workers. Film badges are also used by a few facilities. Although the following discussion focuses on the more widely used TLD detector systems, the basic principles apply to the film detectors also with the added uncertainties associated with the increased susceptibility of film to environmental factors, such as temperature, humidity, pressure, etc. Great care is necessary to assure that the shallow and deep doses are being accurately discriminated and measured.

An ideal dosimeter would directly measure doses at 7, 1000, and perhaps 300 mg/cm^2 (shallow, deep, and lens of eye doses). In practice, the dose at such depths in tissue must be inferred from a combination of measurements with different filters. TLD and film elements are mounted in a badge arrangement which is covered by at least 10 to 30 mg/cm^2 of mylar, paper, or other covering for mechanical and/or protective reasons.

Energy Dependence

Personnel dosimeters are beta energy dependent for the same reason that survey instruments are beta energy dependent. That is, the reading obtained from the TLD or film is proportional to the average rate of energy deposition through the "sensitive volume" or body of the element. If this average energy deposition is less than the deposition at 7 mg/cm^2, then the dosimeter will under-respond.

TLD chips of lithium fluoride (0.125 x 0.125 in.) are about 240 mg/cm^2 thick. Significant attenuation of the beta dose rate takes place through the body of the chip. As a result, these types of TLD chips under-respond to uranium decay betas by a factor of about two.

Other TLD badge systems minimize this problem by employing a thin layer of TL powder "glued" onto a plastic backing. Current TLD personnel dosimeters typically use multiple (typically 4) detectors under different

filter thicknesses. The different responses of each element are used as input to an algorithm which provides an estimate of the effective radiation energy and the doses at depths of interest. Detectors which are very thin minimize energy dependence. Film detectors demonstrate a high energy response dependence for low energy photons, as well as beta energy response dependence (though the beta response is less variable than the TLD chips).

Current systems have the potential of providing accurate and precise information; however, their complexity can lead to problems. The calibration of these systems should be performed by a person with specific expertise, not only in the detector's system but in how the badge responds in high beta or mixed beta and gamma radiation fields.

Angular Dependence

The dosimeter elements must be mounted in a badge or element holder. The assembled badge usually displays severe angular dependence. Fortunately, in most cases, a worker's normal movements will tend to average out some of this dependence. Some badge holder arrangements can "flip" the badge completely over so that the "beta window" of the badge is facing the worker, not the source. The design of the badge holder or strict administrative controls should be utilized to minimize this problem.

Dosimetry Practices

Beta and gamma fields in working areas should be well-characterized. See previous figures and tables as examples. Areas with potential for high beta fields from separated uranium decay products should be posted accordingly. An attempt should be made to correlate survey instrument and dosimeter badge results.

Badge reading frequency should be long enough to accumulate a significant dose (100 mrem range) and short enough to allow adequate

control. Monthly change frequencies are generally sufficient for radiation workers in uranium facilities, though this can vary with the specific work site conditions.

State of the art dosimetry should be used to monitor worker's exposure to uranium. Although multiple badging is not usually necessary, it should be considered for use in very high beta fields produced by separated uranium decay products. The dosimetry system used should meet DOELAP standards and be specifically designated for measuring both shallow and deep doses from uranium.

Potential for badge contamination should be minimized. Where the potential for such badge contamination exists, badges should be frequently checked for contamination.

Extremity Dosimetry

Measurement of the dose to the hands and/or forearms is difficult. Typically, such measurements are made with TLD chips or TL powder in finger rings or wrist dosimeters. Such devices do not allow for all of the sophisticated energy discrimination just discussed. The inhomogeneity of beta radiation fields coupled with the angular dependence of commonly available extremity dosimeters can result in a probability of underestimating the dose. However, the careful consideration of the typical exposure conditions at the work site (handling metal pieces, glove box work, etc.) and calibration of the dosimeters with appropriate sources (uranium plaque sources, etc.), extremity doses can be measured with acceptable accuracy for protective purposes.

Care should be exercised in preventing "obvious" underestimations of extremity dose. For example, finger rings worn on the "top" of the finger (opposite the palm side of the hand) will not measure the dose received by the palm side when handling metal rods, etc. Dosimeters worn on the wrist have been shown to underestimate the beta dose to the fingers and palm. Reference to the Bibliography information sources will provide health physics programs with background in current techniques and considerations.

Dose to Lens of Eye

It is sometimes assumed that if the skin limit is not exceeded the dose limit to the lens of the eye will not be exceeded. Such assumptions should be well supported by calculations or preferably actual measurements. See Figure 7-3 for data indicating significant uranium beta penetration of even face shields. It is suggested and is a common practice in most fabrication areas to require the use of safety glasses which tends to mitigate this concern.

7.4 External Dose Reduction

Reduction of personnel exposures to levels which are ALARA is largely a matter of common sense applied to the principles of time, distance, and shielding. The first step in any dose reduction program is to assure that the radiation fields are adequately identified, characterized, and measured. Only after this step has been performed can an optimum reduction in doses be achieved for a given amount of time, money, and energy.

7.4.1 Time

As a general rule, a reduction in exposure time will yield a reduction in doses. Accordingly, operations resulting in significant doses should be reviewed for possible reductions in worker exposure time. Traffic and material flow in proposed facilities should be closely examined for opportunities to reduce exposure time.

7.4.2 Distance

Beta dose rates from uranium and its decay products decrease rapidly with distance due to geometry and air shielding while gamma and neutron radiation decrease less with distance due to scattering buildup. Uranium facilities usually involve a high percentage of contact work and

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considerable dose reduction can result from simple techniques to make operations semi-remote and allow workers to function. Even short distances can effect significant dose reductions.

7.4.3 Shielding

Shielding is probably the most widely used (and most effective) method of reducing beta doses from uranium. Relatively lightweight, cheap and flexible shielding (e.g., plastic, rubber mats) has been used very effectively. Figure 7-3 demonstrates the spectral basis for shielding and lists a few protective clothing reduction factors. Table 7-7 lists the thicknesses of common shielding materials necessary to stop essentially all of the beta particles from uranium (i.e., Pa-234m). Generally, the light shielding materials are used whenever possible to eliminate bremsstrahlung as well as beta radiation fields.

Protective clothing commonly worn in the nuclear industry can also afford beta dose reduction. Figure 7-3 and Table 7-8 list approximate dose reduction factors provided by such clothing. Particular attention should be paid to the use of gloves for "hands-on" work. Although lightweight rubber gloves provide some reduction, consideration should be given to using heavy leather or even leaded gloves for operations which do not require manual dexterity. Such gloves can be particularly effective in handling materials emitting high beta fields from unsupported uranium decay products.

Contamination build-up inside of work gloves has lead to unacceptable hand doses in some facilities. Re-use of leather or cloth gloves should be reviewed carefully for such build-up. Workers should wear thin, anti-contamination gloves inside of the heavy gloves.

TABLE 7-7. URANIUM BETA SHIELDING

Material	Approximate Material Thickness Required to Stop Pa-234m Betas. (cm)
Air	850.
Aluminum	0.41
Lead	0.10
Lucite	0.92
Pyrex Glass	0.49
Polyethylene	1.2
Stainless Steel (347)	0.14
Water	1.1
Wood	1.7 (approx)
Uranium	0.06

TABLE 7-8. URANIUM BETA DOSE REDUCTION FACTORS

Item	Reduction Factor
Vinyl surgeons gloves	0.95
Latex surgeons gloves	0.87
Lead loaded (10 mil Pb equivalent)	0.77
Lead loaded (30 mil Pb equivalent)	0.13
Pylox gloves	0.62
Leather (medium weight)	0.62
White cotton gloves	0.89
"Tyvek" Coveralls	0.98
"Durafab" paper lab coat	0.96
65% Dacron/35% cotton lab coat	0.91

Dose to the lens of the eye can be effectively reduced through the use of ordinary glasses, safety glasses, or face shields. Such eye protection should be required when workers are dealing with the high beta fields from concentrated uranium decay products.

7.4.4 Geometry

It should be remembered that the beta radiation field from uranium is strictly a surface phenomenon. This fact can be taken advantage of in some circumstances. For example, large plates or sheets of uranium metal, if stored in racks "edge on" will present less of a beta (and gamma) radiation field.

7.5 Record Keeping

Dosimetry systems should be capable of providing routine results within a reasonable time period. The system of badge collection and re-distribution should be well defined and minimize the possibility of lost badges.

Badge reading systems should have established "action levels" to alert technicians or operators to unusual results. Such results should include readings or TLD element ratios in excess of certain levels. If possible, the system should automatically save glow curves of any unusual results.

Personal dose information should be available to the individual worker. Workers should be provided with at least annual "dose report cards" which summarize dose status for the calendar year.

The radiation safety organization should provide summaries of dose information to line management for use in ALARA planning. Such information may include overall individual and group dose status, rate of dose accumulation, and any unusual dose patterns.

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SECTION 8

NUCLEAR CRITICALITY CONTROL

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8-1 Approximate correlation of exposure with distance from a solution excursion of 10^{17} fissions. Arrows appear where it is believed that available estimates are displaced from the probable values 8-9

SECTION 8

NUCLEAR CRITICALITY CONTROL

Nuclear criticality control for uranium facilities involves measures to maintain conditions for processing, handling, storing, or transporting uranium such that a critical mass is prevented. When the fissioning process or criticality is controlled in facilities designed for the purpose the results are beneficial. Nuclear energy is attractive because a relatively small mass of fissionable material can create a large amount of energy which in turn can be used to heat water to drive turbines, etc. However, when a critical mass is accidentally assembled in an unprotected area, the relatively small mass and large release of energy can result in major deleterious effects, including high radiation exposures to personnel, release of radioactive materials to the environment, contamination of facilities, etc.

The nuclear criticality safety (NCS) discipline has as one of its primary purposes the prevention of accidental assemblies of critical masses thus protecting personnel, the public, the plant and associated equipment. In addition, NCS provides technical assistance in planning emergency measures to mitigate the effects of accidental criticality events even though the primary efforts to prevent such an accident have made such events highly improbable.

8.1 Background and Scope

This section is not intended to present a definitive treatment of criticality safety principles or to duplicate existing guides and/or plant-specific manuals or procedures. For those HPs desiring increased understanding of criticality safety principles and parameters, reference is made to this section's Bibliography.

However, because of the technical requirements of the Health Physics discipline, applied radiation safety personnel are required to possess a

basic understanding of nuclear principles and are involved with a majority of the work performed with radioactive materials in the facilities. They represent a knowledgeable, safety-oriented "presence" to assist in detecting inadvertent problems related to NCS. Within DOE experience there have been occasions when a health physics professional or technologist detected plant personnel preparing to collect fissionable material in unsafe containers (plastic bags, large buckets, etc.) and because of general understanding of NCS principles were able to prevent unsafe actions. This section is intended to (a) emphasize the importance of avoiding NCS violations and other related problems, (b) encourage plant-specific training in NCS principles for health physics personnel, (c) encourage the awareness of HP personnel in detecting and avoiding NCS problems at the applied level, and (d) stimulate a cooperative relationship between the radiation safety and NCS disciplines in DOE facilities. In addition, there are actions which would be logical from radiation protection or contamination control standpoints, but which could deleteriously impact NCS. Examples are (a) adding shielding which would add reflection and change the critical mass, (b) wrapping a pipe leaking enriched uranium solution with plastic, and (c) securing leakage paths from hoods which could result in solution accumulation. If HP personnel detect what appears to present NCS implications, it is important that an NCS expert be called to verify and evaluate the situation.

8.2 Criticality Factors

There are several factors which have an effect on establishing or preventing a condition in which a criticality occurs.

8.2.1 U-235 Enrichment

Natural uranium has about 0.7% U-235 (the fissile isotope). Uranium enrichment increases this percentage. Enriched uranium is normally required to provide sufficient fissile material to sustain a critical or sustained nuclear reaction in a small enough mass to meet the needs of the system. Handling of natural (0.7% U-235) or depleted (< 0.7% U-235)

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uranium is generally safe at DOE U processing facilities since deliberate engineering efforts such as moderation with heavy water, reactor grade graphite, etc., is required to create a critical mass with natural uranium. However, safe handling measures should be observed when the enrichment is 0.9% or more.

8.2.2 Critical Mass

The minimum mass of uranium which sustains a chain reaction for certain conditions is called the critical mass. Critical mass is dependent upon U-235 enrichment and other factors such as the amount of moderator. For example, the single parameter limit to the critical mass of U-235 in aqueous solution and water reflected is 760 grams of U-235.

8.2.3 Density or Concentration

Density or concentration is defined as mass per unit volume (g/l, etc.). A uniform solution or slurry less than 10.8 gm U-235/ℓ would be subcritical at any volume, while a concentration four or five times greater could result in the minimum critical mass.

8.2.4 Moderation and Reflection

A moderator is a material which slows down fast neutrons. Moderation occurs in the presence of elements having a low atomic weight such as hydrogen. The hydrogen concentration is usually expressed as the ratio of the number of hydrogen atoms to the number of fissile atoms of the isotope. Thus, the extent of moderation is expressed as H/U-235 ratio. The ratio H/U-235 may range from zero for metal or a dry unhydrated salt to several thousands for dilute aqueous solution. Over this concentration range the critical mass may vary from a few tens of kilograms (with little hydrogen) through a minimum of a few hundred grams (at optimum moderation) to infinity in a very dilute solution where the neutron absorption by hydrogen makes a chain reaction impossible. A moderated system allows a smaller mass of U-235 to become critical.

A reflected system is an assembly of materials containing U-235 which is partly or wholly surrounded by another material which has a greater neutron scattering cross section than air. Thus, a fraction of the neutrons leaving the system (core) are returned to the core (reflected) instead of leaving the system. A good reflector is a material which favors a low neutron absorption and high neutron scattering. Water, concrete, graphite, and stainless steel are typical "good" reflectors, although any material will serve as a reflector. A "fully reflected" system is one which is totally surrounded by the reflector and in which an increase in the reflector thickness results in little if any decrease in the critical mass. For example, eight inches of water is essentially fully reflected.

8.2.5 Geometry or Shape

Leakage of neutrons from a system depends on the shape of the system and on the neutron-reflecting properties of surrounding materials. The shape and size of containers are determined by considering the ratio of surface (S) to the volume (V). The ratio S/V is maintained at a value which prevents a chain reaction regardless of the quantity of material contained. Maximum safe geometries for aqueous uranium solutions are given in the Bibliography references.

8.2.6 Interaction or Arrays

Interaction is the exchange of neutrons between separate containers containing uranium material. An increase in the exchanged neutrons increases the fission reaction rate. Units that are subcritical individually can be made into a critical array if brought near each other.

8.2.7 Neutron Poisons (Absorbers)

Neutron absorbers (poisons) are nonfissionable materials which capture neutrons, thus reducing the number of neutrons which are available for a fission reaction. Cadmium, boron, and chlorine are examples of neutron

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absorbers. Boron in borosilicate glass Raschig rings and chlorine in polyvinyl chloride rings (CPVC) are poisons used in some applications.

8.3 Health Physics Role

8.3.1 Information

The preceding discussion is intended to list general areas or topics of concern in preventing accidental nuclear criticality events. It is imperative that specific operations and/or facilities be evaluated by NCS specialists to establish site-specific NCS programs and controls. The health physics function would involve an understanding of NCS program structure, administrative controls, and some engineering criteria as an aid in administrative control assistance. Health physics is usually more directly involved with NCS alarm systems, Nuclear Accident Dosimetry, and Emergency Response.

Information important to the health physics function are:

1. NCS alarm system including alarm design parameters (type of detectors, detector area coverage, alarm set-points and basic control design).

2. Postulated criticality accidents, including information on type (e.g., burst or "slow cooker").

3. Magnitude of the most likely accidents (number of fissions, neutron flux and energy distribution, fission gamma rates and potential fission gas release).

4. Locations and scenarios for designing the Nuclear Accident Dosimetry (NAD) program and formulating plans for Emergency Response.

8.3.2 Guidelines

A few basic guidelines for health physics responsibilities related to NCS are:

1. Develop through site-specific training, an NCS awareness of criticality parameters sufficient to enable health physics personnel to identify conditions/operations which violate local NCS controls and/or policies. No pretense at NCS expertise should be made. A trained awareness and a cultivated alertness should be sufficient.

2. Maintain adequate monitoring capability for a nuclear criticality excursion. This would include remotely operated high-range gamma instruments, personal alarming dosimeters for emergency response/rescue teams, neutron monitoring instrumentation (in case of sustained low-power critical reaction, "slow cooker"), air sampling capability for fission gases and their daughters, NAD system and backup capability in case evacuation of personnel is required, etc.

3. Assist in training emergency response personnel to quickly identify criticality situations and potential additional hazards. For example, response personnel must be able to quickly determine type of criticality (burst, "slow cooker" or multi-burst), the difference in exposure hazards and methods for quickly estimating exposure levels for rescue or amelioration of the incident.

4. Arrange for providing health physics support (contamination monitoring, dose estimates,etc.) to medical personnel in treatment of exposed/injured employees.

The preceding guidelines are not exhaustive. For further information on this subject, see references in the Bibliography at the end of this Section.

8.4 Criticality Accident Experience

8.4.1 Processing Plant Experience

Current NCS practice has been influenced both by the successful overall experience of the nuclear industry and by the analysis of the few accidental criticality excursions that have occurred. There have been eight criticality accidents in chemical process equipment; all have occurred in aqueous solutions. Five of these eight criticality accidents involved highly enriched uranium and three involved plutonium. Four of these criticality accidents occurred in heavily shielded facilities designed for processing irradiated fuel; in these cases, personnel exposures were low due to the inherent shielding.

8.4.2 Accidental Criticality Consequences to Personnel

Overall, the consequences from the eight accidents have been two deaths, nineteen significant overexposures to radiation, minimal equipment damage and negligible loss of fissile material. None of these incidents resulted in significant exposure to the general public.

8.4.3 Causes

It is interesting that all reported criticality accidents and incidents were the result of one or a combination of the following elements:

a. Equipment difficulties/malfunctions

b. Inadequacy of procedures

c. Violation of procedures.

All of these incidents involved fissionable material enriched to > 5%. In several cases, the criticality excursions followed changes in facility production, utilizing equipment designed for lower enrichment material. Finally, from industry experience it appears that solutions present a greater NCS control concern than solid or metallic uranium systems.

8.4.4 Lessons Learned

The radiological consequences of the incidents in unshielded facilities have been limited by immediate evacuation of personnel alerted by alarms. Especially for the prolonged criticality event, evacuation may be credited with saving lives. Hence, it is a good practice to utilize radiation-initiated alarms where there is significant potential for an accidental criticality.

The two fatalities involved persons within 1 meter of an excursion. Significant exposures were received by others at distances extending to 15 meters (approximately 50 ft). Figure 8-1 generalizes this observation. Personnel doses normalized to a "reference excursion of 10^{17} fissions and crudely adjusted to exposure times of 15 seconds" correlate roughly to source distances as shown in the figure. The center band corresponds to LD 50/30, the range of doses that would be expected to be lethal to 50% of a general population within 30 days of exposure.

Based on Figure 8-1, it may be concluded that within 3 meters of a 10^{17} fission excursion, lethal exposures may be expected; at 20 meters, the expected exposure is 25 rad, a level at which effects are generally not medically detectable. These distances are comparable to those considered dangerous for moderate chemical explosions.

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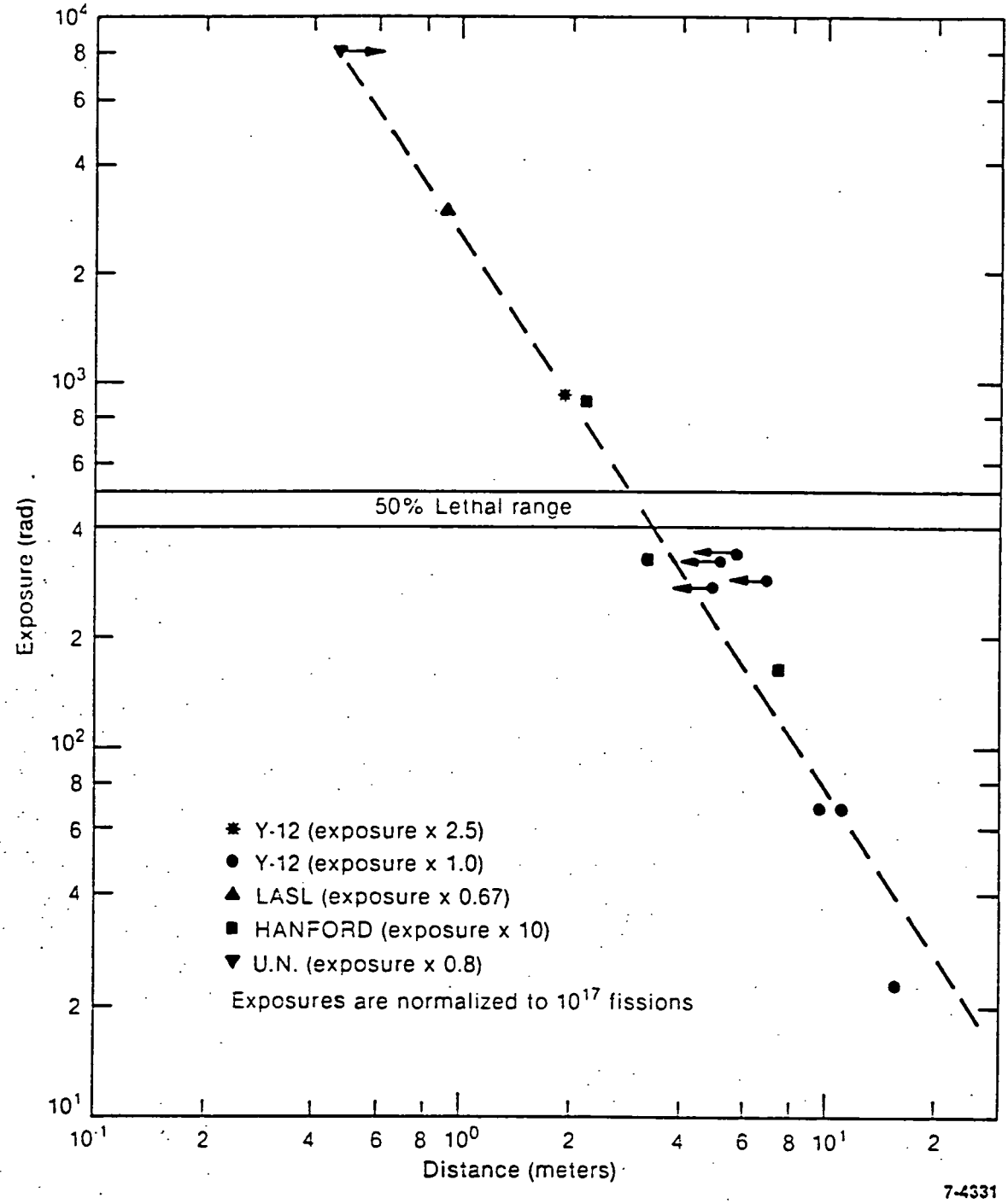

Figure 8-1. Approximate correlation of exposure with distance from a solution excusion of 10^{17} fissions. Arrows appears where it is believed that available estimates are displaced from the probable values.

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SECTION 9

WASTE MANAGEMENT

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SECTION 9

WASTE MANAGEMENT

The purpose of this section is to present a very general discussion of waste management principles for health physics personnel. Waste management is a complex topic and one filled with current controversy, and this section is not intended to present or discuss the philosophy or practices. Since radiation safety personnel are frequently in a unique position to have a marked effect on the reduction of volume and quantities of waste generated, a general understanding of waste management needs is important. Specific health physics procedures for in-plant control of radiological problems from uranium waste are contained in previous sections.

9.1 Potentially Contaminated Wastes

Wastes are generated within a plant or facility as a consequence of creating the uranium product(s) for which the plant was designed. Uranium may be entrained in the air, contaminate equipment, materials or other scrap, and may be contained in low concentrations in liquid wastes and effluents.

Wastes resulting from operation of a uranium facility may include both radioactive, nonradioactive, and mixed materials in the form of liquids and gaseous effluent or solids requiring disposal.

Uranium recovery operations and processes are an operational feature of most major facilities handling large quantities of material for at least two major purposes, i.e., to salvage valuable material and to reduce effluent concentrations and volumes to acceptable levels.

The facility and all waste systems must be designed to minimize wastes which result in the release of radioactive materials to the environs, during both normal plant operation and the occurrence of a Design Basic

Accident (DBA) meeting the regulatory limits as well as, as low as reasonably achievable. Waste systems include retention containers, cleanup systems for liquids and solids, and analytical equipment.

9.1.1 Solid Waste

Facilities should provide for the safe collection, packaging, inventory, storage, and transportation of solid waste that is potentially contaminated with radioactivity. Adequate space for sorting and temporary storage of solid waste, equipment for assay of the waste, and facilities for volume reduction appropriate to the types and quantities of solid waste expected are necessary. All packages containing potentially contaminated solid waste should be appropriately monitored, both before being moved to temporary storage locations and before being loaded for transport to a disposal site.

9.1.2 Liquid Waste

Industrial Wastes. Industrial wastes such as discharge from mop sinks, overflow from positive pressure circulating waste systems, and process steam condensate (if existing) should be analyzed, collected and transferred to a liquid waste treatment plant or similar type treatment area if mandated by the chemical analysis. Provisions should be made for continuous monitoring and recording of radioactivity, flow volume, and pH. The radioactivity monitor should have an alarm located in the liquid waste treatment plant or area. Consideration should be given to retention systems.

Process Wastes. Liquid process wastes should be collected and monitored near the source of generation before batch transfer through appropriate pipelines or tank transfer to a liquid waste treatment plant or area. These wastes should be individually collected at the facility in storage tanks that are equipped with stirrers, sampling and volume measuring devices, and transfer systems. Waste storage tanks and transfer

lines should be designed and constructed so that they are fully inspectable, and that any leakage should be detected and contained before it reaches the environment.

Sanitary Waste. Sanitary wastes include the nonradioactive wastes usually found at a facility, e.g., discharges from noncontaminated chemical laboratories, showers, and lavatories. The sanitary waste system and the uranium handling area should not be connected. Sanitary sewers should discharge into an onsite, approved sanitary-sewage treatment system. Current federal, state, and local codes regarding the discharge of sanitary wastes should be met.

9.2 Design

9.2.1 Objectives

A principle design objective for process systems is to minimize production of wastes at the source. One of the primary design objectives of any Waste Management Program is to provide facilities and equipment to handle the wastes generated and further reduce the amounts, volume, etc. of waste. Volume reduction facilities and equipment for liquid and solid wastes are required as well as air filtration provided to reduce the concentration of contaminants in the air effluent.

9.2.2 Effluents

Airborne and liquid effluent to the uncontrolled environs are of particular concern when societal emphasis on environmental pollution control is high. Process and monitoring equipment are critical to maintaining an acceptable operating posture.

Effluent (both radioactive and nonradioactive) from the uranium handling facility include air and other gaseous exhausts and liquid wastes. The contamination in the effluents should be as low as reasonably

achievable, commensurate with latest accepted technology at the time of design. Emphasis should be placed on reducing total quantities of effluents (both radioactive and nonradioactive) released to the environment. Filter systems should be designed so that the effluent concentrations of uranium should not exceed the derived air concentration (DAC), in Chapter XI (requirements for Radiation protection) of DOE 5480.1A, for uncontrolled areas measured at the point of discharge (e.g., exhaust ducts and stacks) during normal operations. Consideration should be given to recirculation systems for process ventilation where feasible. Provisions should be made for retention systems for liquid effluents. All effluents streams should be sampled or monitored as appropriate to assure accurate measurements of all releases under normal and DBA conditions.

9.3 Treatment

9.3.1 Airborne Waste

Ventilation control systems within a plant are designed to move air from outside "clean" areas to process areas and then to air cleanup systems. Occupied area off gas systems are also vented to the atmosphere and may have clean up systems of their own. Process off gas treatment systems consist of any or all of the following:

Wet Scrubbers are generally used in dusty process off gas situations in which large amounts of uranium are present. The scrubbers are capable of removing and processing large quantities and serve as a prefilter to the remaining cleanup units.

Prefilter systems other than the wet scrubber are bag filters or other rough/course filters. The prefilters are used to remove significant quantities of particulate material from the air off gas and are generally placed before High Efficiency filters (HEPA) in order to extend the life of the more expensive filters.

HEPA filters generally are the final filter in the process off gas and serve to reduce the particulate effluent to insignificant or permissible levels. They may be placed in series to provide the required filtering efficiency.

9.3.2 Liquid Waste

Liquid wastes are of equal concern to that of airborne wastes since the effluent is generally released to the environment and becomes available for dispersion, reconcentration in food chains, and otherwise result in population exposure potential. In the case of liquid wastes the concern for chemical pollutants is generally of equal concern to that of radiological contaminants. In any event the process wastes are generally collected in hold tanks, monitored, processed or treated, and released.

Hold tanks are used to collect liquid effluent prior to release in order that analyses can be performed to establish that the concentrations or total quantities are below permissible levels prior to release. The liquid can be processed or treated to remove radioactive material or neutralize chemicals.

Settling basins are frequently used to provide a means of reducing effluents further prior to release to offsite areas.

Filtration is a simple method of removing insoluble particulate materials entrained in the liquid streams. For some processes it is an effective and inexpensive method. The particulate material collected must be periodically removed and treated as solid waste.

Ion exchange is a clean up system for removing soluble ions from the liquid streams by collecting the material on resin columns. The contaminants must be periodically removed by a regeneration process and the materials processed, concentrated, etc., or by replacing the resin completely and treating it as solid waste.

Conversion to solid forms is a function of nearly all the processes mentioned which converts the materials removed from the liquid and airborne waste streams to more manageable forms for handling and permanent disposal.

9.3.3 Solid Wastes

Solid wastes come from a variety of sources in the plant from machining chips to contaminated clothing. The solid wastes should be concentrated (if possible and/or practical), packaged, and stored on the plant site for an interim time period prior to permanent disposal. Careful documentation is necessary to establish: a) quantities and nature of the waste being disposed, and b) compliance with Resource Conservation and Recovery Act (RCRA) and other disposal and shipping/handling requirements.

On site volume reduction facilities such as incinerators, compactors, or chemical leach from metallic waste sources, can result in volume reduction in the range of 1 to 400 or more.

9.4 Monitoring

Monitoring the airborne effluents is an important aspect of control and documentation. Monitoring should be done in the stack at the discharge point and at the boundary of the uncontrolled area. In addition, total activity discharged and total mass of uranium discharged should be determined and documented to ensure that concentration requirements are not exceeded.

Monitors are of two general types: continuous and passive.

Continuous monitors are constructed with a radiation detector which is placed in a shielded container such that it "views" the activity as it is being collected on a filter from a sample of the stack effluent. The continuous level of radioactivity on the filter is recorded and set up in such a way that preset levels trigger an alarm. This type of monitor is

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less sensitive but provides an alarm in the event of mishap or equipment failure in time to take effective mitigating action.

Passive monitors consist of a continuous (isokinetic, if practical) sample collected of the effluent in the stack. The filter is periodically removed and submitted to radiological and/or chemical analyses. The sensitivity or level of detection is lower for passive sampling systems than for continuous stack samplers, and provide after-the-fact information only.

9.4.1 Air and Gaseous Effluents

All air and other gaseous effluents from confinement areas should be exhausted through a ventilation system designed to remove particulates. The airborne effluent should comply with DOE 5480.1A, Chapters XI and XII. All exhaust ducts (or stacks) that may contain fissile contaminants should be provided with two monitoring systems. One should be of the continuous type (CAMs) and the other a passive sampler. These systems may be a combination unit. The probes for sampling purposes should be designed for isokinetic sampling and located according to good industrial hygiene practices. The design of effluent monitoring systems should appropriately meet the requirements of ANSI N42.18, "Specification and Performance of Onsite Instrumentation for Continuously Monitoring Radioactive Effluents." Nuclear criticality safety should be considered in the design of equipment used to treat and clean up radioactive gaseous effluents.

9.4.2 Liquid Effluents

Emphasis should be placed on reducing total quantities of liquid effluents released to the environment. In addition, the processing of liquid effluents should comply with Chapter XII, "Prevention, Control, and Abatement of Environmental Pollution" of DOE 5480.1B. The contamination in the effluents should be ALARA, commensurate with the latest accepted technology at the time of design. All effluent streams should be sampled

or monitored, as appropriate, to ensure accurate measurement of all releases under normal and DBA conditions. The design of effluent monitoring systems should appropriately meet the requirements of ANSI N42.18, "Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactive Effluents."

9.4.3 Water Collection System

Collection systems should be considered and provided where practical for water runoff from nuclear facilities containing radioactive material, such as from firefighting activities. Nuclear criticality, confinement, sampling, volume determination, and retrievability of liquids and solids should be required in the design of collection systems. The size of the collection system for firefighting water should be based on the maximum amount of water which would be collected in fighting the Design Basis Fire (DBF). The configuration of the system components should be based on conservative assumptions as to the concentration of fissile material which might collect in the system. Recirculating systems should also be considered when there is no possibility of contamination.

For special facilities that process, handle, or store uranium, the water runoff collection system should be designed with the following nuclear criticality safety considerations: (1) the maximum uranium mass loading that could be in the runoff system; (2) the most disadvantageous uranium concentrations, particle size, and uranium dispersion in the water slurry; and (3) the change in concentration of uranium and geometric configuration of the slurry as the uranium settles out of the water.

9.5 Waste Reduction

Field health physics personnel are routinely present at the work sites and are in positions to have a major impact on the control of waste generation.

9.5.1 Source Control

Much of the radioactive waste is generated as a result of materials used in contaminated areas. Identification of unnecessary materials and keeping those materials out of the contaminated area is a major source of potential waste reduction.

Another element in source control is in identifying the types of waste generated so as to reduce or eliminate the source. For example, utilization of washable cloth bags can reduce the amount of contaminated plastic waste generated.

Perhaps the most important method of waste reduction/source control is to minimize contaminated areas or sources. Immediate cleanup of contamination spread and reduction of contaminated areas - pushing the contamination back to the source - reduces the number of sets of personnel protective clothing, tools, etc., that are used in these areas.

9.5.2 Waste Segregation

The tendency at most facilities is to dispose of potentially contaminated material in contaminated waste receptacles. It has been shown that the bulk of "hot" waste has essentially no activity and can be disposed of in the cold waste. Significant volume reduction can be expected through even a rough waste segregation program.

9.5.3 Training

As always, people are the key to any effective program. Effective waste management and reduction are dependent upon personnel who understand the need and techniques of waste control and reduction and are motivated to consistently practice the fundamentals of control and reduction to assure effective implementation. Specific training of radiation workers is an essential element of waste control.

9.5.4 Decontamination

In many situations disposal is a practice in preference to decontamination. Material selection to facilitate decontamination and a disciplined program to detect and remove low-level contamination will save both the cost of the material replacement as well as handling, transportation, and disposal/burial cost.

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SECTION 10

EMERGENCY PREPAREDNESS

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SECTION 10

EMERGENCY PREPAREDNESS

10.1 Introduction

The purpose of emergency response planning is to prepare in advance for protecting public and employee health and safety and to minimize adverse effects to the facility if an accident occurs. To fulfill this need, the Department of Energy, in DOE Order 5500.3, REACTOR AND NONREACTOR NUCLEAR FACILITY EMERGENCY PLANNING, PREPAREDNESS AND RESPONSE PROGRAM FOR DEPARTMENT OF ENERGY OPERATIONS (DOE 1981a), has required operators of its reactor and nonreactor nuclear facilities to develop an emergency planning, preparedness, and response capability that meets the prescribed requirements. Additional DOE requirements regarding emergency preparedness are found in DOE 5500.1A, Emergency Management System (DOE 1987a) and DOE N5500.2, Emergency Preparedness Program and Notification Systems (DOE 1987b).

Emergency planning and preparedness usually involves the development of an emergency plan document and a set of implementing procedures. The emergency plan document describes the overall emergency organization, designates responsibilities, and identifies resources available to respond to emergencies. Implementing procedures provide specific instructions to response personnel for carrying out emergency actions.

As defined in DOE 5500.1A, there is a hierarchical system of DOE emergency plans to include DOE Headquarters (HQ), field offices, and facility emergency plans. Each DOE field office (sometimes referred to as a field element) shall prepare and maintain site-specific radiological emergency response plans for DOE facilities under their contractual responsibility. The facility (sometimes referred to as a DOE contractor) shall prepare and maintain facility-specific emergency response plans that are compatible with those of its field office.

DOE has adopted two radiological emergency planning zones (EPZs) based upon the plume exposure pathway and the ingestion exposure pathway (DOE 1981a). According to DOE 5500.3, the establishment of emergency planning zones in terms of distance around DOE facilities is site specific and developed by the field offices with DOE-HQ concurrence. These distances are determined by comparing doses resulting from potential accident scenarios to protective action guidelines. Because some DOE facilities are located on large federal reservations, the EPZs for some uranium facilities might not include any members of the general public.

This section addresses the requirements and good practices pertaining to emergency planning at uranium facilities. Although specific to uranium facilities, much of the guidance provided may apply to any facility handling radioactive or other toxic and hazardous materials.

10.2 Emergency Plan

10.2.1 Organization and Assignment of Responsibility

The emergency plan shall clearly define the onsite emergency organization of staff personnel for all shifts in order to maintain continuous (24 hour per day) response capability. The plan shall designate the individual emergency coordinator, on site and on shift, that at all times has the responsibility and authority to immediately initiate emergency response actions. These actions may include recommending protective measures to offsite authorities. A line of succession for the emergency coordinator should be identified.

Organizations that support the emergency coordinator and the command and control center should be described as to specific function, interrelationships with other support groups, and the individual and alternates that supervise each group. Typical support functions should include:

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a. technical support (accident assessment)

b. operational support (repair/corrective action)

c. health and safety support (radiological assessment and protection)

d. public information

e. security

f. fire control

g. medical services and rescue

h. logistics (personnel and material procurement).

Organizational charts showing primary responsibilities and reporting lines are useful in assuring clear delineation of functions.

Emergency plans shall identify the local, state, and federal organizations which are part of the overall response organizations for the emergency planning zones. The involvement, responsibilities, and authorities of each group must be specified as well as the methods of contact. Individuals and alternates assigned to supervise each response group should be specified by job title. Emergency duty assignments should correspond as closely as possible to normal duties. The description of the emergency coordinators duties must specify those responsibilities which may not be delegated. Among those are:

(1) the decision to declare an emergency,

(2) notification of offsite authorities of emergency declarations, and

(3) recommendations for protective actions to offsite authorities.

10.2.2 Emergency Response Support and Resources

DOE facilities may rely on offsite organizations for significant emergency response capabilities and resources or use them to augment site response capability. Examples of support that may be provided from offsite organizations are:

a. fire fighting

b. law enforcement

c. medical services

d. radiological monitoring

e. vendor technical support.

In addition to the support that may be provided by local, county and state groups, other DOE facilities and other federal agencies may be available to provide specialized services and support.

The emergency plan shall identify the organizations expected to provide emergency response support, give a brief description of their capability, and specify the items of agreement. These agreements should be formalized in a Memorandum of Understanding.

10.2.3 Emergency Response Levels

To be consistent with other government agencies, as defined in DOE N5500.2 (DOE 9871), DOE has adopted the following four levels of emergency situations to indicate the severity of an accident:

(1) Unusual Event. an event in progress or having occurred which normally would not constitute an emergency but which indicates a potential exists for significant release of radioactive material. Activation of offsite response organizations is not expected. Emergency response actions are limited to onsite areas.

(2) Alert. An event in progress or having occurred which involves an actual or potential substantial reduction of the level of nuclear safety of the facility. Limited offsite releases of radioactive material may occur. The purpose of an alert level is to assure that onsite and offsite emergency response personnel are properly advised and available for activation if the situation becomes more serious, to initiate and perform confirmatory radiation monitoring as required, and to assure appropriate notification of emergency conditions to the responsible organizations within DOE.

(3) Site Emergency. An event in progress or having occurred which involves actual or likely major failures of facility functions which are needed for the protection of onsite personnel, the public health and safety, and the environment. Releases offsite of radioactive material, not exceeding Protective Response Recommendations (PRR's), are likely or are occurring. The purpose of the site emergency level is to assure that emergency control centers are manned, appropriate monitoring teams are dispatched, personnel required for determining onsite protective measures are at duty stations, predetermined protective measures for onsite personnel are initiated and to provide current information to DOE and consultation with offsite officials and organizations.

(4) General Emergency. An event in progress or having occurred which involves actual or imminent substantial reduction of facility safety. Releases offsite are occurring or are expected to occur and exceed PRR's. The purpose of the general emergency level is

to initiate predetermined protective actions for onsite personnel, the public health and safety, and the environment, provide continuous assessment of emergency conditions and exchange of information both onsite and offsite. Declaration of a general emergency will initiate major activation of DOE-wide resources required to effectively mitigate the consequences of emergency conditions and assure the protection of onsite personnel, the public health and safety, and the environment to the extent possible.

These emergency response levels permit escalation and de-escalation from one to the next level depending on present and anticipated plant conditions. The conditions which would determine the classification of an emergency level are site and plant specific. These emergency action levels must be developed and formalized in procedures for use in guiding emergency response personnel.

For uranium facilities, special evaluations should be made in establishing emergency action levels and developing criteria for determining the level of emergency classification. Because of the low specific activity of uranium, few credible accident scenarios produce offsite radiation doses that exceed or even approach Protective Response Recommendation guidelines. However, in many processes involving uranium, the uranium may be in chemical compounds which are extremely toxic or which form, on contact with air, toxic gasses. The emergency action levels may need to be based on the chemical toxicity criteria. In addition, local public relations concerns may dictate emergency response beyond that dictated solely by radiation dose concerns.

10.2.4 Notification Methods and Procedures

Timely and efficient notification of emergency organizations and support groups is vital to an effective emergency preparedness program. Formal procedures and reliable methods of notifying personnel, both onsite

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and offsite, of an event must be developed and described in the emergency plan. Notification of onsite personnel may be accomplished by plant page, plant alarm, telephone, portable radio, pager, or combinations of these methods. Typically, a primary method is established and other means are used for backup. Offsite agencies may be notified by dedicated telephone line, normal telephone line, short-wave radio, teletype machine or combinations of all of these methods. Because of the importance of accurate data and message transmission, common terminology and a standard format for information should be established. This is especially important for transmitting emergency classifications, release data, and protective action recommendations.

10.2.5 Emergency Communications

The emergency plan shall assure that prompt and efficient communications have been provided among the response organizations, emergency personnel, and federal, state, and local agencies. A primary communications system and at least one backup system should be provided and the systems should be independent of each other. Separate power sources should be used and one system should be supplied with emergency power. Industry experience shows that dedicated communication systems between emergency organizations can be very valuable for reliable and continuous information exchange. Routine periodic testing of communications devices is necessary to ensure availability and function in an emergency. Frequency of testing will be determined by the equipment used and should be specified in the emergency plan. Documentation of testing must be maintained.

10.2.6 Emergency Facilities and Equipment

Emergency facilities and equipment shall be provided to support an adequate emergency response. The emergency facilities and equipment required for a site will depend on the type and quantity of hazardous materials at the site and the operations performed. While specific site needs will vary, essentially four functions must be included. These are:

(1) the control room function

(2) technical support function

(3) operational support function

(4) near-site support function.

The control room function, whether a control room or a control area in-plant, is the primary location for efforts to control the processes and establish safe shutdown conditions. The control location may also be the central location for communications and information on plant status. Consideration should be given to assuring habitability of the control area during an emergency so that appropriate control and mitigating actions can be maintained throughout the duration of the emergency.

The technical support function provides the engineering and technical support to the emergency response effort. The technical support function should be manned by the technical experts for the facility processes and they should have ready access to all information concerning plant operations, maintenance and design which would assist them in diagnosing and correcting plant problems. This function should assume the majority of the emergency response duties from the control room to permit the control room personnel to concentrate on problem correction and mitigation of the accident. To permit necessary consultation and interaction between operators and engineers, the technical support function should be physically located as close as possible to the control room.

The operational support function provides the staff that may be required for repair and corrective action to prevent or mitigate an accident. This function also provides the radiological monitoring efforts necessary for in-plant and onsite activities. The operational support function should be located in an area not affected by the event and with access to any tools and equipment needed.

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The near-site support facility is the offsite interface for emergency response. The purpose of this function is to communicate with offsite agencies, arrange for offsite support and assistance, provide a command center for contractor and DOE personnel, and perform assessments of actual and potential offsite releases and their consequences.

Equipment shall be provided for emergency response personnel. For special purposes, dedicated emergency kits may be necessary. Emergency kits should be considered for offsite monitoring personnel, rescue teams, and fire fighting personnel. Contents of these kits may include radiation survey instruments, air sampling equipment, protective clothing, respiratory protection devices, radios for communication, dosimetry devices, procedures, and recording supplies. Emergency kits should be sealed to minimize tampering and routine inventory and inspections conducted. The calibration frequency of instruments and the service life of components such as batteries should be considered in establishing the frequency of emergency kit inspections.

10.2.7 Accident Assessment

Methods and equipment for monitoring and assessing actual or potential offsite consequences of a radiological emergency shall be available and operational. Actual releases are best determined by installed effluent monitoring systems if they are available and the releases are from monitored release points. Releases from unmonitored locations may be estimated from inventory data, nature of the event, and physical characteristics of the material released. Effluent monitoring data and release estimates should be verified and possibly modified by field measurements. Projections of potential releases may be based on plant parameters and failed systems. Special attention should be given to the chemical toxicity of the released uranium and any accompanying toxic gasses.

Appropriate codes and models should be available to calculate actual and potential offsite doses. Real time meteorology data should be available along with measured or estimated source terms and location and height of releases. Additional guidance may be found in IAEA Safety Series 152 (Evaluation of Radiation Emergencies and Accidents, 1974), IAEA Safety Series 86 (Techniques and Decision Making in the Assessment of Offsite Consequences of an Accident in a Nuclear Facility, 1987), and in Dose Projection Considerations for Emergency Conditions at Nuclear Power Plants (Stoetzel et al. 1983).

Because characterization of a uranium release can be difficult in many situations, especially at an unmonitored release point, it is recommended that field monitoring be implemented. Field monitoring data can be valuable in verifying that a release has occurred and in confirming the accuracy of source-term estimates. Field data may also confirm or invalidate the need for protective actions and for changing emergency response levels.

10.2.8 Protective Response

Protective responses taken to avoid or minimize personnel and public exposures to a uranium release should concentrate primarily on minimizing the inhalation or ingestion of materials.

For onsite, three methods of protection are available. The first is to evacuate personnel from the affected areas and any areas with a high potential for contamination. Advanced planning and periodic training drills are necessary to maintain this capability. Transportation (buses, etc.) must be available promptly and preselected routes, if appropriate, should be used to evacuate personnel in a timely manner. Meteorological conditions (wind direction and speed) must be communicated to the person in charge of the evacuation so that personnel may be evacuated without entering the plume.

A second method of protecting onsite personnel is to move them into a protected ventilation zone. Onsite facilities, such as the Emergency Control Station (ECS), should be designed to maintain safe habitability during postulated accident conditions. However, care should be taken not to overcrowd these facilities with nonessential personnel.

The third method of protecting onsite personnel is the use of protective clothing and respiratory protection devices. Although respiratory protection should not be relied on for sustained protection, the devices should be used as a precaution and for short time periods during transit between facilities or for entering/exiting the site during an accident. The choice of respiratory protection (self-containment, supplied air, full-face, etc.) should be decided based on the protection factor needed, degree of freedom of movement needed, availability of respirators, and training of personnel.

Protective responses for offsite areas are implemented by local authorities based upon recommendations from the field office. The responses usually involve two methods, the details of which are agreed upon by the site operator and the local authorities in the early planning stages. The first is protective sheltering. If sheltering is recommended, residents in the affected areas should shut down their ventilation systems, seal their homes and occupied structures as well as possible, and remain inside those structures until instructed to do otherwise. This method gives some protection from airborne contaminants, especially in the case of a quickly passing plume.

The second option is evacuation. This should be recommended only when there is a potential for release and there is time for an effective evacuation. Local authorities have the responsibility for carrying out this action based upon recommendations from the field office. The field offices should be aware of the details of evacuation plans, especially the routes selected and the time to complete various evacuation scenarios. The DOE field office and DOE contractor should also be aware of the state and local authorities' protective action decision-making process.

10.2.9 Radiological Exposure Control

The control of radiological exposures must be maintained, even during the course of an accident. Because normal radiological controls may not be sufficient during abnormal operations, additional controls should be designed for implementation at the appropriate time. This section discusses some of these additional controls.

The emergency plan should establish onsite emergency exposure guidelines that are consistent with DOE 5480.11 and EPA emergency worker and lifesaving activity protective action guidelines, as defined in EPA 520/1-75-001 (EPA 1980). The emergency plan should also include emergency exposure guidelines for performing assessment actions, providing first aid, performing personnel decontamination, providing ambulance service, and providing medical treatment.

Normal exposure controls should not prevent efforts to mitigate the consequences of an accident. Therefore, a responsible person with the authority to approve emergency radiation exposures in excess of established limits should be onsite at all times. This responsibility usually lies with the emergency director after consultation with the most senior health physicist available.

To achieve dose control for emergency workers, personnel dose information shall be available and maintained current. The capability to process dosimeters and have the information promptly available on a continuous basis should exist. A reliable dosimeter distribution system and record system should also be available.

For most uranium facilities, high levels (>rem/hr fields) of external radiation exposures in emergency events could only result from an accidental criticality. In a criticality event, special precautions must be exercised to assure that radiation doses from neutrons are considered, that the reaction has ceased, and that emergency actions do not re-initiate the criticality.

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10.2.10 Medical and Health Support

Medical services for personnel injured during an emergency must be provided. For most uranium facilities, medical assistance takes precedence in almost all instances over contamination control. However, as far as is possible, the spread of uranium contamination to open wounds should be avoided. Also, precautions should be taken to minimize the inhalation and ingestion of contamination and the spread of contamination to medical or often off site facilities.

10.2.11 Recovery and Reentry

After emergency conditions have stabilized and the plant is in a moderately safe shutdown status, recovery of the facility may begin. This effort consists of work to return the plant to its preaccident condition. While the plant may be in a safe shutdown status, extra precautions may be necessary during restoration work because of potential or actual damage to safety systems, process equipment, and structures. Detailed planning should be performed prior to reentry to ensure that adequate precautions and controls are established to protect the health and safety of workers.

10.2.12 Maintenance of Emergency Preparedness

To maintain an adequate emergency response capability, periodic training in emergency duties is required. Training is usually accomplished with a combination of formal classroom lecture and actual performance of duties in a drill.

All personnel with emergency duties should be trained at intervals specified in the emergency plan. The frequency of training depends on the particular function, but, in general, should be at least annually. Personnel of interest are management, those with specialized emergency duties, and a sufficient number of back-up personnel for each position. Training should be performed by the facility or DOE field office emergency

preparedness coordinator or by the facility's training department staff. The onsite training program should include the following topics:

a. purpose of emergency planning

b. emergency organization

c. interrelationships between organizations

d. training on specific duties and procedures

e. training on protective actions

f. dose assessment.

In addition, offsite personnel with the potential for providing aid during an emergency should receive training on the same periodic basis. Some examples are the following:

a. state, local, and municipal agencies (law enforcement, fire protection, and public health)

b. state, local, and municipal government officials

c. private medical doctors, hospital staffs, and the staffs of emergency rescue organizations and ambulance services

d. volunteer fire department personnel

e. military personnel (where appropriate)

f. private industry emergency services personnel.

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The training for offsite groups should include

a. some plant specifics (what are the processes, potential source terms, type of possible accidents, etc.)

b. discussion of emergency response levels and emergency planning zones

c. site organization and offsite organization structure and responsibilities

d. communications and emergency messages.

Facility employees with no emergency responsibilities should receive annual training to familiarize them with the general contents of the emergency plan and appropriate response actions. This training could be done in conjunction with other training courses such as radiation worker training.

Participation in periodic table top drills, drills, and exercises enables personnel to become familiar with their assigned duties and helps identify problems with procedures, training, and equipment. The emergency plan and procedures for each facility should be tested as often as required to ensure that they are adequate. A suggested frequency for drills and exercises is given in IAEA Safety Series No. 73 (IAEA 1985). To test emergency response, the emergency exercises should present the plant operators with a realistic accident scenario in a realistic time frame. Offsite support organizations should be periodically included in the drills and exercises.

Preparation of scenarios and conduct of drills and exercises should be performed by personnel who are not assigned emergency response functions and who are familiar with the facility, its operation, and the emergency plans. Organizations that may perform this function are quality assurance,

training, or personnel specifically assigned by management. The DOE or DOE contractor emergency preparedness coordinator should be assigned the overall responsibility for development, conduct, and follow-up to emergency drills and exercises.

Emergency exercises usually have two types of nonparticipants who are crucial to the conduct of the exercise. Persons who are knowledgeable about the scenario and who supply data to the players are called controllers. These persons keep the scenario proceeding on an established timeline and can observe player response when time permits. Personnel assigned only to observe and critique player response are called evaluators.

Immediately after the exercise is terminated, controllers and evaluators should meet to discuss and document the findings. The findings should be discussed with the players soon after the comments are compiled to establish their validity. A final report including observations and recommendations for program improvements should then be issued by the organization conducting the exercise. Report distribution should include each function participating in the drill or exercise and management. An organization should be assigned overall responsibility for tracking and follow-up on corrective actions to ensure their completion and implementation.

Emergency Procedures

Organizational, personnel evacuation and accountability, emergency notification, readiness actions, personnel monitoring and decontamination, medical emergency requirements, emergency equipment and supplies, and re-entry procedures comprise the list of emergency procedures required at uranium facilities. These procedures are discussed in the following subsections. In addition, procedures dealing with in-plant monitoring, offsite radiological monitoring, and dose assessment should be considered.

10.3 Emergency Procedures

10.3.1 Organizational Procedures

Written procedures should be provided that describe the lines of authority and responsibility for the staff assigned to emergency response functions. These procedures should include identification of responsibilities which may not be delegated.

The written procedures should define the communications protocol, list specific communications media to be used, telephone numbers if applicable, and verification steps to be taken. Emergency communication requirements shall be coordinated with appropriate headquarters organizations as required by DOE Order 5320.1, Telecommunications, Spectrum-Dependent Services, (DOE 1980a).

Emergency action levels must be developed and formalized in procedures for use in guiding emergency response personnel. Emergency procedures shall define the facility specific conditions at which each of the four emergency response levels shall be declared. The procedures shall also describe the criteria and process for escalation and de-escalation of the emergency response levels. As stated in paragraph 10.2.3, special criteria may need to be established based on chemical toxicity of the uranium compound and on toxic gasses that may be released with the uranium or formed in reactions with air after a release.

The most likely major radiological accident for a uranium facility is the rupture of a heated cylinder of UF_6. Calculations show that a release of 4800 kg of natural uranium in 5 minutes from a 14-ton cylinder ruptured outdoors could result in an intake at 200 meters of approximately 260 mg of uranium. This uranium intake would cause a dose of 1.2 rem effective dose equivalent assuming the uranium is highly soluble (solubility class D). The EPA recommends considering protective actions for the public if radiation doses would be in the range of 1 to 5 rem.

Thus, a buffer area of slightly more than 200 meters would be adequate to maintain public radiation doses below protective action guidelines.

However, an accident such as this would also result in significant generation of hydrofluoric acid (HF) fumes which are a much greater hazard than the uranium. Evacuation guides for HF releases have been established by the U.S. Department of Transportation. For a large spill of HF, the Department of Transportation (DOT) recommends isolation in all directions to a distance of 150 meters and evacuation in a downwind direction to a distance of 3 km and a width of 2 km (DOT-P5800.2). These distances should be considered in developing protective action guidance for uranium facility emergencies.

10.3.2 Personnel Evacuation and Accountability Procedures

Written procedures shall be developed for determining the conditions under which evacuation is necessary. They shall also include designation and use of staging areas and assembly points, determination of shelter criteria, selection and use of primary and alternate evacuation routes, provisions for traffic control during evacuation and transportation of evacuees, and coordination of evacuation actions with offsite groups and institutions.

Procedures shall be provided for implementation of the personnel accountability system, for assuring the security of classified matter and all source, by-product, and special nuclear material, for the evacuation of disabled persons, and for necessary communications.

10.3.3 Emergency Notification

Written procedures shall be established for receipt, verification, and further dissemination of emergency information to management, cadres, emergency duty personnel, offsite groups, and others, as appropriate, depending upon the type and severity of the emergency.

Procedures shall be developed and coordinated with appropriate agencies for handling of alerting notifications received via the National Warning System (NAWAS).

10.3.4 Readiness Actions

Procedures shall be developed for readiness actions including response to emergency conditions, safeguards and security alerts, and emergency information. The procedures shall provide for automatically placing in effect an appropriate safeguards and security alert upon receipt of notification of an emergency condition.

10.3.5 Personnel Monitoring and Decontamination

Procedures shall be established to monitor personnel exposed to toxic and radioactive materials and to decontaminate personnel and equipment. In so far as possible, normal facility procedures should be used with provisions for the abnormal conditions so that personnel are familiar with their implementation. Procedures and actions expected of offsite personnel, such as medical staff, should be developed in cooperation with the staff involved and provided to them.

Procedures should provide for authorizing radiation exposures in excess of the normal limits and for the control and recording of doses received.

10.3.6 Medical Emergency Requirements

Procedures shall be established for search and rescue of missing or injured persons. For uranium facilities, special considerations must be given to protecting search and rescue personnel from exposure to toxic gasses. Entries into accident areas may require supplied air or self-contained breathing devices because of the chemical toxicity of some

uranium compounds and/or accompanying toxic gasses. Some by-products of uranium material releases, such as hydrogen fluoride, may cause burns when in contact with skin. Procedures should include instructions for use of additional protective equipment, as necessary, for whole body protection. Greylite protective suits and self-contained breathing apparatus are typically used for emergency entry into UF_6 release areas.

Procedures should include provisions for transport of injured personnel to offsite hospitals and the contamination controls necessary for protection of transport personnel and medical staff and facilities.

10.3.7 Emergency Equipment and Supplies

Procedures shall be established which specify the location of emergency equipment and supplies and the system for assuring that they are available when needed. The procedures should include a listing of the contents of emergency kits and the inventory responsibility and frequency. A checklist of contents and procedures for their use, if necessary, should be included in each kit. The frequency of inventory of emergency kits should consider the calibration frequency required for instruments and the storage life of limited life components.

10.3.8 Reentry Procedures

Reentry into a facility after an emergency shall be controlled by written procedures. The procedures should define the authority and responsibility for decisions to reenter. Methodology for determining accessibility of plant areas, exposure guides for toxic materials and radiation exposure, and technical and safety representatives available for consultation should be established

Special air sampling may be needed prior to reentry to determine the respiratory protection required. Because of interference from particulate daughter products from radon/thoron on filter air samples, impact head type high volume air samplers may be necessary for timely determination of airborne uranium concentrations.

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SECTION 11

DECONTAMINATION AND DECOMMISSIONING

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SECTION 11

DECONTAMINATION AND DECOMMISSIONING

Decontamination and decommissioning typically describes the activities, at the end of useful life of a facility, to restore the facility to a non-contaminated status and permit its unrestricted use. In this manual, the terms will be considered separately. Decontamination techniques applicable during operation are discussed first, and end of life activities are described in the decommissioning section. It is recognized that many decontamination techniques are also used in the decommissioning process.

11.1 Decontamination

11.1.1 Surface Decontamination

Uranium handled in most facilities is a very low specific activity material in the normal range of enrichment (<10% U-235). Quantities of uranium that may constitute a significant surface contamination source are generally visible on hard surfaces of contrasting colors. Consequently, proper facility design and material selection, and good housekeeping practices are effective techniques for maintaining surface uranium contamination at an acceptable level in working areas at many facilities. Such practices, of course, do not eliminate the need for periodic surveys and trend analyses, especially where decay products may accumulate.

Decontamination of surface areas may be as simple as water hosing of floors, washing with detergent and water, or wiping with household dust cloths. Waste material generated from decontamination activities (water, wipe material, etc.) must be contained and disposed of as radioactive waste. A common practice for routine contamination control for uranium is periodic floor maintenance in which the floor is scrubbed with a commercial cleaning machine and a sealer coat (wax) installed. For some locations,

vacuuming of the surface may be appropriate. HEPA filtered vacuum systems are required, if vacuuming is used, to prevent airborne radioactivity from the vacuum exhaust.

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

Depending upon the physical and chemical form of the uranium and the type of surface, uranium may become imbedded in the surface. Removal of embedded material may require physical abrasion, such as scabbling, grinding, sand blasting, or chipping, or may be accomplished using chemical etching techniques. If the surface is porous, complete replacement could be necessary. The use of high pressure water, hydroblasting, has been quite successful for metal surfaces and concrete.

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

11.1.2 Personnel Decontamination

Washing with soap and water is the most successful personnel decontamination technique for uranium contamination. The hands are the most common areas of personal contamination. It may be difficult to decontaminate the cuticles and under the nails. Soft bristle scrub brushes may be used for fingernails and other difficult-to-clean areas as long as the skin barrier is maintained intact. If the contamination has been transferred to the skin with chemical carriers, additional decontamination steps may be required. If the skin barrier has been removed or breached, medical assistance should be sought. Special decontamination agents that may be used for tightly bound contamination include: titanium dioxide

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paste followed by rinsing; a saturated solution of potassium permanganate followed by a rinse using a 5% solution of sodium acid sulfate; and complexing agents such as ethylene diamine tetraceticacid (EDTA) or diethylenetriaminepenta-acetic acid (DTPA). These decontamination agents should be used under the guidance of the medical staff.

Precautions should be taken during decontamination of personnel to assure that loose contamination is prevented from entering body openings. In addition, contaminated wastes should be collected and disposed of as radioactive waste.

11.1.3 Uranium Contamination Detection

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

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

Many of the processes used in uranium facilities may separate and/or concentrate impurities or decay products of uranium. Examples of these processes are uranium recovery from ore, reduction of green salt to metal, UF_6 conversion, casting of metal, and uranium oxidation. Radionuclides of particular importance are protactinium-234M and other decay products and trace impurities such as technitium-99, plutonium-239, and neptunium-237. In addition to the separation processes, some of the decay products of uranium may be selectively accumulated in tank and pipe liner material. Dose rates up to 150 mR per hour attributed to radium accumulation have been measured from neoprene liner material. Dose rates from furnace lids and crucibles have been measured as high as 30 rad per hour.

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

11.2 Decommissioning

When uranium facilities are no longer useful, measures must be taken to assure that doses to the public remain as low as is reasonably achievable and within limits. This may be done in a variety of ways. All (except converting the facility to some other nuclear use) may be termed decommissioning.

The facility may be decontaminated to levels where no controls are needed to limit the exposure of the public. This mode, usually termed "D&D", for decontamination and decommissioning or decontamination and dismantlement, requires the removal of essentially all radioactive (and hazardous) material from the site. This is required if DOE will not

maintain possession of the property.[a] Limits on exposure to the public from such a disposition are the controlling considerations.

A decommissioning mode of "Safstor" is applicable to some nuclear facilities. It involves temporary measures to limit exposure to the public while the radioactive material decays to harmless levels, or to levels that permit dismantling with substantially less occupational radiation exposure. Such a decommissioning mode is not likely to be applicable to uranium facilities unless other, short-half-life radionuclides are also present. The limit of institutional controls is generally considered to be a maximum of approximately 100 years, and no substantial reduction in uranium exposures would be expected in that time. However, a facility that has housed processes that concentrate the uranium decay products might be a candidate for Safstor and delayed decommissioning if a cost benefit analysis showed that a substantial reduction in radiation dose might be expected.

If the site will remain in DOE possession, and is considered suitable for the retention of some radioactive material (the geology, hydrology, population distribution and other environmental factors make it a suitable location to retain waste) then other less comprehensive and less costly methods may be used to assure that population exposures are as low as is reasonably achievable and within limits. Measures such as waste burial, erection of surfaces structures, fencing, etc. may be applicable, as long as they effectively limit dose to the public. Any waste management mode that results in long-half-life radioactive materials remaining on site should also include permanent markers and documents attached to the deed of ownership that detail the location, quantity, and physical and chemical form of the radioactive (and hazardous chemical) materials.

Conversion of a facility for alternate nuclear or other controlled use has sometimes been considered a decommissioning mode, however, it is not

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a. An excess contaminated DOE facility might conceivable be sold to a private party who obtained a DOE license for the facility, but this disposition would be very unusual and outside the scope of this discussion.

truly decommissioned, unless conversion involves removal of all radioactive material. This is seldom possible. Final disposition, when it occurs at the end of the new use, should consider the residual radioactivity on site.

Uranium mill sites and facilities that are included in the major government reservations are expected to be decommissioned in a mode that is reflective of their status as a permanent repository. Isolated metal preparation, fuel fabrication, scrap recovery, and other similar facilities are more likely candidates for unrestricted use.

This guide addresses, to the extent possible, the regulations and practices that are applicable to each of the decommissioning modes, and to the conversion of facilities to alternate uses.

11.2.1 Applicable Standards And Guides

The standards that apply to the decommissioning of a uranium facility include virtually all of those that were applicable during operations, plus some additional ones. The occupational safety and radiation dose limits, radioactive and hazardous chemical disposal regulations, and transportation requirements are unaffected by the activity to which they apply.

Decommissioning will, however, require an evaluation of the effect of the Environmental Policy Act and require a decision regarding the quantities and concentrations of radioactive materials that can be permitted to remain on site.

The National Environmental Policy Act

The decommissioning of a DOE uranium facility will require a determination if the action is a "major or significant government action adversely affecting the environment". If it is determined to be, an Environmental Impact Statement (EIS) is required. If the action does not require a statement either because the possible adverse impacts are insignificant, or because decommissioning was adequately addressed in a

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pre-operational or other EIS, then the decommissioning can proceed in accordance with the information contained in the EIS and with other applicable regulations.

If an EIS is required, one should be prepared in accordance with the applicable DOE guidance as contained in DOE Order 5440.1C, 4-9-85, National Environmental Policy Act, and 40 CFR 1500 to 1508. The EIS will need to address the alternative, which will include retaining radioactive material on site under DOE control, cleaning the site to a level that will be acceptable for unrestricted release and the null or no action alternative of "walk away". The statement will also need to address the amount of material that will remain on site and its impact.

Regulations that specify allowable contamination levels and quantities, such as 10 CFR 20 Appendix B, and those that specify maximum permissible dose to members of the general population, such as 40 CFR 190, will need to be addressed.

Allowable Contamination Levels

The guidelines that are applicable to the decommissioning of nuclear facilities are in a current state of transition, and it is difficult to determine what the final regulations will be. Initially, most uranium facilities were decommissioned to levels specified in NRC Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, which were relatively easily measured. The regulations governing the quantity and concentration of radioactive material that may be controlled under the "general license" (10 CFR 20 Appendix B) may also be relevant.

However, the EPA has been mandated by Congress to develop guidelines that will be applicable to all nuclear facilities as well as to the release of formally contaminated or controlled for unrestricted release. Such guidelines will almost certainly be based on radiation dose to the maximum exposed member of the general population. They may also include guidelines for population dose. Although DOE Order 5480.11 establishes the maximum

dose that may be received by a member of the general population from operation of a DOE facility, the EPA limits are lower. Values of 50, 10, 1 and 0.1 mrem/year are currently being considered by the EPA as the "defacto de minimis" level for the disposal of contaminated material. When, and if, one of these values is established, decommissioned facilities could certainly be released if they met this criteria under any contemplated unrestricted use, and higher levels might be considered acceptable under certain circumstances.

Additional consideration is being given to the acceptable dose from a radioactive waste repository which is not intended to be released to the general public for unrestricted use. Such guidance might apply to a uranium mill tailings pile or to a facility on government owned land that is not intended for unrestricted access. Firm guidance for radiation dose to members of the public from a repository is not yet available, however, one of three criteria will most likely apply: a. the fuel cycle limit of 25 mrem/year from 40 CFR 190; b. the "de minimis" criterion for recycled waste; or c. a criterion that the dose to the most highly exposed member of the population should be no greater than it was from the ore from which the radioactive material was derived.

These criterion require calculation of dose to members of the general population. The scenarios for exposure will have to include all exposure pathways that are credible under the proposed disposition. If the site is part of a closely guarded government reservation, certain pathways such as use of well water directly from the site and ingestion of significant quantities of fruits and vegetables grown on the site may be eliminated. If, however, the site will be released for unrestricted use such scenarios should be considered.

The computer codes used for calculation of dose to the public from decommissioned facilities will include the currently accepted exposure models and site specific or maximum credible parameters for exposure pathways.

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The regulations which address waste transportation and disposal have some bearing on decommissioning. Although they do not determine any safe level that may remain, they do indicate that certain levels are unacceptable if members of the general population will have access to the site. The Code of Federal Regulations, 10 CFR 61 establishes criteria of class A, B and C waste, but does not specify any procedures or disposal possibilities for less than class A waste.

11.2.2 Decommissioning Decision Making and Planning

Decisions to clean a facility to a level that permits unrestricted release are extremely difficult at present (1988) because levels are only established by precedent and in the EIS process. There is no other guidance. Such a decision would greatly facilitate decommissioning.

There will, however, still be important decisions to be made regarding whether the government will maintain control of the site or whether unrestricted use will be permitted. A detailed discussion of the decommissioning planning process is beyond the scope of this document. However, a few general principles that should be kept in mind during construction and operation of the facility are discussed here. The design features of the facility that will facilitate decommissioning are addresses separately in Section 11.2.3.

Background Radiation Levels and Facility Records

The most critical step in decommissioning is begun before the facility becomes operational. Background radiation levels must be established so that the incremental dose occurring from material left on site at the termination of operations can be assessed.

Records maintained during facility operation will be extremely helpful in planning and executing decommissioning. If spills occur, if low level radioactive or potentially radioactive materials are buried on site, or if

other incidents impact decommissioning, they should be considered early in the decommissioning planning.

Criteria

The first step in planning decommissioning is the development of a series of absolute criteria. These will necessarily include such items as compliance with DOE Orders, EPA regulations and other statutes. They may also include commitments to states, landowners, or others, or provisions of the original environmental impact statement.

As these criteria are developed, other high value criteria may also be established. These are likely to include such considerations as maximizing aesthetic and recreational value of the site, performing decommissioning within allocated funds, lowest worker dose, lowest population dose, lowest cost, lowest future surveillance commitment, least impact in case of probable accidents, etc. Depending on the viability of the decommissioning action, the decision making process that has been established, and the level of concern of the public, a notice of a scoping meeting may be given and published in the federal register, and scoping meetings may be held. Similar actions may be taken in determining the applicable criteria and the alternatives that are considered.

Whether or not a formal scoping meeting and EIS are used, it will be necessary to define the options to be considered. Most of the analysis should be expended on those options that fulfill the absolute criteria so that they can be ranked relative to the other high value criteria. General options would typically include:

a. Decontamination and Decommissioning--removing all contamination and all structures to allow unrestricted use of the site;

b. Decontamination--removing contaminated equipment and decontamination of accessible areas; restricted use of the facility for DOE sponsored activities;

c. Safstor--decontamination of accessible areas only, locking, and providing maintenance and surveillance of other areas;

d. Minimum maintenance, severely limited access--removal and disposal of above ground structures, covering of any remaining contamination to a level consistent with the limited access that may occur, and

e. The no action alternative as required by NEPA.

In decommissioning this is normally considered as "walk away". Not all options will be reasonable for all sites.

Cost Benefit Analysis

The options or alternatives should be assessed relative to the absolute and high value criteria. A formal or informal cost benefit analysis is performed to assure that each alternative is appropriately considered.

Independent Verification

Once a disposition is determined, detailed project planning, staffing, arrangements for waste transportation and disposal can be initiated. A decommissioning readiness review is recommended before beginning major decommissioning jobs.

Independent or third party measurements and/or review is usually performed, at the completion of planned decommissioning activities to assure that the established criteria are met, especially if unrestricted access is contemplated.

11.2.3 Design Features

The general design criteria for DOE facilities are contained in the most recent version of DOE's General Design Criteria Manual whereas this document provides guidance for the design of uranium facilities as it affects decommissioning.

Certain features of a facility will greatly effect the cost of decommissioning and the volume of radioactive waste that will require disposal. The following general guidelines apply.

Building Materials

In general, the less permeable building materials are the more easily they can be decontaminated.

Any contaminated concrete with uncoated surfaces will require surface removal and disposal as radioactive waste at the end of life. If there are cracks through which contaminated solutions have penetrated, the entire structure may require disposal as radioactive waste.

Where HEPA filtration is not employed or has failed at some time during facility operation, roofs may become contaminated. Metal roofs can generally be decontaminated sufficiently (if weathering has not already done so) to be released as clean, however, built up roofs, composition roofs, and other types are less likely to be cleanable to unrestricted release levels.

Interior surfaces are most likely to be cleanable if they were completely primed and painted prior to the introduction of radioactive material into the facility. If they are repainted during operation, disposal as clean waste is likely to require paint removal. However, if the coating has deteriorated, cleaning for unrestricted use may be as difficult as if the material had never been painted. Wood will usually become contaminated as will plasterboard and other such materials.

Carpets are generally not recommended for areas which have the potential to become contaminated. Carpets become contaminated, are bulky to dispose of, and do not adequately protect the underlying surface. However, small rugs are often helpful to pick up contamination at the interface between clean and contaminated areas. (These rugs are surveyed frequently and disposed of as radioactive waste when they become contaminated.)

Seamless vinyl floor coverings, with heat sealed joints, protect the underlying concrete or other material from contamination better than conventional floor tiles. If the floor needs to be replaced during the life of the facility it is preferable, where practical to overlay the new material rather than take up the old material exposing the underlying floor.

Ventilation Systems

Ventilation system design will depend on the operations that will be conducted in the facility in addition to decommissioning considerations. Adequate air flow for all operations and good design practices will help keep the facility clean during operations and will facilitate decommissioning. Fiberglass duct work may present a fire hazard (and fires are liable to adversely effect decommissioning as well as operations) and may be more difficult to decontaminate than stainless steel. Welded joints are less likely to collect contamination than bolted ones, however, bolted joints are easier to remove and the most contaminated areas are readily accessible for cleaning.

Piping Systems

Potentially contaminated piping imbedded in concrete is a common and relatively expensive decommissioning problem. Most often piping must be sealed and removed last after all other radioactive material has been removed and the building is being demolished by conventional methods. Often contaminated piping systems provide the major impetus for demolishing a building rather than converting it to some other non-Nuclear use.

It is best to run pipes in chases or tunnels which have been coated to prevent contamination from penetrating the surface in case of a spill or leak. Floor drains should be enclosed in as little concrete as possible to minimize hand jackhammer work at decommissioning.

The material used for piping systems will depend on chemical and physical composition of the solution that will be carried. Wooden pipe, made from Douglas Fir, is most often used for uranium ore slurries containing acid and for abrasive slurries. Some seepage is expected and major pipe breaks are not uncommon. For these reasons, runs of such pipe should be encased in lined trenches or other structures that will minimize the spread of contamination. (Records of past spills and similar events are extremely helpful in planning decommissioning and estimating decommissioning costs.)

While uranium is not generally absorbed by most plastics some of its decay products may accumulate in certain materials. Radiation dose rate buildup may effect both operations and decommissioning. (Radiation dose rates in excess of 150 mrem/hr. have been observed in uranium mill tank liners and piping systems.)

Site Drainage

Any location where contaminated effluents have penetrated the ground is likely to require excavation during decommissioning. Such areas should be minimized by facility design but generally were not in the older DOE facilities which are expected to be the next D&D candidates. Particular attention should be paid to storm runoff from roofs, storage areas, and contaminated equipment storage.

Building ventilators and exhaust systems are likely to contaminate the roof of uranium processing buildings as indicated in Section 4. Runoff from these area, over the life of the facility, will result in contamination of the soil in the runoff area. Therefore, decommissioning will be

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facilitated if, prior to facility operation, runoff is diverted to a waste treatment system.

Excess equipment removed form uranium facilities is often stored outside uncovered or inadequately covered where radionuclides leach from it to contaminate soil. This may greatly increase decommissioning cost. It is preferable to store such materials inside. However, if they must be stored outside, it is advisable to prepare an area by blacktopping or paving. Runoff water should drain to a single point so that effluents can be treated before discharge.

Liquid Effluent Settling and Treatment Systems

Uranium milling and other processes involving the chemical treatment of uranium result in liquid retention basins or liquid/solid settling facilities. From a decommissioning standpoint, the best design for such facilities is a double lined basin with a flexible plastic type liner such as Hypalon. The bottom liner, with waterproof seams, is covered with course gravel and is equipped with liquid level alarms and sampling systems. The upper liner should be impermeable. If the system functions as designed only the upper liner and contained solids will require disposal as radioactive waste during decommissioning. If the pond liner fails at any time during the operating period of the facility, all or a portion of the gravel and lower liner may also require disposal as radioactive waste. This is generally the case to be expected.

Other Features

Installed decontamination and materials handling equipment that facilitate operation and maintenance generally facilitate decommissioning in two ways. First, they can be used for their intended purposes of cleaning and moving equipment during the decommissioning phase. Even more important, they usually contribute to a cleaner, better maintained

facility, where nonfunctional equipment is moved out when it is no longer needed, and work surfaces are kept free of spreadable contamination that makes cleanup difficult.

Operational records of the types and quantities of material handled and of spills, fires, and other abnormal events, are also a vital tool in cost-effective planning and decommissioning.

11.2.4 Residual Contamination Measurements

The type and quantity of residual contamination measurements will be largely determined by these considerations:

a. the future disposition

b. the acceptable dose to the population and/or new users of the site or facility

c. the design of the facility

d. the history of past operations.

If the facility and site are not expected to be accessible to the general population, then site characterization should concentrate on determining that the dose to the new users, if any, will be ALARA, and that possible pathways of exposure to the general population, such as surface and ground water and blowing dust are within the applicable limits and ALARA.

If possible periodic short term intrusion will not be completely precluded, removable surface contamination measurements should be the major focus. Measurements should be sufficiently accurate to assure that transferable contamination will not result in a dose to an intruder exceeding the limits specified in the regulations.

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If longer term intrusion, on the order of weeks, will not be precluded, there may be more of a concern about levels of fixed contamination, about items being removed from the site in sufficient quantity to result in doses above limits, and about the accumulation of contamination in vegetation on site. (Fortunately uranium and its decay products do not bio-accumulate in most ecosystems.)

If the facility will be released for unrestricted use, all of the preceding factors should considered. The uniformity of contamination in soil and building materials should to be determined to assure that limits are met. The operational history must be carefully reviewed to assure that any radionuclides deposited by past incidents are identified and specifically monitored so that they will not be released during final destruction, or future facility modifications, or exposed during excavation, or ground water use.

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SECTION 12

SITING OF URANIUM FACILITIES

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SECTION 12

SITING OF URANIUM FACILITIES

Facilities handling and processing large quantities of uranium should be designed, constructed, and located so that normal operations, anticipated operational occurrences, and accident conditions do not adversely affect adjacent buildings, nor impose an undue risk to the health and safety of the public. While uranium is a relatively low radiotoxic material, appropriate isolation and control must be assured so that radiation doses to the public are maintained as low as reasonably achievable (ALARA). Some factors to be considered are:

a. Natural site characteristics

b. Transportation

c. Other facilities and operations

d. Decontamination and decommissioning

e. Environmental, safety, and health aspects.

In addition, applicable federal and state standards must be met.

Siting criteria are found in DOE 6430.1, Grand Design Criteria Manual (DOE 1983a). DOE 6430.1 is currently being revised and will be reissued. This section provides guidance on the siting of a uranium facility.

12.1 Applicable Standards

The general requirements for the siting of DOE facilities are covered in Chapter 1, Section 3.1 of DOE 6430.1, General Design Criteria Manual

(DOE 1983a); DOE 4300.1A, Real Estate (Real Property) Management (DOE 1983b); and DOE 4320.1A, Site Development and Facility Utilization Planning (DOE 1983c). Additional guidance is provided in DOE/AD/06212-1, Site Development Planning Handbook (DOE 1981a), and other DOE orders such as DOE 5440.1C, National Environmental Policy Act (NEPA) (DOE 1985a); DOE 5480.1B, Environmental Protection, Safety, and Health Protection for DOE Operations (DOE 1986a); DOE 4330.2B, In-House Energy Management (DOE 1982a); DOE 5820.2, Radioactive Waste Management (DOE 1984a); and DOE 4330.4, Real Property Maintenance Management (DOE 1982b). Other references that provide additional guidance in site planning and selection are listed in the bibliography for this section.

DOE 4300.1A (DOE 1983b) specifies the responsibilities and authorities for acquiring property and the evaluations and justifications required, and outlines the methods used for site selection and the specific directors and departments involved. DOE 4320.1A (DOE 1983c) requires the preparation of a Site Development and Facilities Utilization Plan for most DOE sites. The plan is necessary to ensure the future effective and economical development and utilization of DOE facilities. General guidance is provided for the development of criteria for the selection of appropriate sites and facilities to ensure that there is a thorough understanding of program goals, spatial needs, and the potential for existing facilities to meet these needs through sound planning and rational organization. Consideration should be given to the regional setting, land use restrictions, existing facilities and programs, future activities, and the disposition of excess land and facilities.

Additional guidance in siting of facilities is found in LA-10294-MS, Guide to Radiological Accident Consideration for Siting and Design of DOE Nonreactor Nuclear Facilities (Elder et al. 1986), DOE/TIC-11603, Rev. 1, Nonreactor Nuclear Facilities: Standards and Criteria Guide (Brynda et al. 1986), and BNWL-1697 Rev. 1 (Selby et al. 1975). LA-10294-MS provides the experienced safety analyst with accident analysis guidance that can be used in the calculations for siting and design of a nuclear facility.

DOE/TIC-11603 provides DOE field offices and contractors with a standard source document pertaining to the design of a new nuclear facility, modification of an existing facility, and safe operation and decommissioning of all nuclear facilities. BNWL-1697 Rev. 1 provides a base for the development of siting criteria and safety analyses for mixed-oxide fuel fabrication facilities.

A new site should be selected only after careful and thorough analysis and review to ensure that the selection of the site meets program requirements, while considering economic, engineering, and site planning factors and that suitable existing DOE-owned property is not available. Selecting a site involves several steps, beginning with a site-selection survey. Potential sites are examined and reduced to a small group of sites through a preliminary survey of maps. The remaining few sites are carefully analyzed by a site-selection committee using the guidance provided in the Site Development Planning Handbook (DOE 1981a). After the survey is complete, a report is prepared.

The report should contain general information about the site, including site history, regional overview, state, city, and/or county planning information, and coastal zone management information. The existing conditions, such as current mission functions, population, maps, and information on existing land use should be discussed. Discussions on facility use, utility systems, circulation, meteorology, flood plains, soil conditions, geologic faults, wetlands, endangered species, safety and security considerations, and an analysis of existing problems should be included. A planning analysis, presenting the long-range projections of mission, programs, population, and projection methods used, should be performed. A long-range plan and a plan that defines the potential capabilities of the site may also be a part of this report.

As required by the Intergovernmental Cooperation Act (1968), regional, state, and local governmental authorities should be included in the planning and selection process as early as possible and as completely as permitted by the program mission.

12.2 Natural Site Characteristics

Accurate geological, hydrological, and meteorological data must be obtained in the preliminary stages of site selection and development. This information is needed for Preliminary Safety Analysis Reports (PSARs), Environmental Assessments (EAs), Environmental Impact Statements (EISs), and System Design Descriptions (SDDs). Natural phenomena that should be considered in site selection and facility design are earthquakes, lightning, tornados and hurricanes, flooding, water supply, volcanic activity, snow and ice loading, and any other natural attribute of the site that may affect the performance of its mission. Site specific models of natural phenomena events at DOE sites have been prepared by Lawrence Livermore National Laboratory (LLNL), and are cited in DOE 5430.1A.

Baseline environmental concentrations of uranium in soil and water should be determined for the specific site selected. Uranium-238 in soil and rock in the United States may vary from 0.5 to 5 ppm. Measurements of uranium in drinking water show that tap water generally contains less than 0.03 pCi per liter. However, studies indicate that domestic water supplies may have concentrations above 10 pCi per liter, depending on the location.

Similarly, baseline concentrations of uranium in humans due to intake of naturally occurring uranium should be determined. Measurement of normal levels of dietary intake indicate they are about 0.9 to 1.5 micrograms of natural uranium per day. The concentrations of natural uranium in humans range from 0.1 to 0.9 micrograms per kilogram of wet soft tissue and 11 to 30 micrograms per kilogram of bone ash. The levels of uranium detected in workers from bioassay analyses will be affected by the variations in the environmental concentrations and must be considered.

12.2.1 Meteorology

The wind patterns (speed, direction, frequency and duration, and stability) at a site must be tabulated. This data is needed to estimate

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radiation doses to populations from possible releases of radioactive material. Data should also include frequency and intensity of rainfall, snow and ice storms, thunderstorms and lightning strikes, and other events which may affect a facility's power supplies and ventilation, or other safety features.

Nuclear facilities must be built to withstand design basis tornados unless it can be demonstrated that a tornado is not likely to occur. Complete histories of the magnitude and frequencies of such events as heavy rainfall, severe snow and ice storms, severe thunder storms and lighting strikes in the region of the site should be compiled and evaluated to ensure that the location and design of the facility is such that the health and safety of the public is provided.

12.2.2 Hydrology

Precautions should be taken to avoid flood damage, erosion, and water pollution. The water flow of streams, rivers, and reservoirs should be documented, and the maximum precipitation and water levels that might adversely affect plant safety or the storage of radioactive waste should be determined. The design basis 100 year flood may need to be considered in the site selection and facility design to ensure flood protection. This data can be obtained from the National Oceanic and Atmospheric Administration (NOAA).

The effects of seismically-induced dam failures on the upper limit of flood controls at the site should also be considered. The U.S. Geologic Service (USGS) can supply runoff, water distribution, and worst probable flood data. Additional guidance can be obtained from Presidential Executive Order 11296, "Evaluation of Flood Hazard in Locating Federally Owned or Financed Buildings, Roads, and Other Facilities" published by the United States Resources Council (1960).

Finally, the population groups that use water which could be contaminated by plant effluents under both accident and normal conditions must be identified. Water use evaluation should include potable water supplies, both surface and sub-surface, crop irrigation supplies, and recreational uses.

12.2.3 Geology and Seismology

Geologic and seismic data for the site of the proposed uranium facility should be gathered. Earthquake data and maps can be obtained from the National Oceanic and Atmospheric Administration. The geologic conditions underlying all structures, dams, dikes, and pipelines should be examined for the possibility of earth movement that could damage the facility. Natural conditions such as caverns, or potential landslide areas, or man-made conditions caused by mining, or the withdrawal or addition of subsurface fluids should be considered. Facility design may need to comply with the criteria for a design basis earthquake. Location of fault lines, frequency and intensity of earthquakes, location of epicenter, and other seismic data should be analyzed. If the maximum ground acceleration could exceed 0.1 g at the foundation of the plant, special precautions may be necessary. The effects of tectonic structures and active faults that could produce a major earthquake with an epicenter within 200 miles (322 km) of the plant should be estimated. The possible effects of earthquakes from any fault more than 1000 feet (305 m) long within 5 miles (8 km) of the plant shall be considered in the plant design and may render the site unsuitable.

The potential for volcanic activity which could affect the facility should be determined. The potential effects of ash fall on power supplies and safety systems should be evaluated.

The liquefaction potential of the soil and material under the site should be analyzed as should the stability of hillside slopes that could affect the plant. The stability and load-bearing characteristics of the soil at the site should also be determined.

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12.3 Transportation

All nuclear facilities should be isolated from highly populated areas and areas with a high population density. However, the facility should have reasonable access to major transportation networks. Since uranium facilities typically require relatively large volumes of material and frequently heavy loadings, access to rail systems and interstate highway networks may be required.

Since many state governments have the authority to designate traffic routes for shipment of radioactive material, close coordination with state and local agencies is necessary.

12.4 Utilities

Availability of electrical utilities, potable water, and raw water should be considered in the siting of a uranium facility.

12.5 Other Facilities and Operations

In the siting of uranium facilities, the projected effects from nearby industrial, transportation, and military installations and operations should be considered. Potential adverse effects and their impact on the safe operation of the facility should be evaluated. Examples of potential hazards include toxic chemical fumes, flammable gas clouds, aircraft crashes, missiles from explosions, and radioactive material releases.

12.6 Environmental, Safety, and Health Considerations

Uranium facilities should be located where their construction and operation will comply with the provisions of DOE 6430.1, General Design Criteria Manual (DOE 1983a), DOE 5480.4, Environmental Protection, Safety, and Health Protection Standards (DOE 1984b), and DOE 5480.5, Safety of Nuclear Facilities (DOE 1986b), and will not have a significant adverse

environmental impact. An EA and probably an EIS will need to be prepared for the site. DOE 5440.1C, National Environmental Policy Act (DOE 1985a) and Code of Federal Regulations Title 40, Parts 1500 through 1508 (40 CFR 1500 through 1508) (CFR 1986a through 1986i), provide guidance for the preparation and contents of EA and EIS documents.

Facilities that may emit airborne effluents should be located where favorable wind distributions will minimize the levels of contaminants at site boundaries and in nearby populated areas. Consideration of prevailing meteorological conditions and implementation of design limitations could prevent serious offsite consequences of an accidental loss of radiation control at the facilities.

The disposal, storage, or transport of radioactive waste, radioactive mixed waste, and hazardous waste requires careful attention to federal, state, regional, and local regulations. DOE 5480.2, Hazardous and Radioactive Mixed Waste Management (DOE 1982c), establishes procedures for the management of hazardous and radioactive mixed wastes. These procedures follow regulations issued by the Environmental Protection Agency (EPA). DOE 5820.2, Radioactive Waste Management (DOE 1984a), provides procedures for the management of radioactive wastes.

In the siting of any nuclear facility, emphasis must be placed on minimizing the environmental impact and radiation doses to the public. The maximum annual effective dose equivalent (from external radiation, ingestion, and inhalation) permitted by DOE for any member of the public from all routine DOE operations shall not exceed 100 mrem/yr (Vaughan 1985). DOE has established air-pathway-only dose equivalent limits of 25 mrem/yr to the whole body and 75 mrem/yr to any organ as defined in 40 CFR 61 (CFR 1986a). The National Primary Drinking Water Regulations, 40 CFR 141 (CFR 1986k), limit the annual dose from manmade radionuclides in drinking water to 4 mrem. The regulation also limits the gross alpha particle activity (including radium but excluding radon and uranium) to 15 pCi/L. Special precautions should be implemented to limit the potential releases of toxic and radioactive material in facility effluents, both under normal and accident conditions.

The largest potential release of radioactive material resulting from credible accidents should be determined for the specific facility and processes to be conducted. In general, for uranium processing facilities, the maximum credible release scenario involves the rupture of a heated UF_6 cylinder. The impact from an accident such as this would result in the release of uranium and HF, the products of reaction of UF_6 and moisture in air. For a large spill of HF from a tank, the Department of Transportation (DOT-P5800.2) recommends evacuation in all directions to a distance of 150 meters and in the downwind direction to a distance of 3 km. While these distances are considerably greater than would be required for protection from the radiological hazards from the uranium, they may be considered appropriate when establishing a buffer area around a new uranium processing facility.

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SECTION 13

FACILITY DESIGN

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SECTION 13

FACILITY DESIGN

INTRODUCTION

Design criteria are established to provide technical direction and guidance for the planning and design of new facilities, the development of specifications for building acquisitions, and the planning and design of facility additions and alterations. Facility design criteria for DEO uranium facilities can be found in DOE 6430.1A, General Design Criteria (DOE 1983), which is currently being revised. This section provides guidance in the design of uranium facilities such that operation of the facilities will not present an unacceptable risk to the health and safety of personnel, the public, or the environment. The guidance provided herein should be used as a supplement to the required criteria in DOE 6430.1A (1983). Other safety areas such as industrial hygiene and industrial safety are beyond the scope of this manual and are not specifically included; however, federal and state regulations applicable to those disciplines must also be met.

Radiation protection in nuclear facilities is usually achieved by a mixture of administrative and engineered safeguards. A building equipped with a maximum of engineered safeguards and a minimum of administrative controls should be more economical to operate than one with the reverse characteristics. In designing a new facility, all of the necessary physical features can be included; in an old facility, however, it may be physically or economically impossible to meet all of the requirements.

The guidance presented in this Section relate to physical safeguards only; guidance related to administrative control is not included. The term "physical safeguards" is used to refer to the physical, engineered features used to provide contamination control rather than to control against theft (as this term would imply with respect to nuclear material management). In

addition to the radiation protection requirements, facilities containing more than one-third of a minimum critical mass of fissile material are subject to criticality safety requirements, which include the need for a criticality detection system and criticality dosimeters.

13.1 Applicable Standards And Guides

The design criteria contained in DOE Order 6430.1, General Design Criteria Manual, pertain to all new DOE facilities and are mandatory. While uranium processing and handling facilities are not specifically addressed, the majority of the criteria presented are applicable to all facilities which process and handle radioactive materials. A facility that will handle more than 450 grams of plutonium or 450 grams of any combination of plutonium, uranium 235, and uranium 233, must also meet the requirements of DOE Order 5480.5, "Safety of Nuclear Facilities" (DOE 1986). An extensive listing of applicable DOE Orders, standards and guides is provided in the bibliography of this section. Other sources of information are NRC's Regulatory Guides for Fuels and Materials Facilities, International Atomic Energy Agency (IAEA) booklets, and applicable national and international standards.

Additional guidance on the siting and design of facilities is found in LA-10294-MS, Guide to Radiological Accident Consideration for Siting and Design of DOE Nonreactor Nuclear Facilities (Elder et al. 1986), DOE/TIC 11603, Rev. 1, Nonreactor Nuclear Facilities: Standards and Criteria Guide, (Brunda et al., 1988), and BNWL-1697 Rev. 1 (Selby et al., 1975), LA-10294-MS provides the experienced safety analyst with accident analysis guidance that can be used in making the calculations for the siting and design of a nuclear facility. DOE/TIC-11603 provides DOE field offices and contractors with a standard source document pertaining to the design of a new nuclear facility, modification of an existing facility, and safe operation and decommissioning of all nuclear facilities. BNWL-1697 Rev. 1 provides a base for the development of siting criteria and safety analyses for mixed oxide fuel fabrication facilities.

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13.2 Design Objectives

The objective of any good design for a uranium facility is to assure plant, public, and environmental safety during normal or routine operation, and to minimize any potential for loss of life or property in the event of an accident.

The specific facility design chosen depends on the quantity and form of the uranium that will be used. Uranium is a low specific activity material and is, in general, one of the least radioactive of the radioactive elements. However, uranium and its compounds are radioactive, it is classified as a hazardous material, and special precautions must be considered in the design of facilities in which it will be processed and handled. Uranium and its compounds are considered source material and, depending on its enrichment in the isotope U-235, may be classified as special nuclear material. A small quantity of source material is defined in 10 CFR 40.22 as not more than 15 pounds.

The quantity of uranium than can be handled with minimal radiological controls depends on the complexity of the process and the specific form of the material and its enrichment. For production facilities and other facilities, which expect to handle large quantities of material, safety personnel should be consulted and the feasibility of the proposed use should be established, based on the form of the material, the work to be performed, and the engineered and administrative controls to be employed.

The application of these guidelines to specific proposals for the modification of existing facilities or the construction of new facilities requires that judgments be made based on detailed information about the facility, its use, quantities of uranium involved and operations to be performed, degree of need for operating continuity during and/or following postulated accidents, and the potential impact on surrounding facilities and the public. For some facility modifications, the engineering criteria

outlined here may be modified, or reduced, if administrative requirements are increased. A cost/benefit analysis may be necessary to make this decision.

The primary goal of the design objectives is to keep uranium confined in its intended place (i.e., capsule, hood, tank, processing unit, etc.), both during normal operations and under accident conditions. Additional design guidance may be necessary in considering the decontamination, decommissioning, and dismantling (discussed in Section 11) of the facility when it will no longer be needed.

13.2.1 General Design Considerations

It is the policy of the DOE that occupational and public radiation exposure shall not exceed the limits specified in DOE Order 5480.11 (DOE 1988) and shall be maintained as low as reasonably achievable (ALARA).

DOE 6430.1A (DOE 1987) states "As a design objective, exposure of personnel to inhalation of airborne radioactive materials shall be avoided under normal operating conditions to the extent achievable in the directive in the DOE 5480 series on Requirements for Radiation Protection."

While the external radiation doses from uranium operations are typically much less than the regulatory limits (with the possible exception of skin and extremity doses), the ingestion and/or inhalation of radioactive materials are potential exposure pathways that must be carefully considered.

The reduction of radiation exposure to ALARA is a philosophical concept; its actual implementation depends on the interpretation of "reasonably achievable". An optimization process, introduced by the International Commission on Radiation Protection (1982), may be used to determine if an activity is being performed at a sufficiently low level of collective dose equivalent so that any further reduction in dose will not

justify the incremental cost required to accomplish it. In optimization the cost of reducing radiation exposure must be compared with the benefit of the reduction. The value in dollars of a person-rem of radiation dose has not been firmly established nor does this manual suggest a value. For design purposes, the Nuclear Regulatory Commission (10 CFR 50, Appendix I) has recommended $1000 per person-rem. DOE/EV/1830-T5 (DOE 1980) suggests that if a dose reduction could be achieved at a cost of $\leq$ $1000/person-rem, it is cost beneficial and should always be done. Additional discussions on the cost-benefit concept of dose reduction can be found in DOE/EV/1983-T5 (DOE 1980).

Equipment reliability and human factors engineering should be considered in the design of uranium facilities. Both of these factors may significantly affect radiation doses and the effectiveness of personnel response to abnormal conditions. Reliability data may be available for much of the equipment that will be used. If industry information is not available, reliability analyses should be conducted. The degree of reliability that is justified may require an evaluation of the cost of the reliability versus the expected dose reduction. Recommendations, provided in Energy, Research, and Development Administration (ERDA) 76-45-2 SSDC-2, Human Factors in Design, should be considered during the design of control panel arrangements, instrument indicators and readouts, and alarm indicators.

By design, the failure of a single component shall not result in an unacceptable radiological consequence and should not result in an undesirable radiological consequence. Unacceptable radiological consequences include criticality and radiation exposures, or radioactive material releases in excess of DOE Order 5480.11 limits. Undesirable radiological consequences include radiation exposures in excess of administrative limits, loss of containment or confinement of radioactive materials, and skin contaminations. Analyses of hazards and assessment of risks must be made during conceptual and preliminary design activities, and further developed during the detailed design phase. The safety analyses must be in accordance with DOE Orders 5481.B, 5480.5, 5700.2C, and 6430.1A.

In the planning and designing of buildings, other structures, their operating components and systems, all aspects of operation and maintenance must be considered. This includes accessibility, dismantling, replacement, repair, frequency of preventive maintenance, inspection requirements, personnel safety, and daily operations. Facility planning and design must utilize the knowledge and experience of those persons who will be responsible for operating and maintaining the completed facility. The "lessons learned" from the operation and maintenance of existing facilities should be used to avoid repeating mistakes made in past designs.

Equipment that requires periodic inspection, maintenance, and testing should be located in areas with the lowest possible radiation and contamination levels. Equipment that is expected to be contaminated during operation should have provisions for both in-place maintenance and removal to an area of low dose rate for repair. Maintenance areas for repair of contaminated equipment shall include provisions for containment and confinement of radioactive materials.

Engineered safeguards must be designed so that they could continue to function during and following an accident or emergency condition. The need for an Emergency Control Station shall be determined for each facility in the initial design effort to complement the engineered features by providing "a location within or near a designated critical facility or plant area for the purpose of maintaining control, orderly shutdown, and/or surveillance of operations and equipment during an emergency." (DOE Order 5500.1A) Facilities shall be designed to facilitate the arrival and entry of emergency personnel and equipment in a radiological emergency, and to allow access to repair/corrective action teams.

Equipment shall be available to allow an early and reliable determination of the seriousness of an accident. Installed on-line equipment shall be protected to the extent necessary to ensure reliability under accident conditions. To further enhance equipment reliability, the emergency equipment should, to the extent practical, be the same equipment used for routine operations. (DOE Order N5500.2)

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The design effort shall identify the emergency power requirements and the means to satisfy them.

Emergency radiological equipment shall be installed, or located, in areas that permit periodic inspection, testing, and maintenance.

Additional emergency preparedness guidance is provided in Section 10. Decontamination, dismantling, and decommissioning requirements should be considered in the design of a facility. (DOE Order 5480.11) Section 11 provides additional information on these topics.

13.2.2 Confinement

The confinement system is a series of physical barriers that, together with a ventilation system, minimize the potential for release of radioactive material to work areas and the environment under normal and abnormal conditions. The primary design objective for the confinement system shall be to avoid exposure of personnel to inhalation of airborne radioactive material under normal operating conditions to the extent achievable as specified in DOE 5480.11, essentially zero exposure of the public and plant personnel to airborne contamination (DOE 6430.1). Dispersible uranium should be separated from the ambient environment by at least two barriers, and from an operator by at least one barrier. Non-dispersible forms of uranium, such as metallic material, may not require confinement as long as it is protected from surface corrosion or degradation that could generate airborne radioactivity.

Primary confinement refers to the barrier that is or can be directly exposed to uranium, e.g., sealed process equipment (pipes, tanks, hoppers), conveyors, furnaces, machine enclosures and cells, and their ventilation systems. The primary confinement barrier should prevent the dispersion of uranium through either sealed construction, or atmospheric pressure differential, or some combination of the two. For example, process equipment that is not sealed but contains uranium material in process

should be encased in ventilated enclosures or other confinement barriers. Enclosures, bags, and other sealed containers can be considered primary confinement. The selection of primary confinement material must consider the effects of chemical reactivity and the potential for spontaneous ignition of metallic uranium on the material.

The primary confinement barrier protects operators from contamination under normal operating conditions. This type of barrier is likely to be breached under accident conditions (seal failure, improper packaging operations, leaks of flanged joints, etc.).

The primary confinement should be operated at a pressure negative to the secondary confinement in which it is located and be exhausted through a ventilating system. The barrier and its accessory equipment should be designed to prevent accidental liquid flooding. All primary confinement piping joints should be tested for leak tightness. Penetrations in the primary confinement barrier, such as conduit, ports, ducts, pipes, and windows, should be protected against the release of radioactive material.

Air Recirculation Systems

Where practicable and/or necessary by reason of the nature of the process being conducted, recycle ventilation systems should be used in process enclosures. Inert gas systems shall, unless impractical to do so, be designed as recycle systems. However, extreme caution must be exercised in the use of recycle systems for contaminated or potentially contaminated air. A recirculation system shall not direct air to an area where the actual or potential contamination is less than the area from which the air originated. The decision to use a recirculation system in a contaminated area shall be based on a documented safety evaluation that compares the risks versus the benefits. (DOE Order 6430.1A)

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Filtration shall be provided to limit the concentrations of radioactive material in recirculated air to ALARA levels and no greater than the concentrations permitted in the DOE Orders for uncontrolled areas. At least one stage of high efficiency particulate air (HEPA) filters shall be used in ventilation systems that exhaust to occupied areas and to the environment. The design shall allow for in-place testing of individual HEPA filters or filter banks.

Continuous sampling and monitoring of recirculated air for airborne radioactive material shall be provided downstream of fans and filters. Monitoring of the differential pressure across the filter stages should be provided. Monitoring for airborne radioactive material behind the first HEPA filter or stage should be provided. Means for automatic diversion of airflow to a once-through system or stage should be provided. The monitoring system alarm should result in the automatic diversion of airflow to a once-through system or a parallel set of filters.

Secondary Confinement

The secondary confinement barrier encloses the room or compartment in which the primary confinement barrier is located, and provides contamination protection for plant personnel outside the area of secondary confinement. Secondary confinement rooms, compartments, or cells should be separated from each other by fire doors. Both barrier walls and fire doors should be constructed of materials capable of withstanding a design basis accident. The secondary confinement should be designed for pressures consistent with criteria for the ventilation system. The secondary confinement area should be at a positive air pressure with respect to the primary confinement areas.

The use of autoclaves for heating UF_6 cylinders should be evaluated in terms of providing an additional safety barrier.

Building

The building is the structure enclosing both primary and secondary confinement barriers, as well as the offices, change rooms, and other support areas that are not expected to become contaminated. It is the final barrier between the potential contamination and the outside environment. The building structure or any portion thereof may serve as the secondary confinement barrier provided the requirements for both structure and confinement are met. That portion of the structure housing activities involving radioactive material in dispersible form must be able to withstand design basis accidents, site related characteristics, and missiles without a breach of integrity that would result in releases of radioactive material from the structure in excess of maximum permissible amounts.

13.2.3 Design Basis Accident (DBA) Events

Safety class systems (ventilation, electrical, fire protection, and utility systems) shall be designed to provide confinement of radioactive materials under normal operations and DBA conditions. The degree of confinement of radioactive materials shall be sufficient to limit releases to the environment to the lowest reasonably achievable level. In no case shall the applicable exposure regulations be exceeded, either with respect to the operating personnel, or to the public at the boundary or nearest point of public access. Consideration shall be given to the probability and effects of DBAs. Protection of employees within the facility shall be a consideration in all aspects of the design. The nature of the material to be handled, including secondary isotopes and impurities, and other radioactive elements present, shall be considered in making these assessments.

Structural design, including loading combinations and construction of critical items, shall, as a minimum, be in accordance with current editions of pertinent nationally recognized codes and standards. All other facility

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design shall conform to applicable criteria specified in DOE Order 6430.1 and to other site- or process-specific criteria developed for the facility.

Design Basis--Fire--Development of the DBF shall include consideration of conditions which may exist during normal operations and special situations, such as during periods of decontamination, renovation, modification, repair, and maintenance. The structural shell surrounding critical areas and operating area compartments and their supporting members shall be designed with sufficient fire resistance so that it will remain standing and continue to act as a confinement structure during the DBF postulated for the facility assuming failure of any fire suppression system which is not designed as a critical item. Fire resistance of this shell shall be attained as an integral part of the structure (concrete walls, beams, and columns) and not by a composite assembly (membrane fire-proofing). The fire resistance rating must never be less than two hours. Penetrations in this shell shall incorporate, as a minimum, protection against DBF exposures unless greater protection is required by other criteria. The systems identified as critical items for critical areas shall be designed to continue to operate during the DBF. A high degree of reliability and/or redundance shall be required of all protective features of the ventilation system to assure its effective operation even if normal plant utility and fire protection systems fail.

The Design Basis--Explosion is the rupture of a primary confinement barrier with an accompanying energy release equivalent to an internal pressure of 105 psi (7.38 kg/cm^2). (This energy release will not only result in a pressure wave, but may also generate missiles within the process area.)

The Design Basis--Criticality is an accidental excursion of a heterogeneous liquid-powder mixture with a neutron spike yield of 10^{18} fissions, releasing about 30,000 BTU in less than one second, or an accidental pulsating excursion with a total yield of 10^{20} fissions. (This energy release may disperse unencapsulated uranium from a typical

process enclosure and may pressurize the room.) Design of nuclear criticality control provisions shall meet, as a minimum, the requirements of DOE Order 5480.15 (Safety of Nuclear Facilities). Design shall further assure that fissile material shall not be displaced to form a critical mass in the event of an internal or external accident. Whenever possible, poisoned or geometrically favorable tanks and process vessels shall be provided to minimize reliance on administrative control. A system of backflow prevention, such as air gaps, shall be provided to prevent the inadvertent transfer of liquids from geometrically favorable or poisoned containers to unsafe containers. Positive control to prevent the discharge of liquids from geometrically favorable or poisoned containers to unsafe containers shall be provided.

The Design Basis--Power Failure is the loss of total electric power for 60 s, and the loss of normal electric power for 48 hrs. Total electric power means all sources of electric energy, delivered as well as auxiliary and standby. Normal electric power means the services usually supplied by a utility company.

The Design Basis--Accident Water is the result of an uncontrolled water hazard; this occurs when water that is supplied to the plant from a controlled external source is released, through a mishap within the plant, for 30 min. in a manner which results in loss of a system, subsystem, structure, or component important to the integrity of the confinement system. This accident concept includes both the effect of accidental flooding within the plant and the loss of feedwater to any equipment that, without adequate water supply, would prevent the functioning of the confinement system.

The Design Basis Accident--Natural Phenomenon is the effect of site related conditions and includes tornado and other wind and storm conditions, earthquakes, floods, and volcanic activity. Additional information is given in Sections 13.3 and 13.5. Site specific models of natural phenomena events at DOE sites have been prepared by Lawrence Livermore National Laboratory and are cited in DOE 6430.1A.

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13.3 Structural Criteria

The structure and its associated critical equipment, ventilation, electrical, fire protection, and utility systems should be designed to provide confinement of radioactive materials in the event of any DBA that can be postulated for the facility.

The structural design, the load combinations, and the construction of critical safety and fire protection features shall be in accordance with the latest edition of applicable nationally recognized codes. When local codes, or regulations, are more stringent than the nationally recognized codes, the local codes should be followed.

13.3.1 Tornado Resistance

Critical operating areas of the facility shall be designed to withstand the Design Basis Tornado. More definitive guidance is provided in Chapters XXI and XXIII of DOE 6430.1A.

13.3.2 Lightning Protection

Lightning protection should be provided for all facilities. Lightning protection systems shall be designed in accordance with the National Fire Protection Association (NFPA) 78, Lightning Protection Code, (NFPA 1983).

13.3.3 Seismic Design Requirements

Seismic parameters shall be developed for the site to determine a design basis earthquake (DBE) and an operating basis earthquake (OBE). The smaller earthquake, the OBE, shall be no less than 0.05g in terms of ground acceleration. Critical items shall be designed to withstand the DBE and shall be capable of continued operation after the occurrence of an OBE. Critical items are those structures, systems, and components whose

continued integrity and/or operability is essential in preventing and mitigating the consequences of any accident that occurs. These items include:

a. Main structures

b. Fire protection systems

c. Ventilation systems

d. Confinement piping and equipment

e. Critical utilities, instrumentation, monitors and alarms

f. Material in process and in storage (nuclear criticality).

The DBE and OBE shall be assumed capable of occurring at any time, except that the simultaneous occurrence with any other limiting site-related event such as a tornado, fire, or flood need not be considered for design purposes, except where the joint occurrence is causally related. Additional design criteria is given in DOE 6430.1A.

Other Natural Phenomena

Design loads and considerations for other natural phenomena shall provide a conservative margin of safety greater than the maximum historical levels recorded for the site. Protection against flooding shall be based on no less than the probable maximum flood (PMF) for the area as defined by the Corps of Engineers. The possibility of seismically induced damage or failure of upstream dams shall be taken into account in assessing the nature of flood protection required for the facility. If the facility is in a location which may be subject to ash-fall from volcanic action, consideration should be given to the effects of ash-fall on ventilation and electrical systems.

Explosion, Internal Pressurization, Criticality, and Other DBA Causes

Analyses shall be made to determine the probable consequences of DBAs, and critical areas and items shall be designed to withstand these DBAs. That portion of the ventilation system which is an integral part of the critical areas shall be designed to withstand the DBAs so that it will remain intact and continue to act as a confinement system.

13.4 Building Layout

Building layout is extremely important in the operation of a uranium facility. Improper or poor layout can lead to operational difficulties and, in some instances, contribute to the development of abnormal situations that may affect personnel, the environs, and/or the cost of operating the facility. Normally, three areas are involved in the overall building layout:

a. The process area, where uranium or other contaminated material is used, handled or stored;

b. The controlled area, which is normally free of contamination but could potentially become contaminated; and

c. The uncontrolled area, which includes all areas where no radioactive materials are permitted and no radiological controls are normally necessary. Offices and lunch rooms are included in this category.

13.4.1 Objectives

To the extent possible, the following objectives shall be achieved in the design layout of the facility:

a. Planned radiation exposures to personnel shall be within the prescribed limits of DOE Order 5480.11. Radiation exposures to individuals and population groups shall be limited to the lowest level reasonably achievable.

b. The planned or unintentional release of radioactive materials from the facility shall be confined to the limits of DOE Order 5480.11 and ALARA.

13.4.2 General Design Criteria

All planned processing, research and development, scrap and waste handling, analytical, shipping, and receiving operations shall be accommodated. Receiving operations, involving removal of radioactive material from protective shipping containers, shall be performed in a handling area having provisions for confinement.

The possibility of multi-shift per day operation shall be considered in allocating space for personnel support facilities and for any special equipment required to support multi-shift operations.

High risk areas shall be compartmented for isolation to minimize productivity loss and financial loss in case of a DBA.

Where feasible, a modular construction concept should be used to expedite recovery from DBAs and provide versatility.

The use of combination filling, weighing, heating, and sampling stations should be considered in facilities handling heated UF_6 cylinders to minimize movement of filled, heated cylinders.

All movement of personnel, material, and equipment between the process area and the uncontrolled area should be through a controlled area or air lock. Doors providing direct access to the process area from the

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uncontrolled area (including the outside of the building) are permissible provided they are used only as emergency exits or infrequently for nonroutine operations. All such doors should have air tight seals. Those doors without air locks should also have alarms that sound when the doors are open to signal that a breach in the personnel-control system exists.

Personnel exits shall be provided in accordance with the NFPA Life Safety Code. In those areas where an accidental breach of a primary confinement will expose personnel to radioactive material, a distance of approximately 75 ft shall be considered the maximum level distance to place personnel beyond or through the next confinement barrier. Such a barrier would be a partition separating two different air zones, the area of refuge being on the upstream side of the barrier. The assured airflow through the barrier should be in the opposite direction of the exit travel.

Normal administrative traffic should be restricted to the uncontrolled and controlled areas and should not require passage through the process area. Process traffic should be restricted to process and controlled areas and should not require passage through uncontrolled areas.

Indicators, auxiliary units, and equipment control components that do not have to be adjacent to operating equipment should be installed outside high-radiation or highly contaminated areas. Units and components located in high radiation areas should be designed so that they can be removed as quickly as possible.

Equipment requiring frequent servicing or maintenance should be modularized and standardized to the extent possible.

Work spaces around equipment (pumps, valves) in radiation areas that requires regular maintenance should be shielded from the radiation emitted by adjacent components.

Provisions should be made for the quick and easy removal of shielding and insulation that cover areas where maintenance or inspection are necessary activities. Equipment should be designed to permit visual inspection wherever possible.

Passageways should have adequate dimensions for movement, repair, or disposal of equipment.

Equipment components that require frequent operation or maintenance should be at a convenient height for such work.

In areas where it is likely that personnel will wear protective clothing or breathing-air systems, sharp equipment projections, on which clothing may be caught, should be avoided. (DOE/EV/1830-T5)

Water collection systems should be provided for water run-off from any source including fire fighting activities. The collection systems shall be designed to prevent nuclear criticality, to confine radioactive materials, and to facilitate sampling and volume determinations of waste liquids and solids.

To reduce the spread of contaminated liquids, floors should be sloped to local drains. Drains should be in a location that will not allow contaminated liquids to flow into areas occupied by personnel.

All areas that house tanks or equipment containing contaminated liquids should be curbed to limit the spread of liquids. (DOE Order 6430.11)

Noncombustible and heat resistant materials should be used in radiation areas that are vital to the control of radioactive materials and in equipment necessary for the operation of radiological safety systems. These materials should be resistant to radiation damage and should not release toxic or hazardous by-products during degradation in accordance with IAEA Safety Series No. 30 (IAEA 1981).

Floors, walls and ceilings should have a smooth, impervious, seamless finish. Light fixtures should be designed to be sealed flush with the ceiling surface to minimize horizontal surfaces and prevent entry of contamination into the fixtures in accordance with IAEA Safety Series No. 30 (IAEA 1981). Protective coatings (e.g., paint) used in radiation areas should meet the criteria in ANSI N512-1974.

An emergency lighting system should be provided in radiation areas to facilitate egress in emergencies. The emergency lighting shall meet the requirements of NFPA 101 (NFPA 1985).

Space should be allocated for radiological monitoring stations and washing facilities at exits from contaminated areas.

13.4.3 Process Area

The uranium process area is a contiguous group of rooms containing all operations involving uranium, including processing, shipping, receiving, storage, and waste handling. Facility design should provide, to the maximum extent practicable, sufficient space and versatility to accommodate equipment for programmatic changes and process modifications.

The initial line of defense to protect workers in a process area is the primary confinement system, which includes enclosures, conveyor lines, the ventilation system and process piping. The primary confinement system should be designed to minimize the impact on people, facilities and programs of accidents and abnormal operations. The type of confinement enclosure used (hood, glove box, remote operation cell) depends on the amount and dispersibility of uranium that will be handled and on the process involved.

Piping and Valves

Piping and valves for radioactive liquids should not be field-run (i.e., pipe and valve locations shall be located as specified on approved drawings and NOT at the discretion of the installer).

Notches, cracks, crevices, and/or rough surfaces that might retain radioactive materials should be avoided in the design of radioactive process piping systems.

The contaminated piping system should be designed so that effluents from leaks in the system can be collected without releasing the effluents into personnel access areas or to the environment.

When component or system redundancy is required, sufficient separation of equipment should be employed so that redundant systems (or equipment) cannot both be damaged by a single accident.

Heating systems for UF_6 cylinders and cold traps should be designed such that the system cannot be activated unless there is positive verification that a vent path is open to pressure-sensing instrumentation. Heating systems for UF_6 cylinders and cold traps should be designed such that overpressure relief or automatic heat termination is provided upon overpressurization of cylinders or traps being heated.

Stainless steel or other appropriate material should be used in all radwaste and process system piping and equipment to ensure that smooth, nonporous, corrosion-resistant materials are in contact with contaminated liquids.

Piping systems used for conveying radioactive and corrosive materials should be of welded construction whenever practicable. Flanges should be used at equipment nozzles only when necessary for servicing.

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There should be no cross connections between contaminated and non-contaminated systems. If cross connections are necessary, an air gap should be used to prevent backflow of contaminated liquids into the non-contaminated system.

Every pipe entering or exiting a process cell, and/or contaminated area, should be equipped with a block valve.

Contaminated process piping systems should be designed so that pockets or traps are eliminated wherever possible and piping can be flushed and drained except where loop seals are required. Floor drains should have the capability to be sealed.

Reduction in pipeline size in contaminated process piping systems should be made with eccentric reducers installed flat side down to avoid formation of traps. Eccentric reducers are only necessary for horizontal pipe runs.

Changes in direction of process piping should be made with long-radius elbows or bends. Bends are preferred, where practicable, except for solids transport lines, where blinded tees or laterals have been proven to prevent erosion.

If gaskets are required in process piping or associated hardware, the selected gasket material should not deform or degrade and leak when in service. Teflon should be avoided for most applications, but, if needed, its use will require implementation of a most rigorous inspection routine to ensure recognition of degradation and replacement prior to failure.

Pipe sleeves should generally be provided when piping passes through masonry or concrete walls, floors, and roofs, except shielding walls. Sleeves should be sloped to drain toward the controlled area. The space between the pipe and the sleeve should be packed and sealed.

If underground piping for transporting radioactive or hazardous materials is required, it shall be installed inside another pipe or tunnel that provides a second barrier to the soil. Provisions to detect a failure in the primary piping (leak detection) shall be provided. An effective solution may be to install a double walled pipe with an annular space that can be sampled at intervals not exceeding 300 ft. The underground piping should also have cathodic protection.

All valves not functionally required to be in contact with contaminated liquids should be located outside the cell and/or contaminated areas (e.g., steam,air, water) in accordance with IAEA Safety Series No. 30 (IAEA 1981)

Process valves should not be located at piping low points in the piping except in cases where it is necessary to properly drain the piping when needed.

Valve seals and gaskets must be resistant to radiation damage.

Globe valves less than 2 in. in diameter located in drain lines should be Y-patterned to facilitate rodding if plugging occurs.

Straight through valves should generally be used to simplify maintenance and minimize particle traps.

Valves should be designed to operate in the stem-up orientation. Valves and flanges should be located to minimize the consequences of contamination from leaks.

Piping that conveys contaminated process solutions should be equipped with a packless (bellow seal) block valve.

Piping that conveys contaminated process solutions in a normally pressurized system should contain a check valve in addition to the packless block valve to prevent backflow.

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The primary block valve and/or check valve should be of equivalent corrosion-resistant material to the associated piping in the in-cell and/or contaminated areas.

The use of pumps in contaminated piping systems should be avoided to reduce potential contamination problems from pressurization and to reduce the maintenance requirements associated with pumping. The use of gravity flow, jets, or airlifts is a suitable alternative.

Structure

Floors shall be designed in accordance with code requirements considering the maximum loads anticipated.

Storage

Storage facilities in the process areas should be designed to prevent the exposure of operating personnel and to meet the requirements for security and safeguards in DOE Order 5632.4, "Physical Protection of Special Nuclear Material" and other DOE Orders in the 5630 series that collectively comprise the DOE safeguards program to guard against theft or unauthorized diversion of special nuclear materials.

Shielding

Provisions should be made to accommodate the shielding of appropriate items in the process area. All structures (floors, walls, glove boxes, etc.) may require additional shielding during the lifetime of the facility because of contamination, increased throughput, or higher radiation levels of the material being processed.

DOE Order 6430.1 (DOE 1983a) established a radiation level of 1 rem/yr to the whole body as a design guide. In applying this criterion in facility design, efforts shall be made to maintain radiation exposures as

low as reasonably achievable. The design of a routinely occupied portion of a process area should never be based on anticipated dose rates in excess of 100 mrem/hr. DOE 5480.11 (DOE 1988) includes a requirement that dose equivalent rates in a routinely occupied location shall average less than 0.5 mrem/hr or 10 mrem/wk. It further requires that a process area with a dose rate between 100 mrem/hr and 5000 mrem/hr shall be controlled with signs or lockable barriers. For dose equivalent rates of 5000 mrem/h or greater, lockable barriers shall be provided.

Shielding and other radiation protection requirements shall be considered for non-work areas requiring intermittent access, such as for preventative maintenance, component changes, adjustment of systems and equipment, etc.

Concrete radiation shielding should be in accordance with ANS 6.4 and ACI 347.

Straightline penetration of shield walls should be avoided in order to prevent radiation streaming.

Robotics and/or shielded operations performed remotely should be used as much as practicable and shall be used where it is anticipated that exposures to hand and forearms would otherwise approach the design criteria of 10 rem/yr. Also, robotics or other nonhand contact methods should be used where contaminated puncture wounds could occur.

Shielding materials shall be noncombustible or fire resistant, to the extent practicable.

Confinement Devices

Different devices may be used to provide confinement and control of radioactive material. The selection of the appropriate device will depend on the quantity of material, its form, and the operations to be performed.

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Sealed source containers shall be designed to prevent contact with and dispersion of the radioactive material under all conditions and when inadvertently dropped. Sealed sources shall be shielded as required to ensure that personnel in routinely occupied areas do not receive more than 0.5 mrem/h.

Fume hoods may be used for some operations with uranium, depending on the quantity and dispersibility of the material. A safety analysis should be performed to determine allowable quantity, depending on the characteristics of the material to be handled. The location of each hood shall be evaluated with respect to ventilation supply and exhaust points, room entrances and exits, and normal traffic patterns. Hood faces should not be located within 10 ft of the closest air supply or exhaust point. Hoods should not be located in or along normal traffic routes.

An open-faced hood shall be designed and located to provide a constant air velocity across the working face. A face velocity of 125 linear feet per minute or greater over the hood face area shall be provided to ensure personnel protection and contamination control. Much of the nuclear industry uses 150 linear ft/min as the criterion. If room air currents might upset the uniform entrance of air, the hood exhaust requirements shall be increased. Turbulence studies may be necessary to verify adequate control of radioactive material. Physical stops should be provided to ensure that the required face velocity is maintained.

Hood design and filtration systems shall comply with the criteria established in ERDA 76-21, Nuclear Air Cleaning Handbook (ERDA 1976, Industrial Ventilation, A Manual of Recommended Practice, by the American Conference of Governmental Industrial Hygienists (ACGIA 1980), and by Oak Ridge National Laboratory (ORNL in ORNL-NSIC-65, Design, Construction, and Testing of High Efficiency Air Filtration Systems for Nuclear Application (ORNL 1970).

Service outlets (gas, water,etc.) should be located along the sides, or at the back of the hood, and controlled through knobs and switches at

the front of the hood. Electric outlets should be on the outside of the hood. The hood structure should have a smooth, corrosion resistant inner surface that is made of, or coated, with easily decontaminated material.

Consideration should be given to the space and loading if shielding will be required. In addition, provisions for radiological monitoring instrumentation near the hood should be considered.

Conveyors and other enclosures shall be designed to control and minimize the release of radioactive materials during normal operations and under postulated design basis accidents. Non-combustible or fire and corrosion resistant materials should be used in the construction of the confinement system, including any shielding employed. Fixed modular construction should be employed wherever possible, using a standardized attachment system that will permit replacement or relocation within the enclosure system with a minimum spread of contamination.

The enclosure design shall include sufficient work-space to permit removal of materials and easy personnel access to all normal work areas, and provide for the collection, packaging, storage, and/or disposal of waste generated in the operation of the enclosure.

Only equipment essential to the function of the enclosure should be inside the enclosure. Equipment to be located inside the enclosure should be designed to minimize maintenance requirements. Equipment should also be designed (or selected) to minimize contamination traps, to facilitate decontamination, and to simplify decommissioning.

Equipment should be designed to preclude sharp corners, barbs or pointed parts, or pinching points that could puncture gloves or skin. All corners should be rounded, burrs removed, etc.

Ergonomics shall be considered in designing the height of glove box parts and access to inner surfaces and equipment.

A HEPA filter should be installed on the air inlet to the enclosure to prevent the backflow of contamination. Prefilters should be installed upstream of the HEPA filter. The exhaust outlet for each enclosure should have a prefilter and a testable HEPA filter. HEPA filters, downstream of the enclosure, should be readily accessible for filter changeout.

Enclosures should be designed to operate at a negative pressure with respect to the room in which they are operated. Differential pressure gauges should be installed on each enclosure or integrally connected series of enclosures. Control devices to prevent excessive pressure or vacuum buildup shall be positive acting, or automatic, or both. The ventilation system shall be designed to provide and maintain the design negative pressure during normal operations and the design flow through a breach during abnormal conditions. There should be exhaust capacity on demand that will promptly cause an inflow of air at greater than 125 linear feet per minute through a potential breach of a single enclosure penetration of the largest size possible. Filters, scrubbers, demisters, and other air-cleaning devices shall be provided to reduce toxic or noxious gases and airborne particulates to the ventilation system.

Each enclosure or integrally connected series of enclosures should be equipped with an audible alarm that alerts personnel if the enclosure pressure relative to the room in which the enclosure is located falls below a pre-established limit.

The number of penetrations for enclosure services should be minimized. Fittings chosen should provide a positive seal to prevent migration of radioactive material. Penetrations for rotating shafts should not be permitted except where rotating shafts have seals. Seals for rotating shafts are very reliable and are preferred to motors inside the enclosure.

Any gas supply system connected directly to an enclosure shall be designed to prevent pressurization, flow in excess of the exhaust capacity and backflow. Vacuum pump exhaust shall be filtered and exhausted to the enclosure or other acceptable exhaust system.

Process piping to and from enclosures should be equipped with backflow preventors and should be of welded stainless steel construction.

Windows, gloves, and sealants used with them shall be of materials that will resist deterioration by chemicals and radiation.

Access openings should be located to facilitate both operating and maintenance work. The need for two-handed operation, depth of reach, mechanical strength, and positioning with respect to other access openings should be considered in the design. Covers should be provided for each opening. The covers should provide shielding equivalent to the enclosure walls.

Access openings should be designed and installed to facilitate the introduction and removal of equipment and supplies without compromising contamination control. Airlocks should be considered. Access openings and airlocks should be designed to be at negative pressure with respect to the workstation and positive pressure with respect to the enclosure.

Windows shall be constructed of noncombustible or fire resistant materials that are resistant to scratching, breaking, chemical attacks, and radiation degradation. They should be securely fastened and gasketed. Lighting fixtures should be mounted on the enclosure exterior to the extent practicable.

Fire protection shall be provided in the enclosure and conveyor systems to meet DOE improved risk objectives. Automatic fire suppression must be considered when a credible fire could produce a loss (including decontamination) in excess of $250,000. When the potential loss might exceed $1 million, an automatic fire suppression system is mandatory. Discrete work stations within an enclosure should be separated from each other by fire stops to prevent the spread of fire. Where possible, fire stops should be designed to be normally closed. For systems in which fire stops must normally be open, closure should be automatic upon actuation of

the fire sensing system. Fire sensing systems should be fast acting and highly reliable. Instead of such a fire sensing system, an oxygen deficient atmosphere may be provided as the normal or required operating atmosphere within the enclosure. Where automatic systems are not required, fire detection shall be installed. Provisions shall also be made for manual fire suppression where deemed necessary.

Actual sources inside the enclosure should be shielded rather than the enclosure, if possible. However, the enclosures should be equipped with or capable of accepting any necessary shielding.

13.4.4 Controlled Area

All support facilities with a potential for periodic low-level contamination should be located in the controlled area. These facilities include change rooms and decontamination rooms for personnel; health physics laboratories; facilities for the receipt, temporary storage and shipment of radioactive and potentially contaminated materials; maintenance rooms for regulated equipment; mechanical equipment rooms; and other laboratory facilities.

In controlled areas where radiation exposure is not a necessary part of the work being performed, shielding shall be provided to reduce the dose to occupants to less than 0.5 mrem per hour in accordance with DOE 5480.11 (DOE 1988).

Airlocks between controlled and uncontrolled areas should be used to minimize the spread of contamination resulting from an inadvertent release of radioactive materials or from a fire. Airlocks should also be provided at locations where there is a potential for contamination to be spread from an area of high contamination to one of lower contamination.

Where possible each controlled area should have a single access and exit point for personnel during normal operation. Access points should be

accessible through change rooms. Other access and exit points shall be available as required for emergencies and in compliance with NFPA Life Safety Code (NFPA 1985).

Space for step-off pads and radiation monitoring and survey equipment should be provided at the exit from controlled areas that are potentially contaminated and between high- and low-level contamination areas. The space provided should be sized to accommodate the expected work force.

Change Room

Change rooms shall be available for both men and women, with locker capabilities to support the anticipated number of workers and support personnel. Change rooms should include facilities for storing and dispensing clean protective clothing; a well defined area near the exit from the controlled area for the temporary storage of potentially contaminated clothing; and adequate shower facilities. The clean side of the change room should be easily separable from the potentially contaminated side of the room.

Space for step-off pads and radiation monitoring survey equipment shall be provided for personnel and equipment leaving the controlled portion of the change room.

Liquid wastes from potentially contaminated showers should be routed to the liquid rad waste system or to a holding tank that may be sampled and monitored before the waste is released.

The ventilation system should be designed to prevent the spread of contamination from the controlled to the uncontrolled portion of the room.

Personnel Decontamination Room

A personnel decontamination room or station should be provided for each uranium facility. It should be located near or in the change rooms. An adequately equipped station should have communications equipment, a work bench with a cabinet for decontamination supplies, an examination chair, and a sink and showers. Both sink and showers should be connected to a holding-sampling tank or a monitored sewer system. The room should contain equipment for performing skin decontamination and collecting special bioassay samples.

Health Physics Lab-Office

Health physics personnel in a uranium facility should be assigned lab-office space at or near the exit from the process area into the controlled area. As a rule of thumb for determining space needs, two monitors are needed for the first 30 radiation workers and one additional monitor for each additional 20 radiation workers. Space should be included for the readout of radiation protection instrumentation, survey records documentation, for counting equipment, and portable instruments.

Mechanical Equipment

Where possible, mechanical equipment (motors, pumps, valves, etc., that may be a source of contamination) should be located in the controlled area. Enclosures that will contain the contamination or will be easy to decontaminate should be placed around the equipment.

13.5 Service and Utility Systems

The design of utility services must provide reliability consistent with:

a. operational requirements for the control and confinement of radioactive materials; and

b. potential hazards under all probable conditions.

The services and utilities that are important to the continuity of essential plant functions shall be designed to the integrity level as that of the function they serve. Some service or utility systems are connected to other systems or structures that are essential in preventing the release of radioactive materials; such service or utility systems shall be designed so that if they fail, connecting systems will not be damaged.

13.5.1 Ventilation Systems

Ventilation systems include the supply and exhaust systems and the associated ductwork; fans; air cleaning, tempering or humidity control devices; and associated monitoring instrumentation and controls required to confine radioactive materials within the ventilation system and to remove radioactive material from air streams released from the plant.

Design Objectives

The ventilation system shall be designed to provide confinement of dispersible radioactive material within prescribed areas of the facility. It shall also be designed to limit airborne concentrations of radioactive material in occupied areas of the facility and in effluents that reach the public, to less than the applicable concentration guides and ALARA.

The ventilation system, serving as an engineered safety feature, shall be designed to remain operational or fail safely under all operational and credible upset conditions. The failure of any single component shall not compromise the ability of the system to maintain confinement of radioactive materials and control their release to the environment. Specific response requirements of the system and its components shall be established in a safety analysis.

Air Flow and Balance

Design of ventilation systems should assure that air flows are, under all normal conditions, toward areas of progressively higher radioactive contamination. Air handling equipment should be sized sufficiently conservatively that minor upsets in air flow balance (e.g., improper use of an air lock, occurrence of a credible breach in a confinement barrier) do not result in flow reversals. HEPA filters should be provided at ventilation inlets in confinement area barriers to prevent movement of contamination from areas of higher levels to areas of lower levels in case of a flow reversal. Ventilation system balancing should assure that the building air pressure is always negative with respect to the outside atmosphere.

In facilities where floor fans are used for personnel comfort, special precautions should be exercised to assure than air balance for contamination control is maintained.

A minimum of two negative-pressure zones should exist within a process building. The first, in the process confinement system, should serve the spaces within the enclosures, conveyors, and transfer boxes, and other spaces that may contain uranium during the course of normal operations. The second should serve the process areas and other potentially contaminated areas adjacent to the process confinement system. Controlled areas contiguous to process areas and potentially free of contamination will constitute a third zone.

A negative pressure differential, negative with respect to the room, should be maintained in all process confinement systems. A negative pressure differential should be maintained between process and controlled areas and between controlled areas and uncontrolled areas. Air locks between zones should be provided where necessary to insure maintaining proper differential pressures.

Sufficient redundancy and/or spare capacity should be provided to assure adequate ventilation during normal and DBA conditions. Failure of a single component or control function shall not compromise minimum adequate ventilation.

Design of the system should include an analysis to ensure that the ventilation system is capable of operating under DBF conditions. It should be designed to assure, to the maximum extent practicable, that the products of combustion are not spread beyond the room of origin unless directed through appropriate ventilation channels. The exhaust system should be designed to provide cleanup of radioactivity and noxious chemicals from the discharge air and to safely handle the products of combustion.

Provisions may be made for independent shutdown of ventilation systems where this could possibly be an advantage to operations, maintenance, or emergency procedures such as fire fighting. In assessing the desirability of providing for shutdown of a ventilation system under such conditions, full consideration should be given to all possible effects of the shutdown on air flows in other, interfacing ventilation systems. It may be more appropriate to provide for drastically reduced flow rather than system shutdown. Positive means of controlling backflow of air which might transport contamination should be provided. The ventilation system and associated fire suppression system should be designed for fail-safe operation.

The ventilation system should be appropriately instrumented and alarmed, with readouts in control areas located in the utilities services area for the facility, to report and record its behavior. A listing and the function of required and recommended instrumentation are given in ANSI N509-1980, Table 4-1 (ANSI 1980).

Penetrations of the building for ventilation ducts should be kept to a minimum and should be designed to protect the critical systems against a DBA. If the barrier around the process area is the building's outside wall, no penetrations should be permitted.

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Air Supply

Supply air should be appropriately filtered and conditioned commensurate with operational requirements and with levels recommended for comfort.

The ventilation rate in process areas, where uncontained radioactive materials are handled, should be a minimum of ten air changes per hr. A minimum of seven air changes per hour should be provided in support facilities within the process area. Adequate air filters should be used at the intake of the ventilation supply system to minimize dust in the process area and to reduce dust loading on HEPA filters.

A downward flow pattern should be provided at operating stations to direct air from any potential leak point down and away from the operator's face. The distribution of inlet air through a number of small ports or by slot-type distributors decreases the possible occurrence of "dead spots" with little air circulation.

Enclosures, conveyors, and other systems requiring a controlled atmosphere may be equipped for recirculation of the atmosphere. All parts of the recirculation system should operate at pressures negative with respect to the room. Process enclosures that use normal air may receive their air supply from the room through dust-stop and HEPA filters mounted on the enclosure.

Exhaust Systems

The number of required exhaust filtration stages from any area of the facility shall be determined by analysis to limit quantities and concentrations of airborne radioactive or toxic material released to the environment during normal and accident conditions, in conformance with applicable standards, policies and guidelines. In general, each exhaust filter system for room and process air should consist of a minimum of one

HEPA filter for room air and two HEPA filters in series for enclosure or hood exhaust air. Baffles or flow distributors should be provided to distribute incoming air more or less uniformly over the entire face of the filter bank. The filtration system should be designed to allow reliable in-place testing of the HEPA filters and ease of filter replacement to the extent practicable.

The exhaust system for an enclosure or hood should be separate from the exhaust system for room air. Exhaust air should be drawn through a HEPA filter at the enclosure or hood to maintain primary control at that point and minimize contamination of ductwork. Additional HEPA filters in series should be separated at a sufficient distance to permit in-place testing of each stage of the filters.

Integral fire-suppression equipment should be provided as needed within each ventilation system to assure that a DBF could not degrade the integrity of the high efficiency air cleaning system. Where appropriate, a cool-down chamber with water sprinklers, a prefilter-demister, and a spark arrestor screen should precede the first stage of the final HEPA filtration system. This cooling spray should be automatically actuated by appropriate temperature and smoke sensing devices as determined by the accident analysis.

All potentially contaminated air should be exhausted through a common stack if possible. Monitoring and sampling capability shall be provided on exhaust stacks that may contain radioactive or toxic materials. The ventilation exhaust stack should be located as far from any air intake as reasonably possible. Design criteria for effluent monitoring and sampling and elements for consideration in effluent radioactivity measurement are described in DOE/EP-0096, A Guide for Effluent Radiological Measurements at DOE Installations (DOE 1983).

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System Testing and Control

The ventilation system is considered essential to the safety of plant employees and the public and should be designed in accordance with ANSI/ANSE N509-1980. Safety analyses should establish the minimum acceptable response requirements for the ventilation system, its components, and instruments and controls under normal operations, anticipated operational occurrences, and accident conditions. These requirements should determine system and component design characteristics such as installation of standby spare units, provision of emergency power for fans, installation of tornado dampers, seismic qualification of filter units, fail-safe valve positioners, etc.

The system should be able to operate for reasonably long periods without requiring shutdown for servicing or filter replacement. The system's effectiveness should be assessable during operation by means of installed testing and measurement devices, including installed spares for critical components.

Air-cleaning systems should be designed for the convenient, repetitive, and reliable in-place testing of each stage of the system in accordance with ANSI Standard N510. Provisions for in-place testing should include aerosol injection ports, sampling ports, and connecting and bypass ductwork. Independent inspection and testing of HEPA filters prior to installation shall be performed by DOE approved organizations listed in DOE order 6430.1A. Each filter bank should also be tested upon installation and annually thereafter any anytime when conditions have developed that may have damaged the filter, i.e., pressure drop, overpressure, spray, etc. The filter or filter bank should demonstrate a particle removal efficiency of at least 95% for all measurable particles, on a count basis, in accordance with CS-1, Standard for HEPA Filters.

Those portions of the ventilation system essential to preventing releases of radiotoxic materials should continue to function (or

automatically change to a safe failure mode) in the event of abnormal or DBA conditions. The ventilation system fans should produce a maximum exhaust rate greater than the maximum supply rate. If a system fails, exhaust control dampers should fail to the open position, while the supply control dampers should fail to their preset closed position. Supply fans should automatically cut out when exhaust fan capacity is not sufficient to maintain the proper pressure differential. Alarms should be provided to signal the loss of fans or improper air balance. System components or devices that must function under emergency conditions should be able to be tested periodically, preferably without interruption of operations.

Appropriate surveillance instrumentation and manual system operation controls should be provided at one common location. In addition, surveillance instrumentation should be located in an external or protected area that would be accessible during and after all types of DBA events.

13.5.2 Electrical Power

Both normal and emergency power supplies must be available to a uranium facility to assure that critical systems can continue to operate under both normal and accident conditions.

Normal Power

A uranium facility's normal electrical power needs should be met by two primary feeders. The preferred primary feeder provides basic service to the facility and should consist of a radial feeder connected directly to the main substation serving the area. To minimize power outages, this feeder should be an express feeder and have no other loads connected to it.

The alternate primary feeder provides back-up to the preferred primary feeder and should be in ready standby for use by automatic transfer if there's a forced outage or planned maintenance of the preferred primary feeder. The alternate primary feeder should also be a radial feeder

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connected directly to a substation and have no other loads connected to it. To minimize simultaneous outages of the preferred and alternate primary feeders due to lightening or other physical damage, the two feeders should have maximum physical separation.

Emergency Power

The facility should be provided with a reliable, local source of emergency power if both primary sources fail. The emergency power source should be completely independent of both the preferred and alternate primary feeders. The emergency power should be generated on-site by turbines or diesel generators with automatic starting and switch-over equipment. The emergency system should be physically separated from the normal power systems except at the automatic transfer switch so that any electrical or mechanical breakdown of the normal power system will not render the emergency system inoperative.

The time lag between electrical power failure and the resumption of emergency power should not exceed 20 s, and the emergency system should remain energized for at least 5 min., and preferably longer, after the restoration of primary power. The emergency power sources should have sufficient capacity and sufficient fuel supplies stored on-site to maintain the integrity of all critical building systems for 48 hrs. The emergency power system should be able to carry selected loads such as air exhaust and supply systems, fire detection and suppression systems, related instrumentation and control functions, necessary criticality and radiation monitoring instrumentation, certain processing equipment, and any other essential building systems.

Non-critical uses of emergency power should be avoided.

13.5.3 Water Supply

Water storage tanks with multiple or back-up supplies should be provided to simultaneously meet the needs of fire protection, process and potable uses.

The design of the water supply system shall provide water for fire fighting and automatic sprinkler systems criteria described in DOE Order 6430.1A, and Factory Mutual and National Fire Protection Association Standards. Fire protection water supply and distribution design required for critical item protection should assure continuity of protection in the event of a DBA.

Potable water should be distributed to drinking fountains, eye wash fountains, showers, emergency showers, lavatories, toilets, and non-contaminated laboratories. The potable water system must be protected against contamination in accordance with Chapter 4 of DOE 6430.1A. Water mains should not pass through process or controlled areas. Branch lines may be permitted only in process areas for safety showers and fire protection sprinkler systems. Drinking fountains may be located in controlled areas, adjacent to the process areas, where contamination is not likely to occur.

The facility water system should be isolated from the primary water mains by an air gap to prevent any possibility of contamination of public water supplies. If an air gap is not possible, reduced pressure type of backflow prevention devices meeting the requirements of the American Water Works Association C506-78-1983 shall be used.

13.5.4 Fire Protection

The facility must be designed to prevent, detect, suppress, and confine fires and products of combustion.

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Each area in the plant building must be equipped with fire detection devices best suited for that area, as described in DOE Order 6430.1, Chapter X, and in National Fire Codes Titles 71 and 72A-72D. All equipment must be approved by a recognized testing laboratory. Consideration should be given to the spacing, sensitivity, and location of the detectors to assure rapid response.

Most areas of the plant shall have automatic fire-suppression systems. These systems and the facility's structural design must meet, as a minimum, the requirements of NFPA's National Fire Codes, and DOE Orders 6430.1A, Chapters X and XXI and Chapter VII of DOE 5480.1B, and 5632.4. The minimum requirements include the provisions of completely automatic sprinkler systems or equivalent coverage throughout the facility of fire control measures for special hazards where such hazards exist. An assured water supply, adequate for fire-fighting and fire suppression needs over a 4-hour period, must be provided.

Fuels and combustible materials should be stored at a central facility remote from the uranium processing building(s). Piped natural gas should not be provided to uranium process or storage areas. Separate bottles gas systems may be required.

The facility's ventilation system should be designed to withstand any credible fire or explosion. It should be constructed of fire-resistant materials and have fire-detection and fire-suppression equipment, including heat and smoke detectors, alarms, fire doors and dampers, and heat removal systems. The final filter bank of the building's air exhaust system should be protected from damage by hot gases, burning debris, or fire-suppression agents that may be carried through the exhaust ducts during a 4-hour fire.

Overpressure protection may need to be considered for critical items such as certain enclosures, cells and ventilation ducts.

13.5.5 Waste

Waste from uranium handling facilities include both radioactive and non-radioactive materials, and will be in the form of liquid or gaseous effluents, or solids packaged for disposal. A principal design objective for the process systems should be to minimize the production of wastes at the source. A principal design objective for the waste management systems should be to provide facilities and equipment to handle those wastes safely and effectively. The design of the facility shall limit the release of radioactive materials to the environment to less than the DOE and EPA regulations and ALARA. Emphasis should be placed on reducing total quantities of effluents (both radioactive and non-radioactive) released to the environment.

Sanitary Waste

Sanitary wastes include the non-radioactive wastes usually found at a facility, e.g., discharges from non-contaminated chemical laboratories, showers, and lavatories. The sanitary waste system and the uranium handling area where radioactive material could enter the system. Sanitary sewers should discharge into an on-site, approved sanitary sewage treatment system. Current federal, state, and local codes regarding the discharge of sanitary wastes shall be met. The sanitary wastes should be monitored for radioactivity.

Potentially Contaminated Wastes

Potentially contaminated wastes include process coolant water blowdown from heating and cooling systems, process steam condensates, and discharges from mop sinks and personnel decontamination sinks and showers. Natural runoff from roofs of process building may be included in this category. Sufficient holdup capacity must be provided so that wastes can be retained until they are sampled, analyzed and shown to be within acceptable limits for release. Holdup capacity should also be provided for water collected from fire fighting activities including sprinkler activation.

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Potentially contaminated liquid wastes shall be sampled prior to discharge to the environs. Batch sampling and analysis of liquid waste tank may be used.

If liquid waste is discharged to the environs, the effluent concentrations of uranium shall not exceed the Radioactivity Concentration Guide (RCG), of DOE 5480.11, for uncontrolled areas measured at the point of discharge during normal operation and shall be ALARA.

Contaminated Wastes

Any contaminated waste, solid, liquid or gaseous, that does not meet the criteria for release, shall be held on-site and decontaminated to release limits or disposed of as radioactive waste.

13.6 Special Systems and Equipment

Special systems and equipment shall be incorporated in uranium facilities to insure safety of the worker and protection of the public. As a minimum, the following systems should be included.

13.6.1 Air Sampling and Monitoring

Airborne uranium activity measurements can be made using either air sampling or continuous air monitoring. The use of both types may be required for an effective uranium air monitoring program. An air sampling system may use a central building vacuum system with the sample heads located near positions frequently occupied by operating personnel. Alternately, individual portable samplers could be used. Continuous air monitoring systems should be located at areas which have a high potential for increases in airborne radioactivity to alert personnel of sudden increases in airborne radioactivity. The air sampling and monitoring systems should be located and operated such that the air concentrations measured are representative of that which could be breathed by workers.

Many facets of air sampling, such as isotopic composition, particle size, and solubility, that affect the potential intake by workers must be considered. Specific information may be found in Sections 5 and 6, and in the bibliographic materials for this manual.

CAMS shall be positioned and alarm settings established so that significant increases in airborne activity are detected and alarms are triggered to alert personnel to any changed conditions. DOE 5480.11 (DOE 1988) states that ambient air monitoring shall be performed in areas with the potential to exceed 10% of any derived air concentration (DAC). The air sampling and monitoring systems shall comply with ANSI N317-1980, Performance Criteria for Instrumentation Used for In-Plant Plutonium Monitoring (ANSI 1980c) and ANSI N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities (ANSI 1969). The applicable sections of draft ANSI Standards N42.17A-D9, Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions and N42.17B-D5, Performance Specifications for Health Physics Instrumentation - Occupational Airborne Radioactivity Monitoring Instrumentation shall also be considered.

Instrumentation for detecting UF_6 releases should be used in areas of potential airborne UF_6 releases and in conjunction with steam heating to detect UF_6 released to the steam condensate. This instrumentation should provide alarm and/or automatic protection functions, i.e. containment, emergency ventilation, or effluent cleanup.

All air and gaseous effluents that may contain radioactivity shall be exhausted through a ventilation system designed to remove particulates. All exhaust ducts and stacks, which may contain uranium contaminants, should be provided with two monitoring systems, a continuous type and a fixed sampler. These systems may be a combination unit. The air intake probes for sampling purposes should be designed for representative sampling. The fixed sampling system should contain the filter sample which would normally be the record sample. System design, location, installation, and operation should follow the guidance provided in ANSI

N-13.1 - 1969, ERDA 76-21, and DOE/EP-0096. Each of these systems should be connected to an emergency power supply in accordance with DOE 6430.1A. The requirements of Parts 1 and 2 of International Electrotechnical Commission (IEC) Publication 761, Equipment for Continuously Monitoring Radioactivity in Gaseous Effluents (IEC 1983) should also be met. Effluent monitoring systems should provide continuous recording of effluent concentrations and be equipped with alarm annunciation of excessive levels of radioactivity in the effluent discharge stream.

13.6.2 Breathing Air

For facility designs, confinement of airborne radioactive materials is the preferred method of preventing internal deposition of radioactive particulates.

A uranium facility should be provided with a system capable of supplying breathing air to a number of work stations in each occupied area where:

a. Gaseous or airborne radioactive material may exceed the concentration listed in DOE Order 5480.11.

b. Significant quantities of potentially dispersible uranium compounds are handled outside of containment devices.

c. Personnel may be required to enter areas containing large amounts of loose radioactive material for repair, maintenance, decontamination, or operation.

Breathing air supply systems must meet the requirements of ANSI Standard Z-88.2 and Part 1910 of Title 29 of the Code of Federal Regulations. Approved respiratory protection programs should be in place for the control and use of breathing air systems. The system must be monitored for air purity during operation and should be among those systems

supplied with emergency power. A supply reservoir should be provided and sized to permit the safe evacuation of users in the event of system shutdown or failure.

All equipment used for personnel breathing air systems must be approved by appropriate regulatory agencies. Hose couplings and manifolds must be designed to preclude the connection of respiratory equipment to any system other than the breathing air system. Similarly, special precautions must be taken to prevent the connection of other gas systems to the breathing air system.

13.6.3 Personnel Monitoring

Provisions should be made for personnel survey instruments at suitable locations within the process area. Survey instruments should be available for personnel exiting from enclosures and at exits from compartmented facilities. Survey instruments or monitoring instruments shall be available at contamination control change rooms and at exits from controlled areas.

13.6.4 Criticality Safety

Criticality alarm systems (gamma or neutron) must be provided in each area where an accidental criticality is possible. The requirements of DOE Order 5480.5, "Safety of Nuclear Facilities", ANSI/ANS 8.3, "Criticality Accident Alarm System", and ANSI/ANS 8.1-1983 "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors", and ANSI/ANS 8-19-1984 "ANS Administrative Procedures for Nuclear Criticality" shall be met.

13.6.5 Nuclear Accident Dosimeters

All DOE facilities which possess sufficient quantities and kinds of fissile material to potentially constitute a critical mass shall provide nuclear accident dosimetry (DOE 1988). The number of dosimeter units

needed and their placement will depend on the nature of the operation, structural design of the facility, and accessibility of areas to personnel. An analysis that demonstrates that the dosimeters and their placement will meet the criteria established in DOE 5480.5, shall be conducted and documented. The analysis shall include the number of units, their location, and effect of intervening shielding. Ease of recovery after a criticality event should be considered in the placement of the fixed units. Remote retrieval mechanisms may be appropriate.

13.6.6 Other Systems

Many systems employed within a uranium facility are not directly related to personnel safety and radiation protection. However, because of the special impact these systems may have in a nuclear facility, people responsible for personnel protection should be aware of them. Some examples are:

a. Process instrumentation and control indicators to monitor and maintain control over the process and to detect and indicate abnormal and accident conditions;

b. Surveillance systems to assure the integrity of all process piping, tanks, and other containment equipment, including that used for liquid effluents; and

c. Vacuum, air lift, or gravity systems to transfer toxic or corrosive liquids or slurries.

Special controls should be provided for flammable, toxic, and explosive gases, chemicals, and materials admitted to uranium handling areas. Gas and chemical storage facilities, including distribution piping systems, should conform to good design practice and applicable codes and standards. Consideration should be given to compatible groupings which, under accident conditions or leakage, would minimize any adverse combining

of materials. Means for the remote shutoff of piping should be provided. In addition, the following specifics should be followed:

a. Nonflammable hydraulic and lubricating fluids should be used in the uranium handling area.

b. Protective barriers should be provided around high-pressure or other potentially dangerous systems.

c. Incompatible chemicals, materials and processes should be isolated.

d. Properly vented, pressurized gas lines should be used in the uranium handling areas.

13.6.7 Monitoring and Alarms

Pressure-sensing instrumentation provided for heating systems used on UF_6 cylinders and cold traps should provide both alarm and visual display functions.

At least two separate means should be used for determining the quantity of UF_6 loaded into cylinders or cold traps before applying heat to them. "Real time" quantification methods are preferred, such as load cells, mechanical scales, or flow integration. Quantification instrumentation should provide an alarm function when preset limits are exceeded.

In order that abnormal conditions may be correctly interpreted and remedial action taken promptly, all monitoring system readouts and alarm indicators that relate to personnel safety or the integrity of a building should be centralized at a location that is continuously manned and has guaranteed accessibility. The inclusion of the following specific alarms and signals should be considered:

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a. Fire, criticality, evacuation, and security alarms.

b. Gaseous and liquid waste monitors.

c. Ventilation system performance monitors for airflow and pressure differential.

d. Room air monitors.

e. Process monitors for flow, pressure, temperature, and other process parameters that have an impact on safety.

f. Power monitors for power failure or loss of power to critical fans and pumps.

Essential monitoring and alarm systems must be supplied with emergency power so that if normal power fails, they will remain functional at all times. Reliability should be ensured by designed features such as redundant circuits and instruments that perform self-checks and are tamper-proof. All monitoring, surveillance, and alarm systems should be tested periodically.

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