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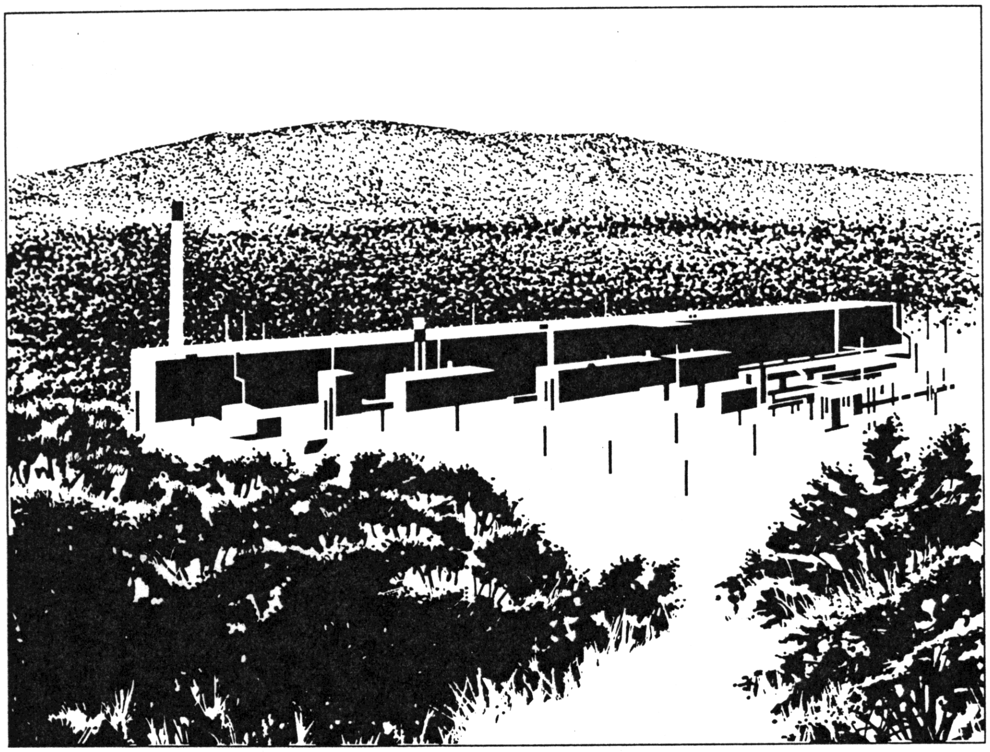
Department of Energy

Health Physics Manual of Good Practices for Plutonium Facilities



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May 1988

Prepared for the U.S. Department of Energy
Assistant Secretary for Environment, Safety, and Health
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute

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HEALTH PHYSICS MANUAL OF GOOD PRACTICES FOR PLUTONIUM FACILITIES

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FOREWORD

Since its discovery by Seaborg, McMillan, Wahl, and Kennedy in the winter of 1940-1941, plutonium has become a principal component of the United States' nuclear defense program. The manufacture and use of plutonium is a very complex process--so complex that it has the distinction of being one of the most, if not the most, studied of all the elements. Much has been said about plutonium. Numerous volumes have been written about it. It is feared by some and respected by all. Some have incorrectly labeled it the most toxic element known. Its toxicity and, more importantly, its radiotoxicity are the reasons that this manual of good practices was originally compiled and has now been revised.

This manual was originally issued in 1977. Since then, much new information has been learned about the properties of plutonium, new processes have been developed, and significant additional experience has been collected. National and international organizations have issued new and revised recommendations and standards for radiation protection and for the design and operation of nuclear facilities including those issued by the U.S. Department of Energy (DOE) to further enhance the safety and control of nuclear facility operation. The revision of this manual is an attempt to incorporate the new information and to make the manual more current for the user.

The manual is directed primarily to facilities whose sole purpose is the handling of large quantities of plutonium for military or industrial use. It is not intended for use by facilities that are engaged in reactor or chemical separation operations nor for partial or occasional use by analytical laboratories. While these facilities would find the manual beneficial, it would be incomplete for their needs.

The manual focuses on the radiation protection aspects of handling plutonium and is based on the experiences of DOE contractors. It includes criteria and guidance in the design, construction, and operational areas, because, as is well known in the nuclear industry, joint efforts by the original manual, "No amount of good management can correct for basic problems, including safety-related problems caused by poor design; nor can an excellent design eliminate problems caused by poor management or the poor work habits of the staff. However, good design, work habits, and management are only tools: safety and good work practices ultimately become the personal responsibility of the individual."

In the manual, the terms "controlled area" and "uncontrolled area" refer to radiologically controlled and uncontrolled areas. These terms are not to be confused with formal definitions of controlled and uncontrolled areas. In addition, the terms "dose rate" and "exposure rate" are used interchangably throughout the manual for ease of reference and are not always used according to their formal definitions.

Nuclear criticality safety is not discussed in detail. Information such as maintaining safe conditions for processing, handling, storing, or transferring plutonium materials is beyond the scope of the manual. A discussion of the health physicist's role in criticality safety is presented.

Of the practices presented in this manual only those taken directly from regulatory documents are mandatory. The final authority is, of course, the regulations themselves, and they should be consulted for resolution of any questions. For the practices that are not required, several alternatives may be available to accomplish a specific task safely, and any one of them may be as acceptable as the one presented here. However, personnel who lack experience in health physics and radiation protection and who may be undertaking new responsibilities in plutonium facilities should find this manual extremely useful.

Edward J. Vallario, Acting Director
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SUMMARY

This good practices manual was originally issued by Pacific Northwest Laboratory (PNL) in 1977. It filled a need within the plutonium handling facilities by providing, in one document, technical information and day-to-day guidance for managers, health physicists, and employees engaged in the handling of plutonium. The increase in the experience base since that time and the value of the original manual prompted the U.S. Department of Energy (DOE) Office of Nuclear Safety to support this revision of the manual.

This manual consists of six sections: Properties of Plutonium, Siting of Plutonium Facilities, Facility Design, Radiation Protection, Emergency Preparedness, and Decontamination and Decommissioning. While not the final authority, the manual is an assemblage of information, rules of thumb, regulations, and good practices to assist those who are intimately involved in plutonium operations.

An in-depth understanding of the nuclear, physical, chemical, and biological properties of plutonium is important in establishing a viable radiation protection and control program at a plutonium facility. These properties of plutonium provide the basis and perspective necessary for appreciating the quality of control needed in handling and processing the material. Guidance in selecting the location of a new plutonium facility may not be directly useful to most readers. However, it provides a perspective for the development and implementation of the environmental surveillance program and the in-plant controls required to ensure that the facility is and remains a good neighbor.

The criteria, guidance, and good practices for the design of a plutonium facility are also applicable to the operation and modification of existing facilities. The design activity provides many opportunities for implementation of features to promote more effective protection and control. The application of "as low as reasonably achievable" (ALARA) principles and optimization analyses are generally most cost-effective during the design phase.

The successful operation of a plutonium facility requires a high-quality radiation protection and contamination control program. Elements essential to such a program include personnel training, appropriate instrumentation,

radiation monitoring and surveillance practices, internal and external exposure control programs, analytical capability, administrative controls and records systems, and an independent quality assurance overview to ensure that it all gets done correctly.

An emergency plan is also required to ensure that the personnel operating a plutonium facility are prepared to respond promptly and effectively to any emergency situation. The plan must ensure that procedures are in place for protecting employee and public health and the facility, both in response to the immediate emergency and in the recovery actions.

When a plutonium facility is no longer useful, provisions must be made for alternative uses, long-term institutional control, or for return of the facility to unrestricted use. While it is recognized that current regulations lack definitive guidance, the discussion of decontamination and decommissioning alerts the plutonium operators to consider specific conditions and practices during current activities that will make the process more effective, whatever the eventual allowable release criteria might be.

ACKNOWLEDGMENTS

The authors thank the many people who provided technical comments and support for this manual. Special thanks to E. J. Vallario, Office of Nuclear Safety, U.S. Department of Energy (DOE), for his continued support and guidance. Thanks to the many members of the DOE family, who provided information, comments, and direction to make this manual more useful; and to the technical review panelists, who provided technical input and reviewed this document for accuracy, appropriateness, and applicability of its content. Thanks also to the members of the DOE Expert Group on Internal Dosimetry for their painstaking technical support in developing Section 4.6, "Internal Exposure Control."

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ACRONYMS

AC	alternating current
ACGIH	American Conference of Governmental Industrial Hygienists
ALARA	as low as reasonably achievable
ALI	annual limit on intake
AMS	Aerial Measuring System
AMAD	activity median aerodynamic diameter
AMDA	acceptable minimum detectable activity
ANSI	American National Standards Institute
ARAC	Atmospheric Release Advisory Capability
ARG	Accident Response Group
ASME	American Society of Mechanical Engineers
ATRAP	United States Air Force Transportable RADIAC Package
AWWA	American Water Works Association
CAM	continuous air monitor
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
D&D	decontamination and decommissioning
DAC	derived air concentration
DBA	design-basis accident
DBE	design-basis earthquake
DBF	design-basis fire
DOE	U.S. Department of Energy
DTPA	diethylenetriaminepentaacetic acid
EA	environmental assessment
EAL	emergency action level
ECC	Emergency Control Center
EDTA	ethylenediaminetetracetic acid
EIS	environmental impact statement
ECS	Emergency Control Station
EOC	Emergency Operations Center
EPA	U.S. Environmental Protection Agency
EPZ	emergency planning zone

ERDA	Energy, Research, and Development Administration
FEMA	Federal Emergency Management Agency
GI	gastrointestinal
GM	Geiger-Mueller
HEHF	Hanford Environmental Health Foundation, Richland, Washington
HEPA	high-efficiency particulate air
HQ	headquarters
HPSSC	Health Physics Society Standards Committee
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IEC	International Electrotechnical Commission
IND	Investigational New Drug
ISO	International Standards Organization
JNACC	Joint Nuclear Accident Coordinating Center
LANL	Los Alamos National Laboratory
LIS	laser isotope separation
LLNL	Lawrence Livermore National Laboratory
LMFBR	liquid metal fast breeder reactor
LWR	light water reactor
MAD	mean aerodynamic diameter
MDA	minimum detectable amount (activity)
MPBB	maximum permissible body burden
MPC	maximum permissible concentration
MPOB	maximum permissible organ burden
NAWAS	National Warning System
NBS	National Bureau of Standards
NCRP	National Council on Radiation Protection and Measurements
NEPA	National Environmental Policy Act
NEST	Nuclear Emergency Search Team
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
OBE	operating basis earthquake

ORNL	Oak Ridge National Laboratory
PNAD	personnel nuclear accident dosimeters
PMF	probable maximum flood
PRR	protective response recommendation
PSAR	preliminary safety analysis report
PVC	polyvinyl chloride
PVDF	polyvinylidene fluoride
QA	quality assurance
RADCON	U.S. Army Radiological Control Team
RAMS	remote area monitoring systems
RAMT	U.S. Army Radiological Advisory Medical Team
RCG	radioactivity concentration guide
RCRA	Resource Conservation and Recovery Act
R&D	research and development
REACTS	Radiation Emergency Assistance Center/Training Site
RPT	radiation protection technologist
RWP	radiation work procedure
SAR	safety analysis report
SDD	system design description
SNAP	Space Nuclear Auxiliary Power (system)
SNM	special nuclear material
TED	track-etch dosimeter
TEPC	tissue-equivalent proportional counter
TGLD	Task Group on Lung Dynamics
TLD	thermoluminescent dosimeter
USGS	United States Geological Service
USTR	United States Transuranium Registry, Richland, Washington
WEP	water-extended polyester
WG	water gage

DEFINITIONS

Absorbed Dose is the energy imparted to matter by ionizing radiation per unit mass of irradiated material at the place of interest in that material. The absorbed dose is expressed in units of rad (gray).

Annual Effective Dose Equivalent (H_E) includes the dose equivalent from both external and internal irradiation and is defined by $\sum_T w_T H_T$ where H_T is the dose equivalent in tissue T and w_T is the weighting factor representing the ratio of the risk arising from irradiation of tissue T to the total risk when the whole body is irradiated uniformly. The annual effective dose equivalent is expressed in units of rem (sievert).

Annual Limit on Intake is the activity of radionuclide which, if taken alone, would irradiate a person, represented by Reference Man (ICRP Publication 23).[a] to the limiting value for control of the workplace.

Committed Dose Equivalent ($H_{T,50}$) is the calculated dose equivalent projected to be received by a tissue or organ over a 50-year period after an intake of the radionuclide into the body. It does not include contributions from external dose. Committed dose equivalent is expressed in units of rem (sievert).

Committed Effective Dose Equivalent ($H_{E,50}$) is the sum of the committed dose equivalents to various tissues in the body, each multiplied by its weighting factor $\sum_T w_T H_{T,50}$.

Derived Air Concentration is the concentration in air obtained by dividing ALI for any given radionuclide by the volume of air breathed by an average worker during a working year (2.4×10^3 m^3). Numerical quantities are given in DOE 5480.11.[b]

(a) International Commission on Radiological Protection (ICRP). 1974. Report of the Task Group on Reference Man. ICRP Publication 23, Pergamon Press, New York, New York.

(b) U.S. Department of Energy (DOE). 1987. Radiation Protection for Occupational Workers. DOE 5480.11, U.S. Department of Energy, Washington, D.C.

Dose Equivalent (H_T) is the product of absorbed dose (D) in rad (gray) in tissue, a quality factor (Q), and other modifying factors (N). Dose equivalent (H_T) is expressed in units of rem (sievert).

Effective Dose Equivalent (H_E) includes the dose equivalent from both external and internal irradiation and is defined by $\sum_T w_T H_T$, where H_T is the dose equivalent in tissue and w_T is the weighting factor representing the ratio of risk arising from irradiation of tissue T to the total risk when the whole body is irradiated uniformly. Effective dose equivalent is expressed in units of rem (sievert).

Weighting Factor (w_T) is used in the calculation of annual and committed effective dose equivalent to equate the risk arising from the irradiation to tissue T to the total risk when the whole body is uniformly irradiated. The weighting factors are:

Organ or Tissue	Weighting Factor
Gonads	0.25
Breasts	0.15
Red Bone Marrow	0.12
Lungs	0.12
Thyroid	0.03
Bone Surfaces	0.03
Remainder[a]	0.30

(a) "Remainder" means the five other organs with the highest dose (i.e., liver, kidney, spleen, thymus, adrenals, pancreas, stomach, small intestine or upper and lower large intestine but excluding skin, lens of the eye, and extremities). The weighting factor for each such organ is 0.06.

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SECTION 1.0

PROPERTIES OF PLUTONIUM

1.0 PROPERTIES OF PLUTONIUM

This section of the manual includes general information about the properties of plutonium, and more detailed information about specific topics is included later in the manual. This section includes information on the following:

- nuclear properties, including the types of radiation emitted, nuclear decay schemes, and specific activities
- physical and chemical properties, including allotropic forms of plutonium, common chemical species and chemical reactivity, and problems of oxidation and pyrophoricity
- biological properties, including modes of entry into the body, and distribution within the body
- manufacture and uses of plutonium, including the present uses for nuclear weapons, nuclear reactors, isotopic heat sources, and new technologies being introduced for isotopic separation of plutonium.

1.1 NUCLEAR PROPERTIES

An important nuclear property of plutonium that makes it useful to man is that it is fissile, i.e., atoms of plutonium split upon exposure to thermal or fast neutrons. Chemical reactions can release a few electron volts of energy per atom, but when a plutonium nucleus splits it releases about 200 MeV of energy and between 2 and 3 neutrons. This release of energy makes plutonium useful for nuclear weapons and for reactor fuel. In fact, in light water reactors (LWRs) much of the power originates from the fission of ^{239}Pu which is produced by neutron capture in ^{238}U. The isotope ^{238}Pu is also useful as a heat source.

All isotopes of plutonium are radioactive. Isotopes with even mass numbers (except mass number 246) are primarily alpha emitters. Isotopes of the mass numbers 232, 233, 234, 235, and 237 also decay by electron capture; isotopes of the mass numbers 241, 243, 245, and 246 decay by beta emission.

Many of the alpha-emitting isotopes also fission spontaneously and emit neutrons. All of the particle emissions are accompanied by x-ray and gamma-ray emissions over a wide range of energies.

A review of the nuclear properties of plutonium (cross sections, nuclear levels, half-lives, fission yields, etc.) can be found in Volume 1 of the Plutonium Handbook: A Guide to the Technology (Wick 1967) and in American National Standards Institute (ANSI) Standard N317-1980, Performance Criteria for Instrumentation Used for In-Plant Plutonium Monitoring (ANSI 1980).

1.1.1 Decay Schemes

Table 1.1 shows the decay modes of some important plutonium isotopes and decay products. For brevity, only the most abundant radiations have been included in this table; more detailed information can be found in papers by Gunnink and Morrow (1967), Klein (1971), and in Publication 38 of the International Commission on Radiological Protection (ICRP) (ICRP 1983). Most of the isotopes are strong alpha emitters making alpha heating a problem for the storage and handling of large amounts of plutonium. The specific activities and decay heats for selected plutonium isotopes and decay products are given in Table 1.2. Kilogram quantities of ^{239}Pu or gram quantities of ^{238}Pu can generate enough heat to melt plastic bags. Sources of ^{238}Pu must be handled with insulated gloves, and special precautions must be taken to ensure a good thermal heat sink during storage.

The plutonium isotopes emit relatively few high-energy gamma rays and even kilogram quantities can be processed without serious gamma exposure problems. In some instances the decay products may become significant in radiation protection and metallurgy. For instance, the isotope ^{236}Pu often constitutes less than 1% of plutonium and is often ignored in dose calculations. However, if the plutonium is shielded by greater than 1 cm of lead or steel, the decay products of ^{236}Pu may be the largest contributors to exposure. The decay product ^{208}Tl emits a highly penetrating gamma ray with an energy of 2.615 MeV. In plutonium that contains a few weight percent ^{241}Pu, the ^{241}Am decay product is important because it emits a large number of 60-keV photons, which can be a significant source of exposure to the hands and forearms when handling plutonium in glove boxes. Also, ^{241}Am can contribute

TABLE 1.1. Radioactive Decay Properties of Selected Plutonium Isotopes and Decay Products Excluding Spontaneous Fission[a]

		Mode of Decay			X Ray		Gamma Ray	
Isotope	Half-Life	Particle	Energy, MeV	Yield, %	Energy, MeV	Yield, %	Energy, MeV	Yield, %
^{236}Pu	2.85 yr	α	5.77	68.0	L's 0.011-0.021[b]	10[c]	0.047	6.9×10^{-2}
			5.72	31.8			0.11	1.2×10^{-2}
^{238}Pu	87.7 yr	α	5.50	71.6	L's 0.011-0.021	10.5[c]	0.0435	3.9×10^{-2}
		α	5.46	28.3			0.0999	7.5×10^{-3}
^{239}Pu	2.41×10^4 yr	α	5.156	73.8	L's 0.0116-0.0215	4.8[c]	0.099	1.3×10^{-3}
		α	5.143	15.2			0.129	6.2×10^{-3}
		α	5.105	10.7			0.375	1.6×10^{-3}
							0.414	1.5×10^{-3}
^{240}Pu	6.54×10^3 yr	α	5.168	73.4	L's 0.0115-0.0215	10.8[c]	0.0452	4.5×10^{-2}
		α	5.124	26.5			0.104	7.0×10^{-3}
^{241}Pu	14.4 yr	β	0.0052[d]	100.00			0.077	2.20×10^{-5}
		α	4.897	2.04×10^{-3}			0.1037	1.01×10^{-4}
		α	4.854	2.97×10^{-4}			0.114	6.0×10^{-6}
		α	4.797-5.055	5% of 2.3×10^{-3}			0.149	1.9×10^{-4}
							0.160	6.71×10^{-6}
^{242}Pu	3.76×10^5 yr	α	4.901	77.5	L's 0.0116-0.0215	4.1[c]	0.0449	3.6×10^{-2}
		α	4.857	22.4			0.104	7.9×10^{-3}
^{241}Am	432 yr	α	5.486	85.2	L's 0.0119-0.0222	37.6[c]	0.0263	2.4
		α	5.443	12.8			0.0332	1.2×10^{-1}
		α	5.389	1.4			0.0595	35.7
^{237}U	6.75 d	β	0.039[e]	0.6	L's 0.0119-0.0206	62.9[c]	0.0264	2.2
		β	0.050[e]	3.0			0.065	1.2
		β	0.065[e]	52.2			0.165	1.8
		β	0.069[e]	44.2			0.208	21.7
							0.268	7.1×10^{-1}
							0.332	1.2
							0.335	9.5×10^{-2}
							0.369	4.7×10^{-2}
							0.371	1.1×10^{-1}

(a) Data from ICRP 38 (1983).
(b) L's = L x rays.
(c) Total for all x rays, value represents an average obtained from data at Pacific Northwest Laboratory, Lawrence Berkeley Laboratory, and Lawrence Livermore Laboratory.
(d) Average beta energy given; maximum beta energy is 0.0222 MeV.
(e) Average beta energy; maximum beta energy for ^{237}U is 0.248 MeV.

TABLE 1.2. Specific Activity and Decay Heats of Selected Transuranic Isotopes[a]

Isotope	Half-Life, yr	Specific Activity, Ci/g	Average Particle Energy per Disintegration, MeV	Decay Heat, W/g
^{236}Pu	2.85	α 534	α 5.75	18.2
^{238}Pu	87.7	α 17.1	α 5.49	0.567
^{239}Pu	2.407×10^{4}	α 6.22×10^{-2}	α 5.14	1.93×10^{-3}
^{240}Pu	6540	α 0.229	α 5.16	7.13×10^{-3}
^{241}Pu	14.4	α 2.52×10^{-3} β 103	α+β 5.37×10^{-3}	3.28×10^{-3}
^{242}Pu	3.76×10^{5}	α 3.93×10^{-3}	α 4.90	1.16×10^{-4}
^{232}U	72.0	α 21.5	α 5.31	0.690
^{233}U	1.59×10^{5}	α 9.75×10^{-3}	4.72	2.84×10^{-4}
^{234}U	2.45×10^{5}	α 6.29×10^{-3}	4.76	1.81×10^{-4}
^{235}U	7.04×10^{8}	α 2.17×10^{-6}	4.24	6.02×10^{-8}
^{236}U	2.34×10^{7}	α 6.5×10^{-5}	4.48	1.77×10^{-6}
^{238}U	4.47×10^{9}	α 3.38×10^{-7}	4.18	8.58×10^{-9}
^{237}Np	2.14×10^{6}	α 7.08×10^{-4}	4.76	2.08×10^{-5}
^{241}Am	432	α 3.43	5.37	0.115

(a) From ICRP 38 (1983).
(b) Include atomic recoil and low-energy x-ray production.

to neutron dose, as explained in the next subsection. Because of its importance to radiation exposure, the fractional amount of ^{241}Am produced by beta decay from ^{241}Pu is given as a function of time since chemical separation in Figure 1.1.

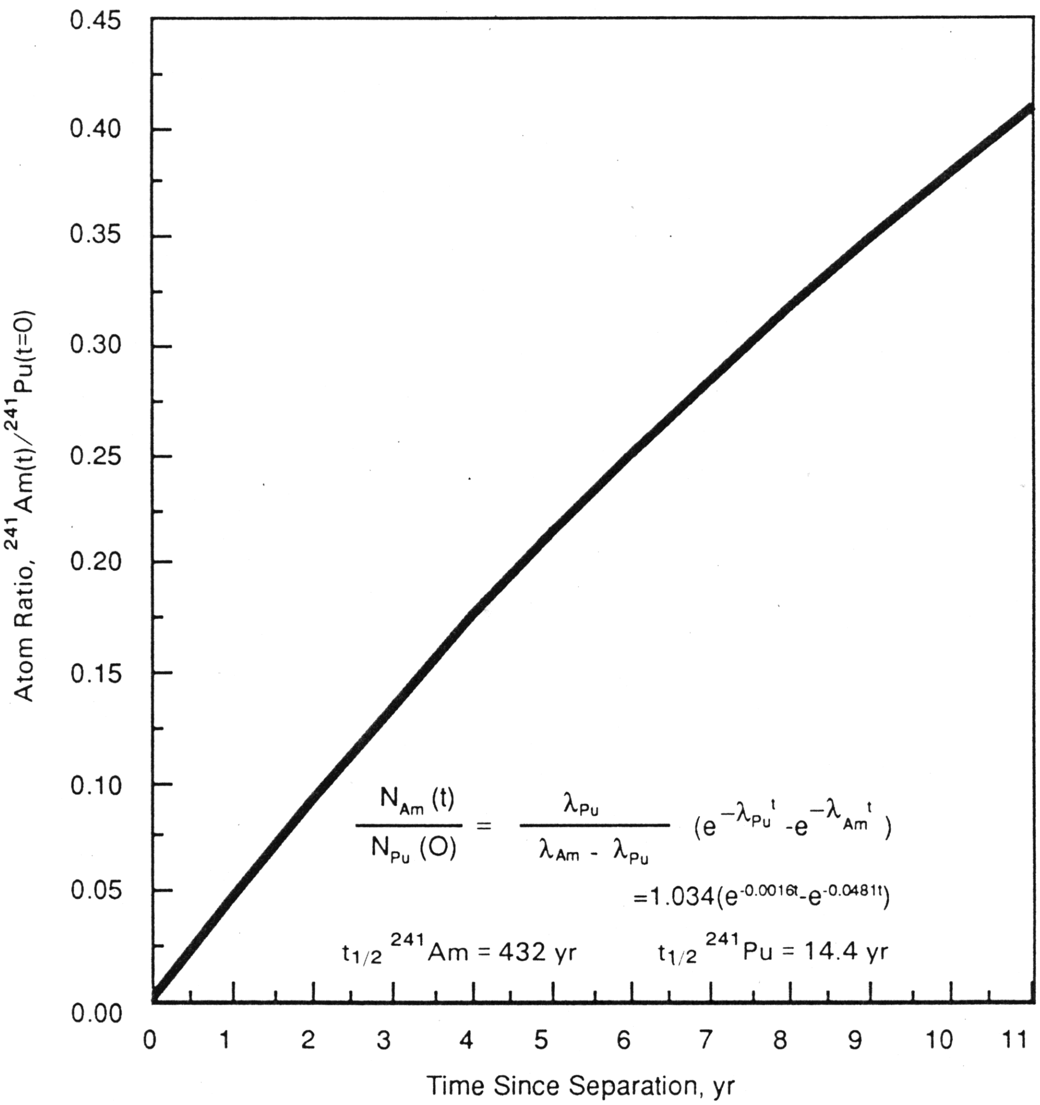

FIGURE 1.1. Atom Ratio of ^{241}Am to ^{241}Pu (T=0) Produced by the Beta Decay of ^{241}Pu as a Function of Time Since Chemical Separation

1.1.2 Neutron Yields and Spectra

Plutonium and plutonium compounds also emit neutrons from spontaneous fission and from alpha-neutron reactions with light elements. The spontaneous fission half-life and the neutron yields from spontaneous fission and alpha-neutron reactions are given in Section 4.8 for plutonium metal and plutonium compounds. The approximate neutron yield from a substance with a known isotopic composition can be determined by adding the contributions from each component. This procedure and its limitations are described in detail in Section 4.8.1, which discusses neutron dose equivalent rates.

The energy spectra of neutrons emitted by plutonium metal and plutonium compounds are given in Figure 1.2. Metallic plutonium emits neutrons with a Maxwellian energy distribution, with an average energy of about 1.9 MeV. Plutonium compounds and alloys also emit neutrons from alpha-neutron reactions, and these neutrons have significantly different energies: PuF_4 emits neutrons with an average energy of 1.3 MeV; 10% plutonium-aluminum alloys emit neutrons with an average energy of 1.6 MeV; and PuO_2 emits neutrons with an average energy of slightly more than 2 MeV.

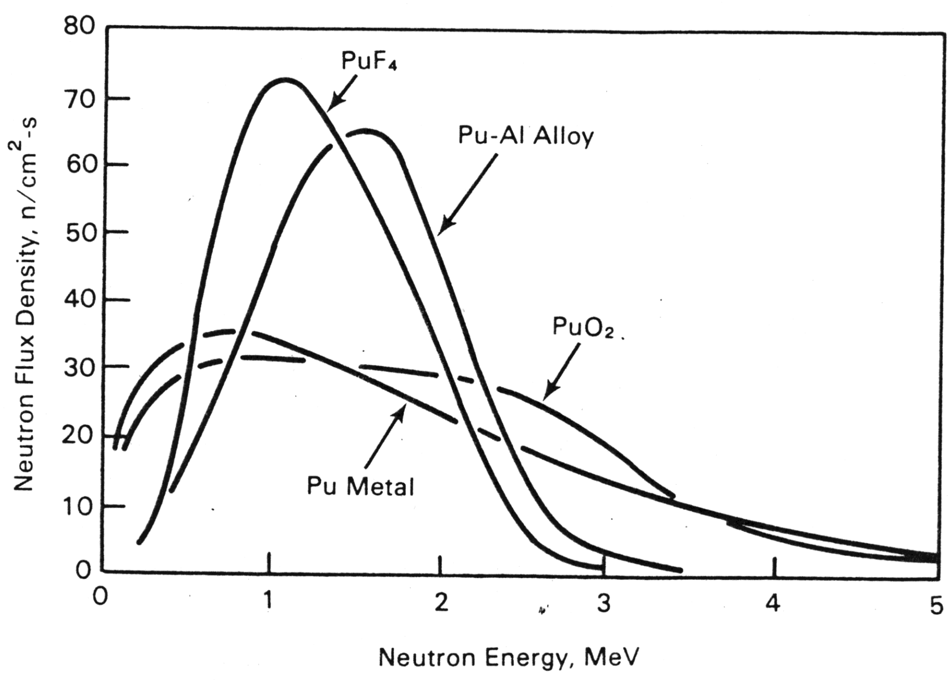

FIGURE 1.2. Plutonium Neutron Energy Spectra

1.2 PHYSICAL AND CHEMICAL PROPERTIES

Plutonium is a silvery-white metal, much like nickel in appearance. In moist air or moist argon, the metal rapidly oxidizes producing a mixture of oxides and hydrides. If exposed long enough, an olive-green powdery surface coating of PuO_2 is formed. With this coating the metal is pyrophoric, so plutonium metal is usually handled in an inert, dry atmosphere of nitrogen or argon. Oxygen retards the effects of moisture and acts as a passivating agent. Between 1% and 5% oxygen is added to a nitrogen atmosphere to passify the surface of metallic plutonium (Schnizlein and Fischer 1964; Wick 1967).

Plutonium metal has a low melting point (640°C) and an unusually high boiling point (3327°C). The metal exists in six allotropic forms, as indicated in Table 1.3. Two of the allotropic forms, δ and δ', contract upon heating; the other forms expand upon heating. At room temperature, plutonium exists in the α phase with a density of about 19.86 g/cm^3. Heating that is caused by high specific activity or machining operations can cause large changes in volume. For this reason, many forms of metallic plutonium are stabilized with gallium alloys in the δ phase, which has a density of about 15.75 g/cm^3 (Merz 1971).

At room temperature, the most stable oxide is PuO_2. Loose PuO_2 powder, as formed by calcination, usually has a density of about 2 g/cm^3. If the oxide is pressed and sintered into pellets, it may have a density of 10.3 to 11.0 g/cm^3.

TABLE 1.3. Allotropic Forms of Plutonium Metal

Phase	Stability Range, °C	Density, g/cm$^{3(a)}$
α	Stable below 115	19.86
β	~115 to 200	17.70
γ	~200 to 310	17.70
δ	310 to 452	15.92
δ'	452 to 480	16
ε	480 to 640	16.51

(a) Theoretical x-ray density. The actual density is slightly lower due to crystal lattice imperfection.

The chemistry of plutonium is complex and many different chemical species often coexist. Plutonium is the fifth element in the actinide series, which consists of elements with properties that stem from partial vacancies in the 5f electron shell. In general, there are four oxidation states: (III), (IV), (V), and (VI).

In aqueous solutions, Pu(III) is oxidized into Pu(IV), which is the most stable state. The compounds PuF_4, $Pu(IO_3)_4$, $Pu(OH)_4$, and plutonium oxalate are insoluble in water. The chlorides, nitrates, perchlorates, and sulfates are soluble in water. Plutonium (IV) ions complex readily with organic and inorganic compounds. The complicated chemistry of plutonium is discussed in detail in Section III of Volume I of the Plutonium Handbook: A Guide to the Technology (Wick 1967) and in Plutonium by Taube (1964).

Important compounds in the chemical processing of plutonium are PuF_4, $Pu(NO_3)_4$, and PuO_2. Plutonium fluoride is an important intermediate step in most conversion processes to produce metal. The usual form is PuF_4, which is a solid at room temperatures. The PuF_6 form is important because of its use in the fluoride volatility separation process at elevated temperatures, although PuF_6 is not chemically stable and readily decomposes. Plutonium nitrate is readily soluble in aqueous solutions and is an intermediate step in plutonium processing. Because it is chemically stable and relatively inert, PuO_2 is the preferred form for shipping and storing plutonium.

Plutonium has been used in reactor fuels in the form of plutonium-aluminum alloys, plutonium oxide, plutonium carbide, plutonium nitride, plutonium cermet, and mixtures of the above compounds with uranium. Care must be exercised in handling carbides and other finely divided compounds because they spontaneously ignite in moist air.

1.2.1 Oxidation of Plutonium During Storage

The problems of oxidation of metallic plutonium during storage were recognized shortly after the discovery of plutonium, and extensive studies of the low-temperature corrosion of plutonium and its alloys have been performed. The heat generated by oxidation may be sufficient to ignite nearby combustible materials.

Massive (i.e., not finely divided) plutonium is relatively inert in dry air and is comparatively easy to handle and store for a few days. Special precautions must be taken when storing metallic plutonium for longer periods of time. Simply enclosing plutonium in an inert atmosphere in a metal can may not be adequate because of the possibility of the introduction of moisture, which greatly accelerates oxidation. It is suggested that metallic plutonium be stored in welded stainless steel cans containing a dry inert atmosphere and that the container be placed upon a heat detector to sense possible chemical reactions that may occur. Metal turnings and scrap should not be stored, but should be converted to oxide.

The corrosion or oxidation of plutonium does not always occur in a linear or predictable manner. The oxidation rate is a complex function of the surrounding atmosphere, the moisture content, and the alloys or impurities present in the metallic plutonium. The reader is referred to the literature for more detail (Wick 1967; Taube 1964). A selected bibliography, including a series of conferences on plutonium transuranics, is given at the end of this section.

1.2.2 Ignition Temperatures and Pyrophoricity of Plutonium

The health physics aspects of an accidental plutonium fire can be serious. A fire can burn through containment structures, resulting in the dispersal of PuO_2 over a wide area and the potential for inhalation exposure during the fire or during subsequent decontamination efforts. Plutonium, some plutonium alloys, and some plutonium compounds are pyrophoric. Finely divided plutonium, such as turnings or powders, are definitely pyrophoric and must be handled with care. Turnings must be stored in a dry atmosphere and should be converted to the oxide as soon as convenient, preferably on the same day they are generated. Certain solvents and organic compounds form flammable mixtures with plutonium. Chlorinated solvents have been involved in several fires with plutonium and its alloys.

Pyrophoric products may be formed on plutonium and certain alloys if they are stored for long times in closed containers. When a container is opened, spontaneous ignition may occur, which can result in the destruction of the container, damage to the glove box, and spread of finely divided or particulate oxides throughout the glove box and its ventilation system. Badly corroded plutonium metal, especially in the presence of moisture, can ignite spontaneously. Moisture in contact with plutonium can decompose the plutonium to form complex mixtures of hydrides and suboxides which are pyrophoric.

Many plutonium compounds are also pyrophoric. The authors of this manual have witnessed the spontaneous ignition of plutonium carbide in air initiated by simply lifting off the lid of a metal can that contains a powder sample. The possible use of plutonium carbides or nitrides as breeder reactor fuels may introduce the possibility of serious handling and fuel fabrication problems.

A number of intermetallic compounds are pyrophoric, particularly those of lead, mercury, and bismuth. Binary plutonium alloys that contain silver, gold, or copper disintegrate to a loose powder approximately 1 day after their exposure to moist air. The presence of impurities can change the corrosion rates and pyrophoricity of metallic plutonium.

Studies have been made of the conditions under which a plutonium fire can occur in a dry glove box. With only 5% oxygen in nitrogen, the metal will burn easily. But at the 1% level, a fire will not continue to burn unless heat is supplied (Wick 1967).

There is considerable variation (280 to 535°C) in the ignition temperature of metallic plutonium depending upon the purity of the metal. Alloying plutonium with other metals can also significantly alter the corrosion resistance and ignition temperature. Resistance to ignition and oxidation is increased by aluminum, cerium, carbon, cobalt, and manganese; resistance is decreased by iron and uranium. Nickel and silicon seem to have no effects (Wick 1967).

Because of the pyrophoric nature of metallic plutonium and its alloys, the preferred form for storing and handling plutonium is PuO_2. To mitigate the possibility of plutonium dispersal resulting from an accidental fire, PuO_2 is the form specified for offsite shipment or transport of plutonium.

1.3 BIOLOGICAL PROPERTIES

The biological properties of plutonium and other transuranics are known primarily from experiments performed on rats, dogs, baboons, and rabbits.

Human data on plutonium are limited. The vast literature on plutonium has been reviewed (Hodge, Stannard, and Hursh 1973; ICRP 1972, 1979, 1986; Liverman et al. 1974). ICRP 48 (1986) is the most recent compilation of data and is the source for most of the information that follows. The ICRP Publications 30 (1979) and 48 (1986) report different gastrointestional (GI) absorption and biodistribution parameters. At the 1987 Washington meeting, the ICRP (1987) adopted the GI absorption fraction described in ICRP 48, the bone and liver partitioning described in ICRP 30, and the retention half-times for bone and liver described in ICRP Publication 48. The Washington statement of the ICRP is equivocal and some individuals have interpreted the statement to mean that the bone and liver partitioning described in ICRP 48 was also adopted. Manual calculations, not shown here, indicate that for inhaled materials, the first year annual effective dose equivalents are nearly the same for both ICRP 30 and 48 models. The committed effective dose equivalents, calculated using the models of the two publications will differ by about 10%; with the ICRP 48 model yielding the lesser value.

1.3.1 Modes of Entry into the Body

The radiation dose delivered to the organs of the body from plutonium taken into the body is related to the chemical and physical properties of plutonium and the route of entry into the body. There are four routes by which plutonium may enter the body: 1) inhalation and deposition in the respiratory tract, 2) injection, 3) ingestion, and 4) percutaneous absorption (absorption through the skin).

Inhalation

The distribution pattern of inhaled material within the lung is related to the activity median aerodynamic diameter (AMAD) of the aerosol provided that the particle sizes of the particulates follow a log-normal distribution (ICRP 1979). The ICRP (1979) has published a mathematical model for estimating the distribution of an aerosol in the respiratory system based on the AMAD. The AMAD for a particular aerosol may be estimated (e.g., Knutson and Lioy 1983; Rajhans 1983; Peterson 1978), although it is common to assume an AMAD of 1 μm.

The length of time that an aerosol will be expected to remain in the lung following inhalation is based on its solubility class. There are three solubility classes described by the ICRP in their Publication 30 (ICRP 1979): D (day), W (week) and Y (year); the times listed indicate the order of magnitude of the half-life of the material in the lung. The biological half-lives for D, W, and Y materials in the lung are 0.5, 50, and 500 days respectively. The actual solubility class can be estimated by measurement of the materials solubility in simulated lung material (e.g., Kalkwarf 1979, 1983) or by using the solubility class determined by the ICRP for various chemical compounds (see Table 1.4).

TABLE 1.4. Solubility Classes of Various Plutonium Compounds[a]

Compound	Solubility Class
None	D
All except oxide	W
Oxide	Y

(a) From ICRP 30 (1979).

Injection

Plutonium (or americium) may be injected in the form of metal slivers, saturated resin beads, oxide powder, or any other form following penetration of the skin by various contaminated objects consequent to some type of accident. In many of these instances, the injected plutonium compound is in the form of a solid rather than a liquid and is injected into the flesh rather than the circulatory system; consequently the injected material will be slowly leached from the injection site into the blood stream, the rate of which will depend on the solubility of the injected material.

Ingestion

Plutonium is not readily absorbed by the gut. The absorption is influenced by the mass ingested, by fasting, by incorporation into foodstuffs, by complexing anions such as citrate and diethylenetriaminepentaacetic acid (DTPA), and by a variety of other factors (ICRP 1986). The fractional absorption of various plutonium and americium compounds by the gut are shown in Table 1.5.

TABLE 1.5. Gastrointestinal Absorption of Plutonium[a]

Compound	Fraction Absorbed, x 10^{-4}
Plutonium	
oxides, except "polydisperse" oxides	0.1
nitrates	1
other compounds or unknown mixtures	10
Americium	
all compounds	10

(a) From ICRP 48 (1986).

Percutaneous Absorption

The intact skin is an effective barrier against plutonium. The ICRP (1986) states that the absorption of plutonium will follow the data presented in Table 1.6 and that the percutaneous absorption of americium and neptunium is probably similar to that of plutonium. If the skin has been damaged, percutaneous absorption will be enhanced.

TABLE 1.6. Percutaneous Absorption of Plutonium[a]

Compound	Duration of Exposure	Percent Absorbed
In diluted aqueous acid	First hour	0.01
In tri-*n*-butylphosphate complex	15 min	0.04

(a) From ICRP 48 (1986).

1.3.2 Distribution Within the Body

Liver and bone appear to be the principal sites of plutonium deposition after it has entered the blood stream; they account for nearly 80% of the deposited plutonium (ICRP 1986). The exact partitioning of plutonium between the liver and bone is not well established. Table 1.7 shows the partitioning suggested in both ICRP 30 (1979) and 48 (1986). The ICRP has not issued a clear statement as to which partitioning is preferred. From the standpoint of

TABLE 1.7. Distribution and Retention of Plutonium Within the Body[a]

Organ	ICRP 30 Fraction	ICRP 48 Fraction	ICRP 48 Half-Time
Bone	0.45	0.50	50 yr
Liver	0.45	0.30	20 yr
Gonads			
Testes	3.5×10^{-4}	3.4×10^{-4}	permanent
Ovaries	1.1×10^{-4}	1.1×10^{-4}	permanent
All others[b]	0.1	0.20	not stated

(a) Based on ICRP 30 (1979) and ICRP 48 (1986).
(b) Includes tissue and early excretion.

the influence of the partitioning on the (calculated) committed effective dose equivalent, the two partitioning schemes are probably equivalent because bone (bone surfaces and red marrow) have a weighting factor nearly 3 times that for liver. When comparing the effective dose equivalent between the two systems, the reduction of the weighted dose equivalent from the liver will be compensated for by an increase in the weighted dose equivalent to the bone. Although there is only a minor difference in the (50-year) committed dose equivalent between the two partitioning systems (on the order of 7 to 10%), there may be major differences in the computed dose equivalents to individual organs. There appears to be a great deal of individual variability in the observed distribution patterns. Because of the large variability, the ICRP (1986, 1987) has stated that the ICRP 30 (1979) model "remains an adequate assumption."

Children can be anticipated to exhibit a partitioning of plutonium and other actinides between liver and skeleton that is different than that of adults; that difference has not been quantified, however.

The retention parameters of plutonium within the body are also shown in Table 1.7. There appears to be no reason to assume that americium has different half-times than does plutonium.

1.3.3 Uptake by and Distribution Within the Embryo and Fetus

The ICRP (1986) states that there is no strong evidence that actinides concentrate preferentially in the embryo or fetus. Further, any depositions in the embryo or fetus are anticipated to be diluted rapidly by growth. Radiation protection programs that adequately protect the mother are anticipated to protect the fetus also.

1.4 MANUFACTURE AND USES OF PLUTONIUM

In the past, most plutonium in U.S. Department of Energy (DOE) facilities was produced for nuclear weapons and was composed of greater than 90 wt% ^{239}Pu and about 6 to 8 wt% ^{240}Pu. This material has been referred to as "weapons grade," "production-grade," or "low-exposure" plutonium. Plutonium has also been produced as a byproduct in the operation of research reactors, nuclear-powered ships, and commercial nuclear power plants. Plutonium-238 is also produced at DOE facilities for use as a heat source in electric generators.

1.4.1 Future Sources and Uses

Because recycling of commercial reactor fuel is not anticipated, future supplies of plutonium will be primarily from DOE production facilities and from reprocessing of current material. In the more distant future, liquid metal fast breeder reactors (LMFBRs) may be a potential source of plutonium.

Because of its high specific alpha activity and high decay heat, ^{238}Pu has been used as an isotopic heat source for devices that generate thermoelectric power, such as the Space Nuclear Auxiliary Power (SNAP) systems used in lunar missions. Small amounts of ^{238}Pu with low ^{236}Pu content were used as a power source for medical prosthetic devices such as cardiac pacemakers and a prototype artificial heart, but lithium batteries have replaced plutonium power sources. Figure 1.3 shows the main modes of production of the plutonium isotopes of interest.

1.4.2 New Technology

Several new technologies are being considered to provide more highly purified plutonium isotopes for various purposes. One of these processes, laser isotope separation (LIS) is a potential new process to purify ^{239}Pu from almost any source of plutonium.

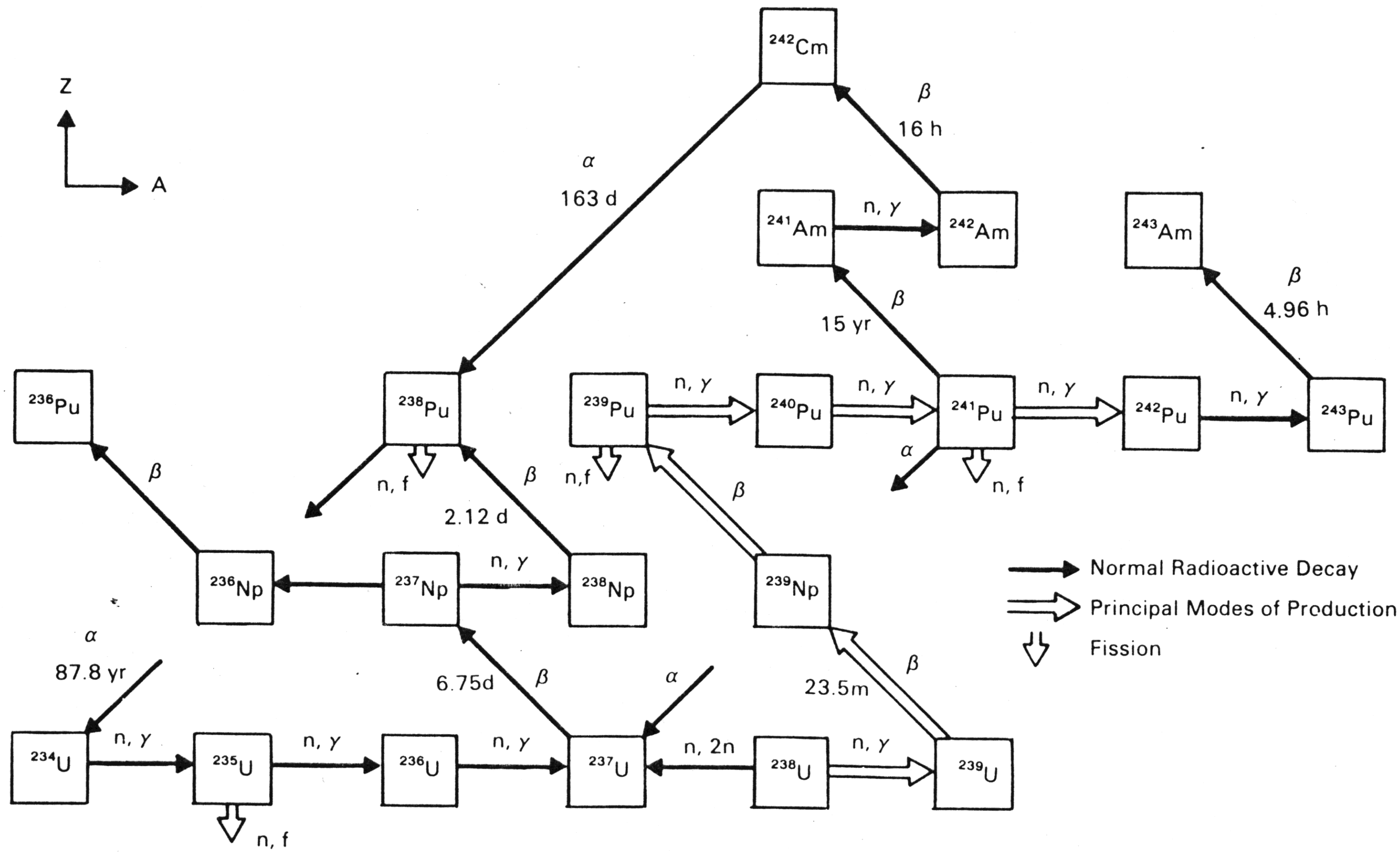

FIGURE 1.3. Principal Modes of Production of Plutonium by Neutron Irradiation of Uranium

The LIS process produces a product enriched in ^{239}Pu and a byproduct that contains the remaining plutonium isotopes. It is conceivable that the byproduct stream could be further purified to produce a specific plutonium isotope, such as ^{238}Pu used for isotopic heat sources.

The LIS process has many benefits. It can significantly reduce external radiation exposure to both neutron and gamma radiations for the product enriched in ^{239}Pu. Potential exposure problems from the byproduct stream are discussed later in this section. Recently, the ICRP (ICRP 1985) and the National Council on Radiation Protection and Measurements (NCRP 1987) have recommended increasing quality factors to a value of 20 for fast neutrons. Thus, it may be desirable to reduce neutron exposures. Neutrons arise primarily from even-numbered plutonium isotopes (mostly ^{238}Pu and ^{240}Pu) as a result of spontaneous fission and alpha-neutron reactions with low atomic number impurities in the plutonium. The ^{239}Pu-enriched product of LIS will have reduced concentration of these isotopes resulting in lower intrinsic neutron exposures. The LIS process can also result in significant reductions in gamma-ray exposures for the product enriched in ^{239}Pu. Much of the whole body and most of the extremity exposure results from surface contamination on the gloves and the interior of the glove box. The ^{241}Am decay product, which results from the beta decay of ^{241}Pu, is a major contributor. Thus, the reduction of ^{241}Pu can significantly reduce exposures to hands and arms and reduce the radiation streaming through glove ports in shielded glove boxes.

The LIS process, while mostly beneficial in exposure reduction, creates some challenges in internal dosimetry. The purified ^{239}Pu product is depleted in the ^{241}Pu isotope, which has a half-life of 15 years and decays by beta particle emission into ^{241}Am. It is the ^{241}Am that emits significant quantities of 60 keV gamma rays, which can be detected by lung and whole body counting methods. The LIS process will result in the loss of the ^{241}Am "tag," which in time is used to assess internal depositions from measurements of its radiations.

The LIS process also results in the production of large quantities of byproduct material consisting of ^{238}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu and perhaps ^{241}Am. Workers could be exposed to releases of either the LIS product and/or byproduct material. There could be wide variations in the isotopic compositions of the ingested or inhaled plutonium so that the assessment of the actual internal deposition by external counting techniques would be extremely difficult. Greater emphasis must be placed on periodic urine and fecal sampling to determine internal depositions and more careful monitoring of the air that the workers breathe. These actions could be tempered by improvements in chest or whole body counting capabilities. It may be prudent to obtain air samples from glove boxes in which LIS operations are performed. Sufficient activity should be present on the air samples so that isotopic analysis and isotopic ratios can be performed to establish a baseline for use in bioassay data interpretation.

While increased problems in accurately estimating the internal dose from the more highly purified plutonium isotopes will be encountered, the area, personnel, and equipment surveillance programs and procedures that are normal for a plutonium facility should remain unaffected. However, the health physicist who is responsible for contamination control in facilities working with the LIS material must remember that some of the early indicators of problems that rely on detecting radiations from impurities may no longer be available.

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SECTION 2.0

SITING OF PLUTONIUM FACILITIES

2.0 SITING OF PLUTONIUM FACILITIES

Technical, political, economic, and many other factors influence the selection of a site for a plutonium facility. For the health physicist involved in the site selection process, the paramount factor is the ability to ensure the protection of site workers and the public from undue risks of radiation. Stringent isolation and control of plutonium are required to ensure that radiation doses to the public are maintained as low as reasonably achievable (ALARA).

The siting of a plutonium facility may significantly impact the design, operation, and procedural requirements of the facility as they relate to the radiation protection of workers and the public. The natural characteristics of the site must be considered to ensure that the facility is designed to provide adequate radiation protection during all postulated accidents. The proximity of the facility to other operations and their potential interactions during normal operation and accident situations influence radiation protection and emergency response requirements.

Facilities that handle and process plutonium should be designed, constructed, and specially equipped for work with radioactive materials that are highly toxic when taken internally. Such facilities should be located so that normal operations and postulated abnormal situations neither adversely affect other plant personnel and adjacent buildings nor impose an undue risk to the health and safety of the public.

This section first discusses the applicable standards that define the requirements for siting DOE facilities. Then six of the factors to be considered during the site selection process are addressed, as follows:

- natural site characteristics
- transportation
- utilities
- other facilities and operations
- security and safeguards
- environmental, safety, and health aspects.

Siting criteria are found in DOE 6430.1, General Design Criteria Manual (DOE

1983a). DOE 6430.1 is currently being revised and will be reissued when completed. This section provides guidance on the siting of a plutonium facility.

2.1 APPLICABLE STANDARDS

The general requirements for the siting of DOE facilities are covered in Chapter 1, Section 3.i 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 on 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.

2.2 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, 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.

2.2.1 Meteorology

The wind patterns (speed, direction, frequency, duration, and stability) at a site must be tabulated. These data are needed to estimate radiation doses to populations from possible releases of radioactive material. The data should also include frequency and intensity of rainfall, snow and ice storms, thunderstorms and lightning strikes, and other events that 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 such events are not likely to occur. Complete histories of the magnitude and frequencies of such events in the region of the site should be compiled and evaluated to ensure that the location and design of each facility provides for the health and safety of the public.

2.2.2 Hydrology

Precautions should be taken to avoid flood damage, erosion, and water pollution. The 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. These 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. Geological Service (USGS) can supply data on runoff, water distribution, and the worst probable flood. 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 (1966).

Finally, the population groups that use water that could be contaminated by plant effluents under both accident and normal conditions must be identified. The evaluation of water use should include potable water supplies (both surface and subsurface), crop irrigation supplies, and recreational uses.

2.2.3 Geology and Seismology

DOE 6430.1 (DOE 1983a) states that careful consideration shall be given to seismic characteristics of a site during site development planning. Geologic and seismic data for the site of the proposed plutonium facilities should be gathered. Earthquake data and maps can be obtained from the USGS. The geologic conditions that underlie all structures, dams, dikes, and pipelines should be examined for the possibility of earth movement that could damage the facilities. Natural conditions, such as caverns or potential landslide areas, and manmade conditions caused by mining or the withdrawal or addition of sub-surface fluids should be considered. The design of the facilities may need to comply with the criteria for a design basis earthquake. The location of fault lines, frequency and intensity of earthquakes, location of epicenter, and other seismic data should be obtained and analyzed. If the maximum ground acceleration could exceed 0.1 gravity at the foundation of the plants, 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-ft (305-m) long within 5 miles (8 km) of the plant should be considered in the plant design.

The potential for volcanic activity that could affect the facility should be determined. The potential effects of ashfall on power supplies and safety systems should be evaluated.

The liquefaction potential of the soil and material under the site and the stability of hillside slopes that could affect the plant should be analyzed. The stability and load-bearing characteristics of the soil at the site should also be determined.

2.3 TRANSPORTATION

All nuclear facilities should be isolated from highly populated areas. However, the facility should also have reasonable access to major transportation networks. Because plutonium facilities will typically require shipment of plutonium to and from the site, access to rail systems or interstate highway networks will be required.

Because many state governments have the authority to designate traffic routes for shipment of radioactive material, close coordination with state and local agencies is necessary.

2.4 UTILITIES

Availability of electrical utilities, potable water, and raw water should be considered in the siting of a plutonium facility.

2.5 OTHER FACILITIES AND OPERATIONS

In the siting of plutonium facilities, the projected effects from nearby industrial, transportation, and military installations and operations should be considered. Potential adverse effects and their impacts on the safe operation of the facility should be evaluated. Examples of potential hazards include the release of toxic chemical fumes, flammable gas clouds, and radioactive materials; aircraft crashes; and missiles from explosions.

2.6 SECURITY AND SAFEGUARDS

The selection of a site for plutonium facilities must consider provisions for securing and safeguarding of the facilities. Special conditions for restricting and controlling access will be required. DOE 5632.4 Physical Protection of Security Interests (DOE 1985b) provides specific responsibilities and authorities in this area.

2.7 ENVIRONMENTAL, SAFETY, AND HEALTH CONSIDERATIONS

Plutonium 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 any 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.

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SECTION 3.0

FACILITY DESIGN

3.0 FACILITY DESIGN

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 DOE plutonium facilities can be found in DOE 6430.1, General Design Criteria (DOE 1983a), which is currently being revised. This section provides guidance in the design of plutonium 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 6430.1 (1983a). 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 engineered and administrative 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. Radiation protection efforts may be significantly enhanced by the incorporation of the appropriate design features rather than relying on administrative controls. However, in many instances, the designer and the health physicist must balance competing objectives to attain the most cost-effective design with a high degree of safety and reliability. In designing a new facility, all of the necessary physical features can be included; however, in an old facility it may be physically or economically impossible to meet all of the requirements.

The guidance presented in this section relate to physical safety and control systems only; guidance related to administrative control is not included. The phrase "safety and control systems" is used here to refer to the physical, engineered features that are used to provide radiation and contamination control. In addition to the radiation protection requirements, facilities that contain more than 450 grams of plutonium are subject to criticality safety requirements, which include the need for a criticality alarm system and criticality dosimeters. Guidance on the security and safeguards of nuclear material (including prevention of theft or diversion) is not included but also must be considered in the design of the facility.

This section addresses the applicable standards and guides, design objectives, structural guidance, building layout, service and utility systems, and special monitoring, safety, and other systems, required for the design of a plutonium facility.

3.1 APPLICABLE STANDARDS AND GUIDES

The design criteria in Chapter XXI of DOE 6430.1, General Design Criteria Manual (DOE 1983a), which pertain specifically to plutonium facilities, shall be applied for all new facilities that contain substantial quantities of in-process plutonium. A facility that will handle more than 1 g of plutonium, under certain specific conditions, shall also meet the security requirements of DOE 5632.4, Physical Protection of Security Interests (DOE 1985). A facility that will handle more than 450 g of plutonium or 450 g of any combination of plutonium, ^{235}U, and ^{233}U, must also meet the requirements of DOE 5480.5, "Safety of Nuclear Facilities" (DOE 1986a). An extensive list of applicable DOE orders, standards, and guides is provided in the reference and bibliography portions of this section. Other sources of information are the U.S. Nuclear Regulatory Commission's (NRC's) Regulatory Guides for Fuels and Materials Facilities, the International Atomic Energy Agency's (IAEA's) booklets, Safety Series No. 39 and 30, The Safe Handling of Plutonium and Manual on Safety Aspects of the Design and Equipment of Hot Laboratories (IAEA 1974 and 1981), 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 (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 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.

3.2 DESIGN OBJECTIVES

The objective of any good design for a plutonium facility is to ensure plant, public, and environmental safety during routine operation, to minimize any potential for loss of life or property in the event of an accident, and to minimize the impacts on public health and the environment in the event of an accident.

The specific facility design chosen depends on the quantity and form of plutonium that will be used. If more than 450 g of plutonium is to be permitted in the facility, the criticality safety criteria must be considered.

Some simple processes involving very small quantities of unsealed plutonium can be carried out safely in well-designed and adequately filtered open-faced hoods such as those found in a general radiochemistry facility. The specific quantity that can be handled in this manner depends on the complexity of the process and the specific form of the material. Any use of unsealed plutonium shall be reviewed by the facility's safety personnel, and the feasibility of the proposed use shall be established based on the form of the material to be used, the work to be performed, and the engineered and administrative controls to be employed. Based on experience, if the quantity of plutonium is 100 mg or more, the process should be performed in a plutonium facility.

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 plutonium involved, operations to be performed, degree of need for operating continuity during and/or after 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 should be performed to make this decision.

The primary goal of the design objectives is to keep the plutonium confined in its intended place (i.e., capsule, hood, glove box, etc.), both during normal operations and under accident conditions. Of equal importance is consideration of the human factors in design that promote efficiency and ease of operation. Additional design criteria may be necessary in considering the requirements for decontamination, decommissioning, and dismantling (discussed in Section 6.0) of the facility when it no longer is needed.

3.2.1 General Design Considerations

U.S. Department of Energy policy states that occupational and public radiation exposure shall not exceed the limits specified in DOE orders and shall be maintained ALARA.

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 ICRP (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 be deemed necessary and thus will not justify the incremental cost required to accomplish it. In optimization, the cost of reducing radiation exposure should 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 reactor design purposes, the NRC has recommended $1000/person-rem, as given in 10 CFR 50, Appendix I (CFR 1985a). DOE/EV-1830-T5 (DOE 1980) suggested that if dose reduction could be achieved at a cost of ≤$2000/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-1830-T5 (DOE 1980).

Equipment reliability and human factors engineering should be considered in the design of plutonium 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. The recommendations that are provided in Publication 76-45-2 SSDC-2 of the Energy Research and Development Administration (ERDA), Human Factors in Design (ERDA 1976a), should be considered during the design of control panel arrangements, instrument indicators and readouts, and alarm indicators.

The equipment should be designed such that the failure of a single component does not result in an "unacceptable radiological consequence." Unacceptable radiological consequences include criticality and unnecessary radiation exposures or unplanned radioactive material releases. Analyses of hazards and assessments of risks shall be made during conceptual and preliminary design activities and further developed during the detailed design phase. The safety analyses shall be performed in accordance with DOE 5481.1B, 5480.5, 5700.2C, and 6430.1 (DOE 1986b, 1986c, 1984, and 1983a).

In the planning and designing of buildings, other structures, and their operating components and systems, all aspects of operation and maintenance should be considered. This includes accessibility, dismantling, replacement, repair, frequency of preventive maintenance, inspection requirements, personnel safety, and daily operations. Facility planning and design should use 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 the areas that have the lowest possible radiation and contamination levels if possible. For equipment that is expected to be contaminated during operation, provisions should be made for both in-place maintenance and for removal to an area of low dose rate for repair. Maintenance areas for repair of contaminated equipment shall include provisions for containment or confinement of radioactive materials.

Engineered safety and control systems should be designed so that they 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 systems 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," in accordance with the definition in DOE 5500.1A (DOE 1987a). Facilities shall be designed to facilitate the arrival and entry of emergency personnel and equipment in the event of a radiological emergency and to allow for access by repair/corrective action teams.

Equipment shall be available to allow for an early and reliable determination of the seriousness of an accident or abnormal event. Consideration should be given to relaying all such equipment alarms to a central control system or a continuously manned area. Installed on-line equipment shall be protected to the extent necessary to ensure its reliability under accident conditions. To further enhance equipment reliability, the emergency equipment should, to the extent practicable, be the same equipment used for routine operations, in accordance with DOE N5500.2 (DOE 1987b).

Emergency power requirements that need to be satisfied and the means to provide the power shall be identified in the design effort.

Emergency radiological equipment shall be installed or located in areas that permit periodic inspection, testing, calibration, and maintenance.

Additional emergency preparedness guidance is provided in Section 5.0. Decontamination, dismantling, and decommissioning requirements should be considered in the design of a facility, in accordance with DOE 5480.11 (DOE 1988). Section 6.0 provides additional information on these topics.

3.2.2 Confinement

The confinement system is a series of physical barriers that, together with a ventilation system, minimizes the potential for release of radioactive material into work areas and the environment under normal and abnormal conditions. The primary design objective for the confinement system shall be an essentially zero exposure of the public and plant personnel to airborne contamination in accordance with DOE 6430.1 (DOE 1983a). Plutonium shall be separated from the ambient environment by at least two barriers and from an operator by at least one barrier.

Primary confinement refers to the barrier that is or can be directly exposed to plutonium, e.g., sealed process equipment (pipes, tanks, hoppers), glove boxes, confinement boxes, open-faced hoods, conveyors, caissons, and cells and their ventilation systems. The primary confinement barrier prevents the dispersion of plutonium through either sealed construction or atmospheric pressure differential or a combination of both. For example, process equipment that is not sealed but contains plutonium material in process should be enclosed in glove boxes or other confinement barriers. Fuel rod cladding, bags, and other sealed containers can be considered primary confinement. The chemical reactivity and the heat generation effect of the plutonium compound should be considered when selecting primary confinement 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 (glove rupture, damaged seals, improper bag-out operations, leaks of flanged joints, etc.).

The primary confinement (with the exceptions of fuel rods, sealed sources, or sealed cans) shall be maintained at a negative air pressure with respect to the secondary confinement in which it is located, and it shall be exhausted through a ventilation system that uses high-efficiency particulate air (HEPA) filters. The barrier and its accessory equipment should be designed to prevent accidental 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.

Where necessary because of the nature of the process being conducted, recycle ventilation systems may be used in process enclosures, hot cells, and canyons. Inert gas systems shall be designed as recycle systems, unless it is impracticable to do so. Recycled inert gas systems should be maintained completely within the primary barrier system. Extreme caution should 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, in accordance with DOE 6430.1 (DOE 1983a). Filtration shall be provided to limit the concentrations of radioactive material in recirculated air to ALARA levels. The design shall allow for in-place testing of 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 should be provided for the differential pressure across the filter stages and for airborne radioactive material behind the first HEPA filter or filter stage. The means for automatic or manual 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 if an automatic system is used.

The secondary confinement barrier encloses the room or compartment in which the primary confinement barrier is located, and provides contamination protection for plant personnel who are outside of the secondary confinement area. High efficiency particulate air filtration shall be required for air supplied to and exhausted from a secondary confinement barrier. Secondary confinement rooms, compartments, or cells should be separated from each other by fire doors or stops. Both the barrier walls and the fire doors shall be constructed of materials that are capable of withstanding a design-basis accident (DBA). The secondary confinement shall be designed for pressures that are consistent with the criteria for the ventilation system. The secondary confinement area shall be at a positive air pressure with respect to the primary confinement areas and at negative pressure with respect to the outside environment and adjacent building areas that are not primary or secondary barriers.

The building is the structure that encloses both the 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 if the requirements for both structure and confinement are met. The portion of the structure that houses activities involving radioactive material in a dispersible form shall be able to withstand DBAs, site-related natural phenomena, and missiles without a breach of integrity that would result in releases of radioactive material from the structure in excess of DOE guidelines.

3.2.3 Design-Basis Accident Events

Critical items are systems whose continued integrity and operation are essential to assure confinement or to measure the release of radioactive materials in the event of a DBF or DBA. Critical items usually consist of ventilation, fire detection and suppression, electrical, and utility systems. The degree of confinement of radioactive materials shall be sufficient to limit environmental releases to ALARA. 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 the isotopes of plutonium and other radioactive elements present, shall be taken into account in making these assessments. Design-basis accidents should be developed specifically for individual facilities as part of the safety analysis report. The following paragraphs discuss typical DBAs.

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 design features shall conform to applicable criteria as specified in DOE 6430.1 (DOE 1983a) and to other site- or process-specific criteria developed for the facility.

Development of the design-basis fire (DBF) shall include consideration of conditions that 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 that 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 fireproofing). In no event shall the fire resistance rating be less than two hours. As a minimum, penetrations in this shell shall incorporate 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 a DBF. A high degree of reliability and/or redundancy shall be required of all protective features of the ventilation system to ensure its effective operation even if normal plant utility and fire protection systems fail. Redundancy in operation should include independent auxiliary services such as electrical power and service air.

The design-basis explosion may involve 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). (Not only will this energy release result in a pressure wave, but it also may generate missiles within the process area.)

The design-basis criticality could involve 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 1 second, or an accidental pulsating excursion with a total yield of 10^{20} fissions. (This energy release may disperse unencapsulated plutonium from a typical glove box and may pressurize the room.) As a minimum, the design of nuclear criticality control provisions shall meet the requirements of DOE 5480.5, Safety of Nuclear Facilities, (DOE 1986c). Geometrically favorable or poisoned tanks and process vessels shall be provided to minimize reliance on administrative control. The use of poisons is acceptable only if their effectiveness can be monitored. 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.

A typical design basis power failure accident may be the loss of total electric power for approximately 1 minute, and the loss of normal electric power for 24 to 48 hours. Total electric power refers to all sources of electric energy, delivered as well as auxiliary and standby. Normal electric power refers to the services usually supplied by a utility company.

The DBA-water is the result of an uncontrolled water hazard. This may occur when water supplied to the plant from a controlled external source is released in an uncontrolled manner for 30 minutes within the plant. The uncontrolled release of water may result in loss of a system, subsystem, structure, or component that is 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 DBA-natural phenomenon is the effect of site-related conditions, including tornado and other wind and storm conditions, earthquakes, floods, and volcanic activity.

3.3 STRUCTURAL CRITERIA

The structure and its associated critical equipment, ventilation, electrical, fire protection, and utility systems shall be designed to confine radioactive materials during 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.

3.3.1 Tornado Resistance

Critical operating areas of the facility shall be designed to withstand the design basis tornado. Specific information on site-specific tornado hazard curves, rotational speeds, elastic or plastic design methods, and other design criteria are provided in DOE 6430.1 (DOE 1983a).

Coats and Murray (1984) developed wind/tornado hazard models aimed at establishing uniform building design criteria for wind/tornado hazards at DOE sites throughout the United States. The model developed for each site expresses the annual probability that the site will experience a tornado greater than some specified magnitude.

3.3.2 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).

3.3.3 Seismic Design Requirements

Specific seismic design criteria for determination of the size of an earthquake, earthquake analysis, earthquake occurrence, load combinations, and stress limits are provided in Chapter XXI of DOE 6430.1 (DOE 1983a). 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 equivalent to at least one-half the DBE 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 to prevent and mitigate the consequences of any accident that occurs. These items include

- main structures
- fire protection systems
- ventilation systems
- confinement piping and equipment
- critical utilities, instrumentation, monitors, and alarms
- material in process and in storage (nuclear criticality).

Coats and Murray (1984) developed seismic hazard models that established uniform building design criteria for earthquakes at DOE sites throughout the United States. The model developed for each site expresses the annual probability that the site will experience an earthquake greater than some specified magnitude.

3.3.4 Other Natural Phenomena

Design loads and considerations for other natural phenomena shall provide a conservative margin of safety that is 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 the flood protection that is required for the facility. If the facility is in a location that may be subject to ashfall from volcanic action, consideration should be given to the effects of ashfall on ventilation and electrical systems.

3.3.5 Explosion, Internal Pressurization, Criticality, and Other Causes of Design-Basis Accidents

Analyses shall be made to determine the probable consequences of DBAs, and critical areas and critical items shall be designed to withstand DBAs. The portion of the ventilation system that is an integral part of the critical areas shall be designed to withstand DBAs so that it will remain intact and continue to act as a confinement system. Building ventilation is an important part of the confinement barrier(s) and, in some cases, air flow may become the only barrier.

3.4 BUILDING LAYOUT

Building layout is extremely important in the operation of a plutonium facility. Improper or poor layout can lead to operational difficulties and in some instances can contribute to the development of abnormal situations that may affect personnel safety, result in unnecessary exposure to the worker and the public, and/or increase the cost of operating the facility. Normally, three areas are involved in the overall building layout. For purposes of this manual, these areas are described as:

- the process area, where plutonium or other radioactive or hazardous materials are used, handled, or stored

- the controlled area, which is normally free of radioactive material but could potentially become contaminated
- the uncontrolled area, which includes all areas where no radioactive materials are permitted and radiological controls normally are not necessary (i.e., offices, lunchrooms, etc).

The terms "controlled area" and "uncontrolled area" defined above refer to radiologically controlled and uncontrolled areas. These terms are not to be confused with the formal definitions of controlled and uncontrolled areas found in DOE orders.

3.4.1 Objectives

The following objectives shall be achieved in the design layout of the facility:

- Planned radiation exposures to personnel shall be within the prescribed limits of DOE 5480.11 (DOE 1988). Radiation exposures to individuals and population groups shall be ALARA.
- The planned or unintentional release of radioactive materials from the facility shall be confined to the limits of DOE 5480.11 (DOE 1988) and ALARA.

3.4.2 General Design Criteria

All planned processing, research and development (R&D), scrap and waste handling, analytical, storage, shipping, and receiving operations shall be accommodated. Receiving operations that involve removal of radioactive material from protective shipping containers shall be performed in a handling area that has provisions for confinement.

Real-time or near real-time accountability systems should be incorporated if possible.

The possibility of operating with multishifts per day shall be taken into account in allocating space for personnel support facilities and for any special equipment that might be required to support multishift operations.

Areas shall be compartmented to isolate the high risk areas, thereby minimizing productivity and financial loss if a DBA occurs.

A modular construction concept should be used where feasible to facilitate recovery from operational accidents and DBAs and provide versatility.

All movement of personnel, material, and equipment between the process area and the uncontrolled area shall be through a controlled area or an air lock. Doors that provide direct access to the process area from the uncontrolled area (including the outside of the building) shall not be permitted. If such doors are required by existing design and operating requirements for emergency exits, special administrative control shall be implemented to ensure adequate ventilation and radioactivity control. All such doors shall have airtight seals. Doors without air locks shall have alarms that sound when the doors are opened to signal a breach in the contamination control system.

Personnel exits shall be provided in accordance with the NFPA Life Safety Code (NFPA 1985a). Personnel working in areas where an accidental breach of primary confinement will expose them to radioactive material shall be located within 75 feet of an exit that leads into the next confinement barrier. Such a barrier should be a partition separating two different air control zones, the area of refuge being on the upstream side of the barrier. The airflow through the barrier should be in the opposite direction of the exit travel.

Normal administrative traffic shall 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.

Consideration should be given for provision of a ready room near or within the process area where maintenance, operating, and monitoring personnel may be readily available. The room should be in a low background area. Storage should be provided for instruments and tools needed for routine work.

Process areas shall be located to permit ease of egress and material movement to ensure rapid evacuation in case of an accident and minimum potential for contamination spread during movement of material.

Indicators, auxiliary units, and equipment control components that do not have to be adjacent to operating equipment should be installed outside of radiation or contaminated areas. Units and components without internal contamination that are located in radiation areas should be designed so that they can be removed as quickly as possible.

Equipment that requires frequent servicing or maintenance should be of modular construction, standardized to the extent possible, and located outside the process area if possible.

In radiation areas, work spaces around equipment (pumps, valves etc.) that requires maintenance should be shielded to conform to the design basis radiation levels.

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 the movement, repair, installation, or removal of proposed or anticipated equipment.

Ergonomic factors should be considered in the selection and placement of equipment components to facilitate operation and maintenance.

In any area where personnel may wear protective clothing or use breathing air systems, the use of sharp equipment projections, which could tear clothing or breathing air system hoses or cause wounds, shall be avoided in accordance with DOE/EV/1830-T5 (DOE 1980).

Water-collection systems shall be provided for water runoff from any controlled area. Water from firefighting activities should be considered. 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.

Area drainage and collection systems should be designed to minimize the spread of radioactive contamination, especially to areas occupied by personnel.

Curbs should be constructed around all areas that house tanks or equipment that contain contaminated liquids to limit the potential spread of liquids, in accordance with DOE 6430.1 (DOE 1983a).

Noncombustible and heat-resistant materials should be used in radiation areas that are vital to the control of radioactive materials and in equipment that is necessary for the operation of radiological safety systems. These materials should be resistant to radiation damage and should not release toxic or hazardous byproducts during degradation in accordance with IAEA Safety Series No. 30 (IAEA 1981).

Floors, walls, and ceilings should have a smooth, impervious, and seamless finish. The junction between the floor and walls shall be coved, and corners should be rounded. 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 (ANSI 1974).

An emergency lighting system shall be provided in radiation areas to facilitate egress in emergencies. The emergency lighting shall meet the requirements of NFPA 101 (NFPA 1985a).

Space shall be allocated for radiological monitoring stations at exits from radiation areas.

3.4.3 Process Area

The plutonium process area is typically a group of contiguous rooms that contains all operations involving plutonium, including processing, shipping, receiving, storage, and waste handling. To the maximum extent practicable, the facility design shall provide 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, glove boxes, conveyor lines, the ventilation system, and process piping. The primary confinement system shall be designed to minimize the impact of accidents and abnormal operations on people, facilities, and programs. The type of confinement enclosure used (hood, glove box, remote operation cell, etc.) depends on the amount and dispersibility of unsealed plutonium that will be handled and on the process involved. Generally, if the quantity of unsealed plutonium exceeds 100 mg, the use of a glove box shall be considered. However, the applicability of this guideline will vary based on the individual merits of each case.

Piping and Valves

Piping and valves for radioactive liquids shall 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 shall be avoided in the design of radioactive piping systems.

The piping system that collects contaminated liquids shall be designed so that effluents from leaks in the system can be collected without releasing the liquids 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 made nonfunctional by a single accident.

Stainless steel should be used in all radioactive waste and process system piping and equipment to ensure that smooth, nonporous, corrosion-resistant materials are in contact with contaminated liquids. For some applications, polyvinylidene fluoride (PVDF) piping may be preferred for inside of confinement enclosures because of its ease of fabrication, smoothness, nonporosity, and corrosion resistance. However, it undergoes severe degradation after about 7×10^7 rads of exposure. In general, organic materials should not be used in process piping systems. Other materials may be used if engineering analyses demonstrate that criteria are met for strength, smoothness, porosity, and corrosion resistance for the liquids to be handled.

Piping systems used for conveying radioactive and corrosive materials should be of welded construction whenever practicable. Flanges should be used only when absolutely necessary for servicing.

Positive measures shall be taken to prevent any radioactive material in the facility from entering a utility service. This may be achieved by using backflow prevention devices and by prohibiting direct cross-connections inside the facility. The most successful backflow prevention device is the deliberate separation of lines.

Every pipe that enters or exits a process cell or contaminated area from or to occupied areas should be equipped with block valves.

Process piping systems carrying radioactive liquids shall be designed to eliminate traps wherever possible and to permit flushing and draining except for those with loop seals. Floor drains should have the capability to be sealed.

Reduction in the size of pipelines in contaminated process piping systems should be made with eccentric reducers installed flat side down to avoid the formation of traps. Eccentric reducers are only necessary for horizontal pipe runs.

Changes in the direction of process piping should be made with long-radius elbows or bends. Long-radius bends should be used, where practicable, except in lines that transport solids, where blinded tees or laterals have been proven to prevent erosion. Blinded tees will encourage solids buildup. The number of bends should be minimized and pipe diameter should be increased.

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.

Except for shielding walls, pipe sleeves should generally be provided when piping passes through masonry or concrete walls, floors, and roofs. The sleeves should be sloped to drain toward the controlled area. The space between the pipe and the sleeve should be packed and sealed. If the sleeve is to be sealed, then additional provisions should be made for draining the annulus.

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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 that are not functionally required to be in contact with contaminated liquids should be located in nonradiation areas (e.g., steam, air, water) in accordance with IAEA Safety Series No. 30 (IAEA 1981).

Process valves should not be located at low points in the piping except in cases where it is necessary in order to properly drain the piping when needed.

Valve seals and gaskets should be resistant to radiation damage.

Straight-through valves generally should be used to simplify maintenance and minimize particle traps.

Valves shall be designed to operate in the stem-up orientation, which would limit potential leakage when the pipe is unpressurized. Valves and flanges shall be located to minimize the consequences of contamination from leaks.

Generally, process solutions should have primary and secondary confinement. However, in rare instances where process solutions are allowed to flow outside of confinement, they should only flow by gravity and the pressure head should be limited to an equivalent of about 10 ft of water.

The corrosion resistance of the primary block valve and/or check valve and all associated piping in the in-cell and/or contaminated areas should be equivalent.

The use of pumps in contaminated piping systems should be avoided to reduce potential contamination problems that result from pressurization and to reduce the maintenance requirements associated with pumping. The use of gravity flow, jets, vacuum, or airlifts is a suitable alternative. Vacuum transfers are preferred. If jets or airlift transfers are used, an adequate waste air cleanup system should be provided.

Structure

Floors shall be designed in accordance with code requirements considering the maximum loads anticipated.

Storage

In-process storage should not be permitted; however, temporary storage of the product in the process area until it can be taken to an appropriate storage area should be permitted.

Storage facilities in the process areas shall be designed to prevent the exposure of operating personnel and to meet the requirements for security and safeguards as given in DOE 5632.4, Physical Protection of Security Interests (DOE 1985), 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 material (SNM).

Shielding

Provisions shall be made to accommodate the shielding of all items in the process area. All structures (floors, walls, glove boxes, etc.) may require additional shielding during the lifetime of the facility because of increased throughput or higher radiation levels of the material being processed.

DOE 6430.1 (DOE 1983a) establishes 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/h. DOE 5480.11 (DOE 1988) includes a requirements that dose equivalent rates in a routinely occupied location shall average less than 0.5 mrem/h or 20 mrem/wk. It further requires that a process area with a dose rate between 100 mrem/h and 5000 mrem/h shall be controlled with signs or lockable barriers. For dose equivalent rates of 5000 mrem/h or greater, lockable barriers shall be provided.

Concrete radiation shielding should be in accordance with ANSI N101.6-1972, Concrete Radiation Shields (ANSI 1972).

Straightline penetration of shield walls shall 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 hands and forearms would otherwise approach the design criteria of 10 rem/yr. Also, robotics or other non-hand contact methods shall be used where contaminated puncture wounds could occur.

Shielding materials shall be noncombustible or fire resistant, to the maximum extent practicable.

Confinement Devices

Different devices may be used to confine and control radioactive material. The selection of the appropriate device will depend on the quantity of material, its form, and the operations to be performed. For specific operations, encapsulation may be the confinement of choice. Sealed source containers shall be designed to prevent contact with and dispersion of the radioactive material under all normal 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.

Seismic protection shall be provided to minimize movement of confinement enclosures if ground movement occurs.

Fume hoods may be used for some operations with plutonium, depending on the quantity and dispersibility of the material. In general, plutonium fume hood operations shall be limited to wet chemistry processes and less than 100 mg of plutonium. For some operations, such as metallography and x-ray analysis, larger quantities may 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 room exhaust point, which might disturb air flow into the hood. 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 greater than 125 linear ft/min over the hood face area shall be provided to ensure control of radioactive materials. 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 should be increased. Turbulence studies may be necessary to verify adequate control of radioactive material. Physical stops should be provided to ensure that the required hood face velocity is maintained.

Hood design and filtration systems shall comply with the criteria established in ERDA 76-21, Nuclear Air Cleaning Handbook (ERDA 1976b), Industrial Ventilation, A Manual of Recommended Practice by the American Conference of Governmental Industrial Hygienists (ACGIH 1980), and by Oak Ridge National Laboratory (ORNL) in ORNL-NSIC-65, Design, Construction, and Testing of High Efficiency Air Filtration Systems for Nuclear Applications (ORNL 1970).

The hood structure shall have a smooth, corrosion-resistant, inner surface that is made of or coated with easily cleaned material.

Glove boxes, conveyors, and other enclosures shall be designed to control and minimize the release of radioactive materials during normal operations and postulated DBAs. Noncombustible or fire- and corrosion-resistant materials should be used in the construction of the confinement system, including any shielding employed. Fixed modular construction shall be employed wherever possible, using a standardized attachment system that will permit replacement or relocation of the contents within the glove box system with a minimum spread of contamination. Fire dampers shall be provided between glove boxes to limit the spread of fire. Fire dampers shall be tested frequently to assure proper operation when needed.

The process design should minimize required hands-on operation in glove boxes and other primary confinement units.

The glove box design shall include sufficient work space to permit removal of materials and easy personnel access to all normal work areas, and it shall provide for the collection, packaging, storage, and/or disposal of waste generated by the operation of the glove box.

Consideration should be given to incorporating transfer systems such as a double door, sealed transfer system for removal of plutonium from a glove box. Types of removal and transfer systems are given in IAEA Safety Series No. 30 (IAEA 1981). These types of removal systems are designed to permit entry and removal of material without breaching the integrity of the glove box.

The ease of visibility for activities, accessibility of necessary in-box controls, and ease of cleanup and waste removal should be considered in the design of glove boxes. Glove boxes should be designed and constructed to reduce points of material accumulation.

Equipment shall be designed to preclude sharp corners, barbs or pointed parts, and pinching points that could puncture glove box gloves or skin. All corners shall be rounded, burrs removed, etc.

Ergonomics shall be considered in designing the height of glove box ports and access to inner surfaces and equipment.

Each glove box should be equipped with an audible alarm that can be tripped to signal radiological problems. Individuals should be able to activate the alarm without removing their arms from the glove box. The alarm should sound in a continuously occupied area where it should, as a minimum, identify the room in which the alarm originated.

A HEPA filter shall be installed on the air inlet to the glove box if required to prevent the backflow of contamination. Prefilters should be installed upstream of the HEPA filter where appropriate. The exhaust outlet for each glove box shall have HEPA filters to keep the ventilation duct work clean. This filter should not be counted as a formal HEPA stage and need not meet all the test capabilities for HEPA filtration; however, it should be tested prior to installation. Push-through filter change-out systems should be used if possible. The HEPA filters downstream of the glove box shall be readily accessible for filter change-out and testable.

Glove box air inlets and inlet filters shall be protected or oriented to prevent inadvertent entry of water into the box (e.g., a fire sprinkler system discharge or water line leak).

Glove boxes shall be designed to operate at a negative pressure [0.75 ± 0.25 in. water gage (WG)] with respect to the room in which they are operated. Differential pressure gauges shall be installed on each glove box or integrally connected series of glove boxes. During abnormal conditions control devices to prevent excessive pressure or vacuum shall be either 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. There shall be exhaust capacity on demand that will promptly cause an inflow of air greater than 125 linear ft/min through a breach of at least a single glove box penetration of the largest size possible. Filters, scrubbers, demisters, and other air-cleaning devices shall be provided to reduce the quantities of toxic or noxious gases and airborne particulates that enter the ventilation system prior to its entry into the exhaust system.

Each glove box or integrally connected series of glove boxes shall be equipped with an audible alarm that alerts personnel when a system pressure or vacuum loss is occurring. The alarm should be set at -0.5 in. WG relative to the room in which the glove box is located.

The number of penetrations for glove box services should be minimized. The fittings shall provide a positive seal to prevent the migration of radioactive material. For the same reason, penetrations for rotating shafts shall not be permitted except where rotating shafts have seals. Seals for rotating shafts are very reliable and are preferred to motors inside the glove box.

Vacuum systems connected to a glove box shall be designed to prevent an evacuation and possible implosion of the glove box.

Any gas supply system connected directly to a glove box shall be designed to prevent pressurization, flow in excess of the exhaust capacity, and backflow. Flammable or combustible gases should not be used in glove boxes but, if required, shall be supplied from the smallest practical size of cylinders. Flammable gas piped to a plutonium processing building shall not enter the building at a pressure exceeding 6 in. water (DOE 1983a). Vacuum pump exhaust shall be filtered and exhausted to the glove box or other acceptable exhaust system.

If process water is provided to a glove box and the water must be valved ON when the box is unattended, a system shall be installed to automatically close a block valve in the water supply line if a buildup of water is detected on the box floor or in the box sump.

Process piping to and from glove boxes should be equipped with backflow prevention devices and should be of welded stainless steel construction. Vacuum breaker type devices are generally more reliable than other types.

Glove box components, including windows, gloves, and sealants, shall be of materials that will resist deterioration by chemicals and radiation.

Glove ports shall be designed to allow for the replacement of gloves while maintaining control of radioactive material. The ports 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 ports should be considered in the design. Covers or plugs should be provided for each port. The covers or plugs should provide shielding equivalent to the glove box walls.

Bag-out ports, sphincter seals, and air locks shall be designed and installed to facilitate the introduction and removal of equipment and supplies without compromising contamination control. Air lock gaskets at the bottom rim shall be protected from any physical damage potentially incurred by removing items. Air locks shall be designed to be at negative pressure with respect to the work station and positive pressure with respect to the glove box.

Windows shall be constructed of noncombustible or fire-resistant materials that resist scratching, breaking, and radiation degradation. Wire glass should be considered except where precluded by requirements for visual acuity. In those instances, tempered or safety glass may be suitable. Windows should be kept as small as possible while still meeting visual requirements. A push-in window design should be considered for ease of replacement. Use of PVDF lining or laminations on windows may reduce their degradation and increase the ease of their decontamination. The windows shall be securely fastened and gasketed or sealed. The gasketing material should be resistant to degradation by radiation or other materials to which it will be exposed. Lighting fixtures shall be mounted on the glove box exterior to the extent practicable.

Generally, organic (plastic) materials are not recommended for use in plutonium glove boxes. However, when dealing with process streams containing large quantities of fluorides or chloride ions, organic (plastic) pipe and equipment are sometimes required. When using organics in the glove box, care must be exercised in the selection of the material to minimize alpha deterioration.

Fire protection shall be provided in the glove box, enclosure, and conveyor systems to meet DOE improved-risk objectives. Automatic fire suppression shall 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 (DOE 1983a). Discrete work stations within an enclosure should be separated from each other by fire stops to prevent the spread of fire. 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 (DOE 1983a). Instead of a fire-sensing system, an oxygen-deficient atmosphere may be provided as the normal or required operating atmosphere within the enclosure. Where automatic fire suppression systems are not required, a fire-detection system shall be installed. Provisions shall also be made for manual fire suppression where it is deemed necessary.

The actual sources inside the glove box should be shielded, if possible, instead of shielding the glove box, However, the glove box should be equipped with or capable of accepting any necessary neutron and/or gamma shielding.

3.4.4 Controlled Area

All support facilities that have a potential for periodic low-level contamination shall 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 rate to occupants to less than 0.5 mrem/h in accordance with DOE 5480.11 (DOE 1988).

Air locks between controlled and uncontrolled areas shall be used to provide confinement of the controlled area if an inadvertent release of radioactive materials or a fire occurs. Air locks should also be provided in controlled areas where there is a potential for radioactive contamination to be spread from an area of high contamination to one of lower contamination.

Where possible, each controlled area shall have a single access and exit point for personnel during normal operation. Access points shall be accessible through change rooms. Other access and exit points shall be available as required for emergencies and in compliance with the NFPA Life Safety Code (NFPA 1985a).

Space for step-off pads and radiation monitoring and survey equipment shall 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 Rooms

Change rooms shall be available for both men and women, with lockers 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 ventilated 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 shall 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 shall be routed to the liquid radioactive waste system or to a holding tank that may be sampled 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) shall be provided for each plutonium facility. It should be located near or in the change rooms. A decontamination room with the capability to decontaminate male and female personnel simultaneously should be considered. The use of installed partitions or curtains should be considered for this purpose. An adequately equipped decontamination room should have communications equipment, a workbench with a cabinet for decontamination supplies, an examination chair, a sink, and showers. Both the sink and showers shall be connected to a holding tank for sampling or routed to the process waste. The room should contain equipment for performing nasal irrigations and initial surveys of nasal swipes.

Health Physics Lab Office

Health physics personnel in a plutonium 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, one radiation protection technologist (RPT) should be available for every 10 radiation workers. Space should be included for the readout of radiation protection instrumentation, survey records documentation, counting equipment, and portable instruments.

Mechanical Equipment

Where possible, mechanical equipment (motors, pumps, valves, etc., that may be a source of radioactive contamination) shall be located in the process area. Enclosures that will contain the contamination should be placed around the equipment. Such enclosures should be easy to decontaminate.

3.5 SERVICE AND UTILITY SYSTEMS

Utility services shall be designed to provide reliability that is consistent with 1) the operational requirements for the control and confinement of radioactive materials and 2) the 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 same integrity level as the function they serve. Some service or utility systems are connected to other systems or structures that are essential to prevent the release of radioactive materials. Such service or utility systems must be designed so that if they fail, connecting systems will remain functional.

3.5.1 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. The design of ventilation system components does not include process vessels, primary confinement or containment housing, or the building structure.

Design Objectives

The ventilation system shall be designed to confine 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, which serves as an engineered safety and control system, shall be designed to remain operational or fail safely under all operational and DBA conditions. The failure of any single component shall not compromise the ability of the system to maintain confinement of radioactive materials or control their release to the environment. Specific response requirements of the system and its components shall be identified through a safety analysis.

Air Flow and Balance

The design of ventilation systems shall ensure that, under all normal conditions, the air flows toward areas of progressively higher radioactive material inventories. Air-handling equipment shall be sized conservatively enough that minor fluctuations in air flow balance (e.g., improper use of an air lock, occurrence of a credible breach in a confinement barrier, etc.) do not result in air flowing from higher to lower radioactive material inventory areas. To prevent the movement of contamination from high radioactive material areas to low radioactive material areas in case of a flow reversal, HEPA filters shall be provided at ventilation inlets in confinement area barriers.

A minimum of two negative-pressure zones should exist within a process building. The first, the process confinement system, should serve the spaces within the glove boxes, conveyors, transfer boxes, and other spaces that may contain plutonium 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 that are contiguous to process areas and potentially free of contamination constitute a third zone. Some facilities have a minimum of three zones and frequently four.

A minimum pressure differential of between 0.75 and 1.0 in. (1.9 and 2.5 cm) WG, negative with respect to the room, shall be maintained in all process confinement systems. A negative pressure differential of at least 0.1 in. (0.25 cm) WG shall be maintained between process and controlled areas and between controlled areas and uncontrolled areas. Air locks between zones should be provided where necessary to ensure that proper differential pressures are maintained. Differential pressure between the containment enclosure and the outside atmospheric pressure may be as great as 3 in. of water (ERDA 1976b).

The design of the ventilation system shall include an analysis to demonstrate that the system is capable of operating under the DBF conditions. To the maximum extent practicable, the system shall be designed to ensure that the products of combustion are not spread beyond the room of origin unless directed through appropriate ventilation channels. The exhaust system shall be designed to provide cleanup of radioactive material and noxious chemicals from the discharge air and to safely handle the products of combustion.

Provisions should be made for independent shutdown of ventilation systems where this could be an advantage to operations, maintenance, or emergency procedures such as firefighting. In assessing the desirability of providing for shutdown of a ventilation system under such conditions, full consideration shall 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. For example, reducing air supply to 10% and exhaust flow to 20% of operating values would minimize ventilation and maintain negative pressure. Positive means of controlling the backflow of air, which might transport contamination, shall be provided. The ventilation system and the associated fire-suppression system shall be designed for fail-safe operation.

The ventilation system shall be appropriately instrumented and alarmed, with readouts in continuously occupied control rooms. A listing and the function of required and recommended instrumentation are given in ANSI N509-1980, Table 4-1 (ANSI 1980a).

Building penetrations for ventilation ducts should be kept to a minimum and should be designed to protect the critical systems against a DBA. No penetrations should be permitted, if the barrier around the process area is the outside wall of the building.

Room air in controlled and process areas may be recirculated if the recirculating air system is provided with two HEPA filter banks in series. One of the filter banks should be in the exhaust duct leading from the room(s) where airborne activity might be introduced. An air monitor shall be located between the two filters and set to alarm when the air concentration reaches a preset point. Air flow should then be diverted either manually or automatically to a once-through system using the air monitor alarm indication to trip the system. Recirculation from a zone of higher contamination to a zone of lower contamination shall be prohibited.

Air Supply

Supply air shall be appropriately filtered and conditioned in accordance with operational requirements and with the levels recommended for comfort.

The ventilation rate in process areas, where uncontained radioactive materials are handled, should be from 12 to 60 air changes per hour (ORNL 1970) depending on whether the area is normally occupied by workers, the need for removal of process or decay heat, and the need for removal of decay fumes. A minimum of 8 air changes per hour should be provided in support facilities within the process area. Adequate air filters shall be used at the intake of the ventilation supply system to minimize dust in the process area and to reduce the dust loading on HEPA filters.

A downward airflow pattern should be provided at worker locations to direct air from any potential leak point down and away from the worker's face. Consideration should be given to the distribution of inlet air through a number of small ports or by slot-type distributors to decrease the possible occurrence of "dead spots" with little air circulation.

Glove boxes, conveyors, and other systems that require a controlled atmosphere may be equipped with a recirculating air system. All parts of the system should operate at air pressures that are 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 glove box.

Consideration should be given to isolating process rooms from each other during accidents. The principle of compartmentation and separation should be extended to systems handling ventilation in working areas, by the most practicable use of individual ventilation systems. Emergency back-up should be provided through combinations of manifolds and damper cutovers between adjacent individual ventilation systems. Redundancy can be minimized by the provision of a back-up unit for each two individual systems.

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. Materials released shall be in conformance with applicable standards, policies, and guidelines. In general, each exhaust filter system should consist of a minimum of two HEPA filters for room air and three HEPA filters in series for glove box or hood exhaust air. Only two stages of glove box or hood exhaust filters need to be equipped for in-place testing.

The filtration system shall be designed to allow for reliable in-place testing of the HEPA filters and ease of filter replacement to the extent practicable.

The exhaust system for a glove box or hood shall be separate from the exhaust system for room air. The hood exhaust system need not be separate from room exhaust ventilation if ventilation is once-through. Exhaust air shall be drawn through a HEPA filter at the glove box or hood exhaust point to maintain primary control at that point and minimize contamination of ductwork. This filter shall not be counted as a confinement barrier unless it is testable in place. Additional HEPA filters in series should be separated at a sufficient distance to permit in-place testing of each stage of the filters.

Dampers should be installed in the glove box, hood, and room exhaust ducts so that required air-pressure differentials can be maintained. Automatic backflow dampers should be installed in series with the exhaust dampers. Manual controls, or automatic controls with manual override, should be provided as needed for ventilation systems or their components for flexibility of operation.

Integral fire-suppression equipment shall be provided as needed within each ventilation system to ensure 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. The water spray from a cool-down chamber 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. Continuous monitoring and a representative, redundant sampling capability shall be provided on exhaust stacks that may contain radioactive or toxic materials. The ventilation exhaust stack shall be located as far away from any air intake as is 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 1983b). International Electrotechnical Commission (IEC) Standard 761, Parts 1 and 2, provides requirements for equipment for continuously monitoring radioactivity in gaseous effluents (IEC 1983).

System Testing and Control

The ventilation system is considered an essential safety and control system and should be designed in accordance with ANSI/ASME N509-1980 (ANSI 1980a). The minimum acceptable response requirements for the ventilation system, its components, instruments, and controls, shall be established based on results of safety analyses for normal, abnormal, and accident conditions. These requirements shall include system and component design characteristics, such as the installation of standby spare units, provision of emergency power for fans, installation of tornado dampers, seismic qualification of filter units, and fail-safe valve positioners.

The ventilation system shall be designed to operate effectively and to permit servicing or filter replacement while operating. The system's effectiveness shall be assessable during operation by means of installed testing and measurement devices.

Air-cleaning systems shall be designed for the convenient, repetitive, and reliable in-place testing of each stage of the system for which credit is taken in accordance with ANSI N510-1980 (ANSI 1980b). Provisions for in-place testing shall include aerosol injection ports, sampling ports, and connecting and bypass ductwork. Independent inspection and testing of HEPA filters prior to their installation shall be performed by DOE-approved organizations listed in Chapter V (page V-16) of DOE 6430.1 (DOE 1983a). Each filter bank also shall be tested upon installation and annually thereafter and anytime when conditions have developed that may have damaged the filter, i.e., pressure drop, over pressure, water spray, etc. The filter or filter bank shall demonstrate a particle-removal efficiency of at least 99.95%, for all particles having a mean aerodynamic diameter (MAD) $\geq$0.3 micron, on a count basis.

The portions of the ventilation system that are essential to preventing releases of radioactive materials shall continue to function (or automatically change to a safe failure mode) under abnormal or DBA conditions. The ventilation system fans shall produce a maximum exhaust rate that is greater than the maximum supply rate. Exhaust fans shall be provided with emergency power in the event of loss of normal electrical power supplies. Exhaust and supply fans should be redundant. If the system fails, exhaust control dampers shall fail in the open position and the supply control dampers shall fail in their preset closed position. Supply fans should automatically cut off when the exhaust fan capacity in service is not sufficient to maintain the proper pressure differential. Alarms shall be provided to signal the loss of fan capacity or improper air balance. System components or devices that must function under emergency conditions shall 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.

3.5.2 Electrical Power

Both normal and emergency power supplies shall be available to a plutonium facility to ensure that critical systems can continue to operate under both normal and accident conditions.

Normal Power

A plutonium facility's normal electrical power needs shall be met by two primary feeders. The preferred primary feeder shall provide basic service to the facility and consist of a radial feeder connected directly to the main substation serving the area. To minimize power outages, this feeder shall be an express feeder and shall not have any other loads connected to it.

The alternate primary feeder shall be in ready standby to provide back-up power to the preferred primary feeder power supply. In the event of a forced outage or planned maintenance of the preferred primary feeder, the power load shall automatically transfer to alternate feeder. The alternate primary feeder should also be a radial feeder connected directly to a substation and should have no other loads connected to it. To minimize simultaneous outages of the preferred and alternate primary feeders due to lightning or other physical damage, the two feeders shall have maximum physical separation.

Emergency Power

The facility shall be provided with a reliable, local source of emergency power if both primary sources fail. The emergency power source shall be completely independent of the preferred and alternate primary feeders. The emergency power should be generated onsite 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 seconds, and the emergency system should remain energized for at least 5 minutes after the restoration of primary power to allow for an orderly transition. The emergency power sources should have sufficient capacity and sufficient fuel supplies stored onsite to maintain the integrity of all critical building systems for approximately 48 hours. The amount of time that emergency power is necessary should be determined by the requirements for bringing the processes to safe shutdown condition. Chemical and thermal inertia also should be considered. The emergency power system should be able to carry identified critical loads such as air exhaust and supply systems, fire-detection and fire-suppression systems, related instrumentation and control functions, necessary criticality and radiation monitoring instrumentation, certain processing equipment, and any other essential building systems identified during safety analysis. Sensitive safety equipment shall be tested to verify that it will remain operable during the switchover and after enduring the electrical transient.

Noncritical uses of emergency power shall be avoided.

3.5.3 Water Supply

Water storage tanks with multiple or back-up supplies shall 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 in accordance with the criteria in Chapter X of DOE 6430.1 (DOE 1983a) and Factory Mutual and National Fire Protection Association Standards. The fire-protection water supply and distribution design required for critical item protection shall ensure the continuity of protection in the event of a DBA.

Potable water shall be distributed to drinking fountains, eyewash fountains, showers, emergency showers, lavatories, and noncontaminated laboratories. Raw water may be used in toilets and urinals. The potable water system shall be protected against contamination in accordance with Chapter V of DOE 6430.1 (DOE 1983a). Water mains should not pass through process or controlled areas. Branch lines may be permitted in process areas for safety showers and fire-protection sprinkler systems only. Drinking fountains may be located in controlled areas, adjacent to the process areas, where contamination is not likely to occur.

The facility water system preferably should be isolated from 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 [AWWA 1983] shall be used. Process water supplied to the process and controlled areas must be isolated from the potable water system. Cross-connections shall not be permitted.

3.5.4 Fire Protection

Each area in the plant building shall be equipped with fire-detection devices that are best suited for that area, as described in Chapter X of DOE 6430.1 (DOE 1983a) and in NFPA National Fire Codes 71 and 72A through 72D (NFPA 1985b, 1985c, 1986a, 1986b, 1986c). All equipment shall be approved by a recognized testing laboratory. The spacing, sensitivity, and location of the detectors should be given careful consideration to ensure rapid response.

Most areas of the plant shall have automatic fire-suppression systems; exceptions are identified in DOE 6430.1 (DOE 1983a). As a minimum, these systems and the structural design of the facility shall meet the requirements of the NFPA's National Fire Codes listed previously, Chapters X and XXI of DOE 6430.1 (DOE 1983a), Chapter VII of DOE 5480.1B (DOE 1986a), and DOE 5632.4 (DOE 1985). The minimum requirements shall include the provisions of completely automatic sprinkler systems or equivalent coverage throughout the facility, with fire control measures for any special hazards. An ensured water supply, adequate for firefighting and fire-suppression needs over a 4-hour period, shall be provided.

All fire detectors and/or automatic fire-suppression systems shall be connected to fire alarm annunciators. The annunciator system should be sufficiently subdivided to identify the location of a fire.

Fuels and combustible materials should be stored at a central facility that is remote from the plutonium processing building(s). Piped natural gas shall not be provided to plutonium process or storage areas. Separate bottled gas systems should be provided where required.

The ventilation system of the facility shall be designed to withstand any credible fire or explosion. It shall be constructed of non-combustible 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.

Over-pressure protection should be considered for critical items such as glove boxes, cells, and ventilation ducts.

3.5.5 Waste

Waste from plutonium handling facilities includes radioactive, radioactive mixed, and hazardous (nonradioactive) materials and will be in the form of liquid or gaseous effluents and solids packaged for shipment offsite. A principal design objective for the process systems shall be to minimize the production of wastes at the source. A principal design objective for the waste management systems shall be to provide facilities and equipment to handle the wastes safely and effectively. The design of the facility shall limit the environmental release of radioactive, radioactive mixed, and hazardous materials to less than the DOE and EPA regulations. Emphasis shall be placed on reducing total quantities of effluents (both radioactive and nonradioactive) released to the environment.

Sanitary Waste

Sanitary waste includes the nonradioactive wastes usually found at a facility, e.g., discharges from noncontaminated chemical laboratories, showers, and lavatories. The sanitary waste system shall not be located in the plutonium handling area where radioactive material could enter the system. Sanitary sewers should discharge into an onsite, approved, sanitary sewage treatment system. Current federal, state, and local codes regarding the discharge of sanitary wastes shall be met. The sanitary wastes shall be sampled for radioactivity (DOE 1983a).

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 the roofs of process buildings may be included in this category. Sufficient holdup capacity shall be provided so that wastes can be retained until they are sampled, analyzed, and shown to be within acceptable limits for release. Holdup capacity shall also be provided for water that is collected from firefighting activities, including sprinkler activation. While providing holding tank capacity for the full volume of firefighting water may not be economically feasible, consideration should be given to ensuring that the water is retained onsite and can be sampled prior to release.

Potentially contaminated liquid wastes shall be sampled prior to their discharge to the environs. Batch sampling and analysis of liquid waste holdup tanks may be used.

If liquid waste is discharged to the environs, the effluent concentrations of plutonium shall not exceed the Radioactivity Concentration Guide (RCG) of DOE 5480.1B (DOE 1986a) for uncontrolled areas measured at the point of discharge during normal operation and shall be ALARA.

Contaminated Waste

Any contaminated waste (solid, liquid, or gas) that does not meet the criteria for release either shall be held onsite and decontaminated to the point that it meets release limits or disposed of as radioactive waste.

3.6 SPECIAL SYSTEMS AND EQUIPMENT

Special systems and equipment shall be incorporated in plutonium facilities to ensure the safety of the worker and protection of the public. As a minimum, the following systems shall be included:

- air sampling and monitoring
- breathing air
- personnel monitoring
- criticality safety
- nuclear accident dosimeters
- monitoring and alarms.

These systems and equipment plus some that may not be directly related to personnel safety and radiation protection are discussed in the following subsections.

3.6.1 Air Sampling and Monitoring

Airborne alpha activity can be measured by either air sampling or continuous air monitoring. The use of both is required for an effective program. An air sampling system may involve the use of a central building vacuum system with the sample heads located near positions that are frequently occupied by operating personnel. Alternatively, individual portable samplers could be used. The use of flexible intake lines to sample heads will allow placement of the sample head near the worker's breathing zone. Continuous air monitors (CAMs) shall be located throughout the areas where radioactive materials are handled to alert personnel of sudden increases in airborne plutonium. The air sampling and monitoring systems shall be located and operated such that the air concentrations measured are representative of those that 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 shall be considered. Specific information may be found in Sections 1.0 and 4.0, in the references and bibliographic materials for this manual, and in

Mishima et al.[a] Continuous air monitors 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 percent 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[b] and N42.17B-D5, Performance Specifications for Health Physics Instrumentation - Occupational Airborne Radioactivity Monitoring Instrumentation[c] shall also be considered.

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 that may contain plutonium contaminants shall be provided with a continuous-monitoring system and a fixed sampling system. These systems may be a combination unit. The air intake probes 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 shall follow the guidance provided in ANSI N13.1-1969 (ANSI 1969), ERDA 76-21 (ERDA 1976b), and DOE/EP-0096

(a) Mishima et al. 1988. Health Physics Manual of Good Practices for the Prompt Detection of Airborne Plutonium in the Workplace. Draft Report, Pacific Northwest Laboratory, Richland, Washington.
(b) American National Standards Institute (ANSI). 1988. Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions. Draft ANSI N42.17A-D8, American National Standards Institute, New York, New York.
(c) American National Standards Institute (ANSI). 1987. Performance Specifications for Health Physics Instrumentation - Occupational Airborne Radioactivity Monitoring Instrumentation. Draft ANSI N42.17B-D5, American National Standards Institute, New York, New York.

(DOE 1983b). Each of these systems shall be connected to an emergency power supply in accordance with DOE 6430.1 (DOE 1983a). The requirements of Parts 1 and 2 of IEC Publication 761, Equipment for Continuously Monitoring Radioactivity in Gaseous Effluents (IEC 1983) should also be met. Effluent monitoring systems should provide a continuous recording of effluent concentrations and be equipped with alarm annunciation of abnormal radioactivity levels in the effluent discharge stream.

3.6.2 Breathing Air

For facility design, confinement of airborne radioactive materials shall be the required method of preventing internal deposition of radioactive particulates. However, during operation and maintenance of the facility, situations may occur (accidents, special maintenance, spill recovery, etc.) where air-supplied respiratory protection is required.

A plutonium facility should be provided with a system that is capable of supplying breathing air to a number of work stations in each occupied area where the following conditions exist:

- Gaseous or airborne radioactive material may exceed the concentrations listed in DOE 5480.11 (DOE 1988), for occupational exposure.
- Potentially dispersible plutonium compounds exceeding 100 mg are handled outside of containment devices.
- Personnel may be required to enter cells or other areas that contain large amounts of loose radioactive material for repair, maintenance, decontamination, or operation.

Breathing air systems may be portable or semiportable bottled systems or installed compressor systems. The facility design requirements shall be determined by the system selected.

Breathing air supply systems shall meet the requirements of ANSI Z88.2-1980 (ANSI 1980d) and 29 CFR 1910 (CFR 1985b). Air line connections for the breathing air must be unique to preclude connecting other gas supplies to the breathing air lines or the breathing air lines to other gas supplies. Additional criteria for design of breathing air systems found in the references mentioned above shall be considered.

3.6.3 Personnel Monitoring

The facility design shall provide for location of personnel monitoring devices in the vicinity of the workplace in accordance with DOE 6430.1 (DOE 1983a). To minimize the potential spread of radioactive contamination, personnel survey instruments shall be available at suitable locations within the process area, such as for personnel exiting from glove boxes, at bag-out stations, and at exits from compartmentalized facilities. Survey instruments or monitoring instruments shall be available at contamination control change rooms and at exits from controlled areas.

3.6.4 Criticality Safety

Criticality alarm systems (gamma or neutron) shall be provided in each area where an accidental criticality is possible. The requirements of DOE 5480.5, Safety of Nuclear Facilities (DOE 1986c), ANSI/ANS 8.3-1986, Criticality Accident Alarm Systems (ANSI 1986), ANSI/ANS 8.1-1983, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (ANSI 1983), and ANSI/ANS 8.19-1984, ANS Administrative Procedures for Nuclear Criticality (ANSI 1984) shall be met.

3.6.5 Nuclear Accident Dosimeters

All DOE facilities that possess sufficient quantities and kinds of fissile material to potentially constitute a critical mass shall provide nuclear accident dosimetry (DOE 1988). The number of dosimeters 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 of the dosimeters and their placement shall be conducted and documented to demonstrate that they meet the criteria listed below. The analysis shall include the number of dosimeters, their locations, and the effect of intervening shielding. Ease of recovery after a criticality event should be considered in the placement of the fixed dosimeters. Remote retrieval mechanisms may be appropriate.

Each fixed nuclear accident dosimeter shall be able to

- determine the neutron dose in rad at its location within 25% accuracy

- measure the approximate neutron spectrum to permit the conversion of rad to rem
- measure the neutron components of the total dose in the range from 10 rad to about 10,000 rad
- measure the fission gamma radiation in the presence of neutrons at its location within approximately 20% accuracy
- measure the gamma component of the total dose in the range from 10 rem to about 10,000 rem.

3.6.6 Monitoring and Alarms

So that abnormal conditions can be interpreted correctly and remedial action can be 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 staffed and has guaranteed accessibility. The inclusion of the following specific alarms and signals should be considered:

- fire, criticality, evacuation, and security alarms
- gaseous and liquid waste monitors
- ventilation system performance monitors for airflow and pressure differential
- room air monitors
- process monitors for flow, pressure, temperature, and other process parameters that have an impact on safety
- power monitors for power failure or loss of power to critical fans and pumps.

Essential monitoring and alarm systems shall be supplied with emergency power if normal power fails, because they must remain functional at all times. Reliability should be ensured by designed features such as redundant circuits, and instruments that perform self-checks and are tamperproof. All monitoring, surveillance, and alarm systems should be tested periodically.

3.6.7 Other Systems

Many systems employed within a plutonium facility are not directly related to personnel safety and radiation protection. However, because of the special impact that these systems may have on a plutonium facility, individuals responsible for personnel protection shall be aware of them. Some examples are:

- process instrumentation and control indicators to monitor and maintain control over the process and to detect and indicate abnormal and accident conditions
- surveillance systems to ensure the integrity of all process piping, tanks, and other containment equipment, including those used for liquid effluents
- vacuum, airlift, 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 that are used in plutonium handling areas. Gas and chemical storage facilities, including distribution piping systems, should conform to good design practice and applicable codes and standards. Consideration shall be given to compatible groupings that, under accident or leakage conditions, would minimize any adverse combining of materials. Means for remote shutoff of piping should be provided. In addition, the following rules should be observed:

- Nonflammable hydraulic and lubricating fluids should be used in the plutonium handling area.
- Protective barriers should be provided around high-pressure or other potentially dangerous systems.
- Incompatible chemicals, materials, and processes should be isolated from one another.
- Pressurized gas lines used in the plutonium handling areas must be properly vented.

Facilities for equipment maintenance should be provided.

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SECTION 4.0

RADIATION PROTECTION

4.0 RADIATION PROTECTION

The radiation protection field is concerned with the protection of individuals, their progeny, and humanity as a whole, while still allowing for necessary activities from which radiation exposure might occur. The aim of radiation protection is to prevent nonstochastic effects and to limit the probability of stochastic effects to acceptable levels. Most decisions about human activities are based on an implicit form of balancing risks and benefits leading to the conclusion of whether or not the conduct of a particular practice produces a positive net benefit. In radiation protection the application of the above concept does not always provide sufficient protection for the individual worker. For these reasons, the ICRP, in its Publication 26 (ICRP 1977), recommended the following criteria for a system of dose limitation:

- No practice shall be adopted unless its introduction produces a positive net benefit.
- All exposures shall be kept as low as reasonably achievable, with economic and social factors being taken into account.
- The dose equivalent to individuals shall not exceed the limits recommended for the appropriate circumstances.

The successful operation of a plutonium facility requires maximum attention to providing adequate radiation protection and maintaining contamination control through the implementation of a quality health physics program. (In the context of this document "health physics" and "radiation protection" can be used interchangeably when referring to programs or personnel.) A health physics program in a plutonium facility shall ensure the detection of all types of radiation (i.e., alpha, beta, gamma, x-ray, and neutron) over large energy ranges. The radiation detection instruments shall be properly calibrated and routinely checked. Emphasis shall be on establishing controls for internal and external radiation exposure using ALARA guidelines. Prompt and accurate dose assessment is important in determining each worker's risk immediately and in establishing a historical record. This section defines the basis for the establishment of a sound health physics program at a plutonium facility.

4.1 PERSONNEL REQUIREMENTS

The individuals who are responsible for health physics programs at plutonium facilities agree that experienced health physics personnel are necessary for a successful program. Experienced personnel can train others. The selection of operating personnel is also important. Operating personnel shall follow the rules and procedures established by the safety and health physics departments to ensure their own personal safety and that of others. Health physics personnel shall have the authority to balance operations with safety. An attitude that a worker's safety and protection comes first shall be reflected in the support provided by operating management and in the authority given to health physics managers and supervisors.

Management commitment to safety is the most important characteristic of an effective radiation protection program. If the management commitment to safety is strong, the radiation protection program will be valued and respected. The radiation protection program shall be provided adequate authority to permit performance of necessary assignments and program implementation. Management commitment to the ALARA concept is particularly important (DOE 1980a). Adequate personnel, equipment, and funding shall be available when this commitment exists.

4.1.1 Staff Selection

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

At least one professional health physicist shall be on the staff of each major plutonium facility. This individual shall have several years of health physics experience in the operation of plutonium facilities. As a rule of thumb, the facility should be staffed with one non-professional RPT for each

10 operating personnel. However, the exact number of RPTs should be based upon analyses of the operating processes and facilities. The complexity of the process or the facility, inherent problems associated with facility operation, and the age of the facility may increase the number of RPTs needed.

Where possible, the RPTs and other members of the health physics staff shall have a minimum of one year's experience working at a plutonium facility. Such experience is an important prerequisite to allowing them to work unsupervised. Personnel hired without such experience shall work an internship of six months under the leadership of a qualified RPT or supervisor with experience in that facility.

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

4.1.2 Performance Standards

Personnel working at a plutonium facility should be able to perform the work required in a safe and efficient manner. In accordance with Report 91 of the NCRP (NCRP 1987) and DOE 5480.11 (DOE 1988), radiation workers shall be advised that, "During the entire gestation period, the maximum permissible dose equivalent to the fetus from occupational exposure of the expectant mother should not exceed 0.5 rem." The NCRP (1987) further recommends that "once a pregnancy becomes known, exposure to the embryo-fetus shall be no greater than 0.05 rem in any month (excluding medical exposure)."

Individuals with skin breaks or open wounds shall be required to check with medical personnel before working with unencapsulated plutonium. The medical personnel along with supervisory personnel shall determine whether the work to be performed can be conducted safely considering the nature of the wound.

Performance standards for personnel assigned to plutonium facilities shall be extremely high. This means that personnel shall

- know and follow rules explicitly
- not take procedural shortcuts
- be sure that decisions and procedures are properly reviewed prior to initiating any new activities.

Because the entire operational and safety program depends heavily on the staff's attitude and commitment, emphasis shall be placed on the maintenance of good morale.

4.1.3 Health Physics and Operations Organizations

The management of a plutonium facility shall establish a clear organizational chain of command. Because health physics personnel shall have the authority to balance operations with safety, they shall not report directly to the administrators of operations. When shift work is involved, the operations shift supervisor may make minor health physics decisions in support of the shift's RPTs; however, decisions involving basic policies and procedures shall be directed to a separate health physics organization.

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

The radiation protection program shall be audited by independent reviewers at least biennially to determine its adequacy in meeting both operations and regulatory requirements. Contractor internal audits of all functional elements of the radiation protection program shall be conducted as often as necessary, but no less frequently than every three years in accordance with DOE 5480.11 (DOE 1988). The program for following up on deficiencies identified during audits and inspections should include an automatic internal review every 30 days until all deficiencies have been corrected. DOE 5482.1B (DOE 1986a) states that the internal audit system shall be reviewed by management a minimum of once every three years to ensure the adequacy of its performance.

4.2 INSTRUMENTATION CONSIDERATIONS

The radiation from the radioactive decay of plutonium includes alpha, beta, gamma and x-ray (photons), and neutron radiation (see Section 1.0). An effective monitoring program for plutonium requires radiation detection instruments that are responsive to all of these forms of radiation. The instruments shall meet the performance criteria outlined in the applicable U.S. and international standards and be properly calibrated for their intended use.

4.2.1 Types of Instruments and Measurements

Alpha-sensitive instruments are necessary for most contamination control surveys. Exposure rate surveys are normally conducted with photon-sensitive instruments with known energy responses for photons with energies greater than or equal to 10 keV. Neutron surveys become important when processing tens of grams of ^{238}Pu or hundreds of grams of mixed isotopes of plutonium, particularly compounds (i.e., PuO_2, PuF_4, etc.). The neutron survey is important in instances where photon shields, such as leaded glass, are used; such shields normally stop all of the charged particles, most of the low-energy photons, and essentially none of the neutrons. Under these circumstances, neutron radiation is likely to be the major contributor to whole body dose.

Continuous air monitors have been used extensively in plutonium facilities. Continuous air monitors and sample extraction lines that go to CAMs and continuous radiation dose monitors should be placed outside the glove boxes and hoods. With the advent of new separation processes, new wastes are being introduced in plutonium facilities. In-line processing instrumentation is critical to accurately monitor the work stations, and a safety analysis review should be performed to determine instrument locations. Continuous air monitors may not have adequate detection capabilities for real-time monitoring at the DAC level. For ^{239}Pu, the annual limit on intake (ALI) is 5.4 nCi for Class W compounds based on the DAC of 2×10^{-12} µCi/mL as given in DOE 5480.11 (DOE 1988). Representative manufacturers' specifications on the performance level of such a CAM, range from 1 DAC in 4 hours (4 DAC-h) to 1 DAC in 8 hours (8 DAC-h) for alarm. Continuous air monitors typically have poor large-particle response due to particle loss during transport to the filter inside the system. Background levels of radon-thoron decay products may be present in concentrations up to 50 to 100 times greater than the level of plutonium of interest. If calibrated properly, alpha CAMs will subtract background levels of radon-thoron decay products; however, in practice the detection limit is at least a factor of 2 higher due to fluctuations in radon levels.

Transuranic aerosol measurement units have been developed and adapted to be used in the workplace. These units avoid preferential plate-out of larger particles by using an in-line filter. Higher flow rates than those normally used with CAMs may be used. Increased detection is obtained on a quasi-real time basis by high-volume air sampling and counting in a separate vacuum chamber. Detection levels of less than 0.5 DAC-h have been quoted for these units. It has been demonstrated that high-volume impact samplers used at some facilities have demonstrated detection capabilities of 0.1 DAC-h in the laboratory and 1 DAC-h in the field. Other monitoring systems that use diffusion, impaction, or electronic discrimination to reduce the effect of background resulting in an increased detection capability, have also been used and are being improved upon. Swinth et al. (1987) reported on new developments in continuous monitoring of airborne activity.

4.2.2 General Performance Criteria for Instruments

Programs for in-plant monitoring of plutonium consist mainly of airborne and surface contamination surveys and dose rate surveys. The general and specific performance criteria for the instrumentation needed to conduct these programs are described in ANSI N317-1980 (ANSI 1980a). Performance specifications are also given in ANSI N323-1978 (ANSI 1978), draft ANSI N42.17A-D9[(a)], and draft ANSI N42.17C-D4[(b)] for portable health physics instrumentation and IEC Publication 325 (IEC 1983a) for alpha and beta contamination meters and monitors. Criteria for air monitoring instrumentation are contained in ANSI

(a) American National Standards Institute (ANSI). 1988. Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions. Draft ANSI N42.17A-D9, American National Standards Institute, New York, New York.

(b) American National Standards Institute (ANSI). 1987. Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Extreme Environmental Conditions. Draft ANSI N42.17C-D4, American National Standards Institute, New York, New York.

N13.1-1969 (ANSI 1969), ANSI N13.10-1974 (ANSI 1974), IEC Publication 761-2 (1983b), draft IEC Publication 761-6,[a] and draft ANSI N42.17B-D5.[b] Criticality alarm systems are discussed in ANSI/ANS 8.3-1986 (ANSI 1986). The criteria discussed in the following subsections are specified in these standards as referenced.

Portable Survey Instruments

ANSI N317-1980 (ANSI 1980a) discusses several criteria related to the performance of portable survey instruments; these include the following requirements:

- The overall accuracy shall be within ±20%, and the precision shall be within ±10% at the 95% confidence level.
- The response time (i.e., the time for the instrument reading to go from zero to 90% of full scale) shall be $\leq$10 seconds on the most sensitive scale and $\leq$2 seconds at readings of 100 mrem/h, 100 mR/h, and 500 dpm or greater.
- The instrument shall be able to maintain accuracy and precision for a minimum of 24 hours of continuous operation.
- The instrument shall have a minimum battery lifetime of 200 hours of continuous operation.

More current standards, such as ANSI N42.17A-D9[c], will, when issued, give specifications that differ slightly.

(a) International Electrotechnical Commission (IEC). 1983. Equipment for Continuously Monitoring Radioactivity in Gaseous Effluents - Part 6: Specific Requirements for Transuranic Aerosol Effluent Monitors. Draft IEC Publication 761-6, International Electrotechnical Commission, Geneva, Switzerland.
(b) American National Standards Institute (ANSI). 1987. Performance Specifications for Health Physics Instrumentation - Occupational Airborne Radioactivity Monitoring Instrumentation. Draft ANSI N42.17B-D5, American National Standards Institute, New York, New York.
(c) American National Standards Institute (ANSI). 1988. Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions. Draft ANSI N42.17A-D9, American National Standards Institute, New York, New York.

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

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

ANSI N317-1980 (ANSI 1980a) states that the minimum detection capability for alpha survey instruments ideally should be 220 dpm/100 cm^2 of surface area and shall not be more than 500 dpm/100 cm^2. This requirement shall be met in the presence of a radiation field of 0.10 rem/h of neutrons in the energy range of thermal to 10 MeV, and/or in the presence of 0.10 rem/h of photons in the energy range of 0.010 to 1.25 MeV. The operating range should be from 0 dpm to at least 100,000 dpm/100 cm^2 of surface area. The response of the instrument to beta-interfering radiation is an important specification that shall be stated by the manufacturer.

Photon survey instruments shall meet the accuracy requirements stated in ANSI N317-1980 (ANSI 1980a) over the energy range of 0.01 to 1.25 MeV. The angular response of this type of instrument should be within ±15% over a 2π steradian frontal direction using at least two photon sources with energies ranging from 0.06 to 1.25 MeV. Experience has shown that this response specification is not met by most instruments at lower energies due to attenuation of the photon. The energy dependence shall be within ±15% over the range of 0.01 to 1.25 MeV and the operating range shall be from 0.5 mR/h to at least 5000 mR/h. Experience has shown that ±20% over 0.01 to 1.25 MeV is more realistic. This specification applies to a specific window selection, (e.g., below 0.05 MeV the electron equilibrium cap or beta shield must be removed).

According to ANSI N317-1980 (ANSI 1980a), the response of neutron survey instruments for neutron energies in the range of thermal to 10 MeV shall approximate the dose equivalents given in that standard for instruments that are designed for dose equivalent rate measurements. The angular response for neutron instruments should be within ±15% in a 2π steradian frontal direction for ^{252}Cf energy neutrons or equivalent. The operating range shall be from 0 to at least 2000 mrem/h.

Draft ANSI N42.17A-D9[a] has a broader scope than ANSI N317-1980 (ANSI 1980a) but the criteria in it apply to portable survey instruments. Additional criteria include geotropism (maximum change of 6% from reference reading for all orientations), temperature shock, mechanical shock, vibration, and ambient pressure (maximum change of 15% from reference reading for these last four criteria). Some differences exist between draft ANSI N42.17A-D9 and ANSI N317-1980 (ANSI 1980a). In most cases the criteria for the new standard are more applicable because these criteria are based on substantial testing, which was sponsored by DOE. In draft ANSI N42.17A-D9, precision is tied into a measurement level; for example, it quotes a precision of 15% <500 cpm and 10% >500 cpm. Also, with the advent of liquid crystal displays and other digital readouts, "response time" is defined as the time it takes for the reading to move from 10 to 90% of the equilibrium or steady-state reading. Another significant difference in the proposed standard is that the battery lifetime specification is 100 hours instead of 200 hours.

For direct alpha contamination surveys the use of audible signals (headphones or speaker) greatly facilitates the detection of "hot spots."

The IEC Publication 325 (IEC 1983a) provides additional guidance on the uniformity of probe response for alpha and beta contamination meters. Surface sensitivity measurements are also discussed in this standard.

Fixed Monitoring Instruments

Airborne contamination monitors, surface contamination monitors, photon and neutron area monitors, and emergency instrumentation are discussed in the following subsections.

Airborne Contamination Monitors. Airborne contamination monitors, normally of the continuous type commonly referred to as CAMs, must meet the following criteria according to ANSI N317-1980 (ANSI 1980a). The minimum detection level of ^{239}Pu, in terms of maximum permissible concentration (MPC), shall be 8 MPC-h (8 DAC-h) at the point of sampling in the presence of nominal

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(a) American National Standards Institute (ANSI). 1988. Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions. Draft ANSI N42.17A-D9-1988, American National Standards Institute, New York, New York.

amounts of naturally occurring alpha emitters such as radon and thoron and their decay products. (No guidance is provided on what a "nominal" amount is, however.) The operating range shall be at least 100 minimum detection levels [i.e., up to 800 MPC-h (800 DAC-h) for ^{239}Pu]. Instrument error shall not exceed ±20% of the reading over the upper 80% of the operating range. The reproducibility of the system for any given measurement should be within ±10% at the 95% confidence level for a mid-scale or mid-decade reading. The instrument shall be capable of operating with less than a 5% change in calibration over the ambient temperature range expected. The instrument shall be equipped with an adjustable alarm set point (audible and visible alarms) that can be set at any point over the stated range. The air flow rate shall be indicated and adjustable. Voltage and frequency variations of ±15% within design values shall result in reading variations of no greater than 5% at the minimum detection level.

The primary purpose of any CAM is to detect the presence of airborne radioactivity and activate an alarm to warn personnel in the area so that actions can be taken to minimize personnel exposures (ANSI 1980a). The goal for any CAM should be to perform this function as quickly as possible and at the lowest detectable level of radioactive airborne concentration. The quantity of airborne radioactivity that will result in an alarm within a given time interval is defined in units of DAC per hour (DAC-h) for a particular radionuclide and is a function of the nuclide's airborne concentration in DACs, the sampling rate, the lower limit of detection of the instrument, and the time needed for the alarm to occur. Mishima et al.[a] provides guidance on each of these functions.

Draft ANSI N42.17B-D5[b] provides additional performance criteria for air monitors used to detect plutonium. This standard provides specifications for general criteria (sampler design, units of readout, alarm threshold, etc.),

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(a) Mishima et al. 1988. Health Physics Manual of Good Practices for the Prompt Detection of Airborne Plutonium in the Workplace. Draft Report, Pacific Northwest Laboratory, Richland, Washington.
(b) American National Standards Institute (ANSI). 1987. Performance Specifications for Health Physics Instrumentation - Occupational Airborne Radioactivity Monitoring Instrumentation. Draft ANSI N42.17B-D5, American National Standards Institute, New York, New York.

electronic criteria (alarms, stability, response time, coefficient of variation, and line noise susceptibility), radiation response, interfering responses (radiofrequency, microwave, electrostatic, and magnetic fields), environmental criteria (temperature, humidity, and pressure), and air-circuit criteria. More detailed specifications are provided in draft ANSI N42.17-D5 than in ANSI N317-1980 (1980a); however, the environmental criteria and the limits of variation are not as restrictive as those in ANSI N317-1980. With respect to accuracy, ANSI N317-1980 requires $<\pm 20\%$ and draft ANSI N42.17B requires 40% at the 95% confidence level. For the environmental criteria, ANSI N317-1980 requires that the readings change less than 5% under ambient conditions, while draft ANSI N42.17B-D5 gives a 15% limit of variation. As discussed previously, criteria from the draft standard are more applicable because they are supported by instrument testing.

ANSI N13.1-1969 (ANSI 1969) provides detailed guidance on sampling methods. One criterion that relates to CAMs is that air sample lines between air inlet and filter media shall be eliminated where possible and where not possible they shall be designed to meet the sampling criteria contained in the standard (e.g., short lines, proper sampling rate, smooth bends). The use of tygon tubing as sample lines should be minimized or eliminated. Air in-leakage from surrounding areas can be a problem when using sampling lines. Testing for air in-leakage shall be performed at least annually or when seals or "O" rings are replaced.

Surface Contamination Monitors. Surface contamination monitors include hand and/or shoe counters and instruments (or probes) with sufficient flexibility to survey pieces of equipment, including exterior clothing. ANSI N317-1980 (ANSI 1980a) states that these instruments shall have an audible alarm, a frequency that is proportional to the count rate or a preselectable trip setting, and upon reaching that level shall activate an audible or visible alarm, or both. These instruments shall be calibrated according to the requirements in ANSI N323-1978 (ANSI 1978) and be equipped with a traceable check source. Fixed instruments should be powered by alternating current (AC) and provided with an emergency power source.

Photon and Neutron Area Monitors. Photon and neutron area monitors measure the intensity of photon and neutron radiation in areas where significant quantities of plutonium are stored and/or handled. ANSI N317-1980 (ANSI 1980a) states that these monitors shall have a preselectable trip setting with audible annunciators and shall provide electronic signals for remote alarms if they are used as alarming devices. Each instrument shall also be equipped with a visual meter or digital readout. All neutron and photon area monitors should be AC-powered and provided with an emergency power source. Many of the requirements that apply to portable survey instruments as stated in ANSI N317-1980 (ANSI 1980a) and draft ANSI N42.17A[a] also apply to this type of instrumentation. Calibrations shall be performed according to the requirements in ANSI N323-1978 (ANSI 1978).

Emergency Instrumentation

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

Criticality Alarm Systems. ANSI/ANS 8.3-1986 (ANSI 1986) discusses the performance and design criteria for criticality accident alarm systems. The criteria include the following:

- Criticality alarm systems shall be designed to detect immediately the minimum accident of concern; the minimum accident may be assumed to deliver the equivalent of an absorbed dose in free air of 20 rad at a distance of 2 m from the reacting material within 60 seconds.
- Systems shall be designed so that instrument response and alarm latching shall occur as a result of radiation transients of 1-millisecond duration. The alarm signal shall be for evacuation purposes only and of sufficient volume and coverage to be heard in all areas that are to be evacuated. Very high audio background noise in some areas may require that the alarm be supplemented with

(a) American National Standards Institute (ANSI). 1988. Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions. Draft ANSI N42.17A-D9, American National Standards Institute, New York, New York.

visual signals; however, high background noise is a dangerous situation that should be prevented by design. Instrument response to radiation shall be calibrated periodically to confirm the continuing performance of the instrument. The calibration interval may be determined on the basis of experience but shall be no less frequent than annually. Tests should be performed at least monthly and the results of testing shall be documented.

The standard does not quantify criteria for reliability or the rejection of false alarms. Consideration shall be given to the rejection of false alarms as accomplished by placing three detectors near the material of interest and using a logic circuit. This circuit requires that two of the three detectors shall read high for an alarm to sound.

Fixed Nuclear Accident Dosimeters. All DOE facilities that have sufficient quantities and kinds of fissile material to potentially constitute a critical mass shall provide nuclear accident dosimetry. Requirements for fixed nuclear accident dosimeters are found in DOE 5480.11 (DOE 1988). A review of these requirements is found in Section 3.6.5.

Effluent Monitors. Facilities that deal with unencapsulated plutonium shall have continuously operating effluent monitors to determine whether or not plutonium is being released to the environment. Effluent monitor criteria in IEC Publication 761-1 (IEC 1983c) and IEC Publication 761-6[(a)] should be performed. Similar to airborne contamination monitors, effluent monitors shall be tested for air in-leakage at least annually or when seals or "O" rings are replaced.

Other Emergency Instrumentation. Other emergency instrumentation shall provide ranges for all radiation dose rates and contamination levels potentially encountered at the time of an accident. Normally, dose rate capabilities from a few millirem per hour to a few hundred rem per hour shall be required

(a) American National Standards Institute (ANSI). 1988. Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions. Draft ANSI N42.17A-D9, American National Standards Institute, New York, New York.

while capability requirements for the contamination level may range upward from 500 dpm/100 cm^2 for alpha contaminants and 2000 dpm/100 cm^2 for beta-gamma emitters. Performance specifications for emergency radiological monitoring instrumentation can be found in ANSI N320-1979 (ANSI 1979) and BNWL-1742 (Anderson et al. 1974).

4.2.3 Instrument Calibrations and Testing

Radiation doses and energies in the work areas shall be well characterized. Calibration of instruments shall be conducted where possible under conditions and with radiation energies similar to those encountered at the work stations. Knowledge of the work area radiation spectra and instrument energy response shall permit the application of correction factors when it is not possible to calibrate with a source that has the same energy spectrum. All calibration sources shall be traceable to recognized national standards. Neutron energy spectral information shall be considered particularly important because neutron instruments and dosimetry are highly energy dependent.

When the work areas have been well characterized, the calibration facility used by the plutonium plant shall be set up to represent as closely as possible the work area's radiation fields. Californium-252 or PuBe calibration sources should be used for work areas that process plutonium metal and plutonium oxide because their neutron energy distribution is similar to those compounds. Facilities that process PuF_4 should try to use a PuF_4 source. Most work areas at processing plants are high scatter areas and thus have significant quantities of low-energy neutrons. Because it may not be feasible to have sources and scatter geometries representative of all work locations at the facility, it shall be important to determine specific spectra and correction factors for work locations to correct for the calibration. Scatter conditions shall be taken into account when setting up a calibration facility. The effect of room scatter in a neutron calibration facility can be significant and may account for as much as 20% of the measured dose equivalent rate. The Schwartz and Eisenhauer (1982) methods shall be used to correct for room scatter.

ANSI N323-1978 (ANSI 1978) provides requirements on the calibration of instruments. The reproducibility of the instrument readings shall be known prior to making calibration adjustments. This is particularly important if the instrument has failed to pass a periodic performance test (i.e., the instrument response varies by more than ±20% from a set of reference readings using a check source) or if the instrument has been repaired. The effect of energy dependence, temperature, humidity, ambient pressure, and source-to-detector geometry shall be known when performing the primary calibration. Primary calibration shall be performed at least annually.

Standards referenced in Section 4.2.2 discuss specific performance testing of radiation detection instruments. Testing procedures in these standards shall be used for periodic requalification of instruments or detailed testing of instruments.

The calibration of photon monitoring instruments over the energy range from a few kiloelectron volts to several million electron volts is best accomplished with an x-ray machine and appropriate filters (NBS 1986) that provide known x-ray spectra from a few kiloelectron volts to approximately 300 keV. Radionuclide sources shall be used for higher energies. Most ion chambers used to measure photon radiations have a relatively flat energy response above 80 to 100 keV; ^{137}Cs or ^{60}Co are typically used to calibrate these instruments. These sources also shall be used to calibrate Geiger--Mueller (GM) type detectors. It should be noted that some GM detectors (e.g., those with no energy compensation) can show a large energy dependence especially below approximately 200 keV.

The calibration of alpha-detection instruments normally shall be performed with ^{239}Pu or ^{230}Th sources. Several sources of different activities shall be used to calibrate different ranges.

Whenever possible, beta detectors shall be calibrated to the beta energies of interest in the workplace. A natural or depleted uranium slab source can be used for calibration of beta detectors when beta radiations in the workplace have similar energies to the uranium. International Standards Organization beta sources (ISO 1983) shall be used for all other purposes. The energy dependence of beta detectors can be tested using the calibration sources listed in the ISO publication (1983); these include ^{90}Sr-^{90}Y, ^{204}Tl, and ^{147}Pm.

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

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

4.3 RADIATION PROTECTION PROCEDURES

A plutonium facility shall have a written policy on radiation protection including ALARA. All radiation protection procedures and controls shall have recognizable or formal technical bases for limits, methods, and personnel protection standards. Procedures shall be adequately documented, updated periodically, and maintained in a centralized historical file. A control system shall be established to assure accountability of all copies and that all new procedures are included in the historical files. A designated period of time for maintaining historical files shall be established. DOE 1324.2 (DOE 1980b) and ANSI N13.6-1972 (ANSI 1972) provide guidance on how long to maintain historical files. In addition, radiation protection procedures shall have a documented approval system and established intervals for review and/or revision. A tracking system should be developed to ensure that the required reviews and revisions occur.

Radiation protection procedures shall be provided for but not limited to the following topics:

- posting and labeling of facilities
- development and maintenance of all radiation protection records
- reporting of unusual radiation occurrences
- use of radiation monitoring instruments
- use of radiation sources (e.g., reference calibration)
- tracking of personnel medical exposures
- reporting of radiation exposures
- use of protective clothing
- responding to radiological emergency events
- surveying and monitoring
- counting room equipment and use
- instrument maintenance and control
- development and use of radiation work procedures
- responsibilities of operations staff for contamination control and personnel surveys.

Two topics, radiation work procedures (RWPs) and facility posting and labeling are discussed in more detail.

4.3.1 Radiation Work Procedures

Radiation work procedures (sometimes referred to as radiation work permits) shall be used for all radiation work. Each RWP shall contain provisions for protective clothing, work limitations, job descriptions, radiological conditions, and special instructions. Radiation work procedures shall be written by health physics staff with assistance from operating personnel who provide the information on the job to be performed. All RWPs shall be approved by the health physics staff and should be reviewed at least annually or more frequently under changing radiological or job conditions.

Radiation workers shall read and understand the applicable RWP before performing work in a radiation area. Radiation work procedures should be located at the access point to the controlled area (change room) for workers to review prior to entering the work area. To assure that workers read the RWP prior to entry, some facilities have a sign-off sheet that workers must sign stating that they have read and understand the RWP. Radiation work procedures should also be posted at the entrance to the work location for quick references by workers. Out-of-date RWPs shall be removed in a timely manner.

4.3.2 Facility Posting and Labeling

Areas in plutonium facilities shall be posted in accordance with the requirements in DOE 5480.11 (DOE 1988). The technical criteria and dose rate and/or levels for defining radiation, high radiation, very high radiation, contamination, and airborne radioactivity areas shall be established, documented, and consistently applied. The health physics staff shall establish and document the radiation levels that require

- areas to be barricaded and marked to prevent personnel from inadvertently entering them
- areas to be physically locked to preclude unauthorized personnel from entering them.

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

The health physics staff shall post or have readily available current radiation surveys of radiation areas at the health physics access control point for use in pre-job planning. Airborne activity areas shall also be posted to alert personnel of possible respiratory protection requirements.

4.4 CONTAMINATION CONTROL

The primary control for contamination in a plutonium plant is the facility design. Contamination is confined by enclosing the process areas and the ventilation systems. Section 3.0 of this manual addresses the different levels of confinement in a plutonium facility. The design objective for the confinement system shall be an essentially zero exposure of plant personnel and the public to airborne contamination as stated in DOE 6430.1 (DOE 1983) and DOE 5480.11 (DOE 1988). To ensure that this objective is met, additional requirements shall be implemented in the form of operational controls, radiation surveys, and the use of protective clothing.

In this section, the terms process area, controlled area, and uncontrolled area are used as defined in Section 3.4.

4.4.1 Operational Procedures

This section addresses the importance of procedures for housekeeping, maintaining accountability and personnel protection related to processing technology, using and storing gloves properly, and vacuuming in a plutonium facility.

Housekeeping

The three housekeeping practices listed below shall be followed in a plutonium facility:

- The inventory of contaminated and potentially contaminated scrap and equipment shall be kept to a minimum because all such materials are subject to special monitoring and accountability.
- Radioactive contamination shall be controlled and the spread of contaminants and the potential for accidents involving contaminants shall be minimized. (In at least one instance, poor housekeeping contributed to a serious criticality accident.) Management at all levels shall continuously emphasize the importance of good housekeeping and operating procedures shall be written to ensure good housekeeping practices.
- Conditions that lead to personnel exposure shall be minimized.

Where possible materials that are not absolutely necessary to an operation shall be kept out of the contaminated or potentially contaminated area. All packaging and unnecessary protective coverings shall be removed before materials are introduced into the process area. Likewise, items that are not necessary to the process shall be promptly removed, particularly from glove boxes, and not left to accumulate and become safety hazards, potential fire hazards, sources of radioactive (dust) accumulation, or sources of exposure.

Good housekeeping practices inside glove boxes should emphasize fire and explosion control. Only metal or nonflammable plastic containers shall be used for the accumulation of scrap and wastes of any kind in the glove boxes and throughout plutonium facilities. Accumulation of combustible materials in glove boxes shall be minimized. When explosive, flammable, or volatile liquids are allowed, they shall be rigidly controlled and used only in inert gas atmospheres unless a safety analysis review shows otherwise. All residues shall be removed immediately at the conclusion of each job or cleaning operation.

Considerable effort has been expended on the development of coated and corrosion-resistant tools. Some efforts have been marginally successful, but in most cases throw-away tools shall be favored. Electropolishing of contaminated metal tools and equipment has been shown to be a good method of decontamination and allows for their reuse in some cases. Where possible all tools with sharp edges or points (e.g., screwdrivers, ice picks) shall be kept out of glove boxes.

Management shall constantly demand good housekeeping. Mandatory, routine, cleanup periods are becoming more common due to the increasing cost of storing and disposing of contaminated materials. Better housekeeping is required due to real-time, computerized accountability for nuclear materials. It has been demonstrated that kilogram quantities of plutonium oxide dust can accumulate in glove boxes unless they are routinely cleaned. Much of the exposure to workers originates from layers of plutonium oxide dust on the surface of gloves and the internal surfaces of glove boxes. In processes where plutonium oxide powder is handled, the glove boxes should be cleaned weekly to reduce the accumulation of dust layers and reduce worker exposure.

Processing Technology

The processing technology that is unique to plutonium has resulted from the substance's radiotoxicity coupled with criticality restrictions. Security measures have resulted in additional, far-reaching restrictive complications in some areas. Special aspects of accountability and personnel protection impose many restrictions and require complicated process machinery and complicated maintenance.

Problems in the handling of plutonium solids are unique. The introduction of high-exposure plutonium (i.e., >10 wt% ^{240}Pu) is providing new challenges to the solids-handling industry with respect to the technology for personnel exposure control and automation.

The laser isotope separation process (Section 1.4.2) that produces pure plutonium isotopes has been developed recently. This process and its resultant waste streams present new and unique problems from a radiation protection standpoint. Where near-absolute containment will be required to avoid contamination of the workplace, the short half-life radionuclides (relative to product contamination, handling, methods, and equipment) shall be considered. For example, ^{238}Pu will require 300 times better containment than ^{239}Pu if equally low activity levels are to be maintained in work areas (see Table 1.2).

Gloves

To create a double barrier between the source and all extremities, surgeon gloves shall be worn in addition to the glove box gloves.

In general, black neoprene gloves are the standard glove box glove and the most economical to use where process conditions do not produce rapid glove deterioration. However, alpha particles from surface dust layers can induce surface cracking in black neoprene. Hypalon® is more resistant to surface cracking, acid deterioration, and ozone effects, and this characteristic will, in many cases, make Hypalon gloves the most economical, despite their higher unit cost.

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In recent years many new types of glove box gloves have been developed. Glove usage should be tailored to the particular needs of the job. For processes that require maximum dexterity, the 0.014-in. (0.038-cm) neoprene gloves are still superior. Coated Hypalon gloves are superior to neoprene for glove box process operations that involve nitric acid or ozone levels that may cause deterioration. Ethylenepropylenediamine monomer (EPDM) gloves are used in some facilities and have good flexibility and are resistant to degradation caused by radiation and ozone. Greenhalgh, Smith, and Powell (1979) reported that Hypalon and EPDM gloves have greater than 30 times the longevity of neoprene in low-level ozone concentration atmospheres. Viton® gloves have proven to have a longer life than neoprene gloves under many operating conditions, but suffer somewhat from stiffness. Where high gamma radiation levels are encountered, lead-loaded gloves may be necessary. However, their stiffness and workers' loss of manual dexterity shall be considered in determining their influence on work efficiency and the total dose received.

Persons who perform operations that involve microspheres of ^{238}Pu, coated or uncoated, shall be aware that the heat generation of a single 100- to 200-μm-dia sphere can melt through glove material. In addition, containment of a quantity of microspheres, especially coated microspheres, is difficult because of electrostatic repulsion. Microspheres have been observed climbing the walls of a glass beaker and spreading throughout a glove box.

Glove storage problems occur occasionally. Experiments and static tests have not provided an adequate explanation of the sporadic problems that have been encountered. Test results in which gloves were stored under different lighting conditions (ultraviolet and fluorescent) and under stressed conditions (creased or bent) have not been consistent. Tests of gloves seem to indicate that glove degradation is caused by the combined effect of ionizing radiation, ozone, and lighting. The glove inventory shall be rotated to prevent the inventory from becoming outdated while on the shelf.

All gloves in normal use at plutonium processing installations should be inspected prior to each use. All operating personnel shall perform contamination surveys of themselves after every glove usage. The glove inspections

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should be made each time by the same team of trained individuals, and the condition of each glove shall be recorded so that glove failures can be anticipated and preventative measures can be taken. The development of a statistical basis for establishing the frequency of glove changes shall be considered because it may be cost effective. For example, the change-out frequency could be planned so that gloves are changed at some fraction of the mean time between failures or more preferably some fraction of the minimum time between failures. This type of change-out program could also eliminate personnel doses and potential contamination spread incidents associated with too-frequent glove replacement. This procedure may require that each glove use be categorized. A routine replacement program will not replace an inspection program, but it is a supplement to the inspections. The surgeon's gloves should, of course, be surveyed after the inspection of each glove box glove. Gloves that are in questionable condition should be changed without delay. Gloves that are not in use for the remainder of that shift should be capped off with a glove cover or plastic bag. Gloves not in use should be stored inside the glove box in such a manner that they do not interfere with operations.

Vacuuming

The subject of vacuuming within the glove box is somewhat controversial. Experience has shown vacuuming to be the most effective and quickest way to clean a controlled-atmosphere (dry) glove box. It is not particularly effective for high-humidity or wet-process glove boxes, particularly those that involve acids. After acids have been used in a glove box, washing and wiping is the only method to clean the etched surfaces.

Vacuuming is particularly effective in dry-atmosphere and inerted enclosures where the levels of radioactive dust quickly increase personnel exposure. In many cases, vacuuming reduces the exposure level more than a wipe-down with a damp cloth, and it can be done more quickly and with less waste material generated. Two factors weigh against vacuuming: possible safety hazards from electrical sparks, and the occasional difficulty of operating in inert atmospheres (although the last item need not be of importance). However, in dry glove boxes with dusty operations using high-exposure plutonium, personnel exposure control is a problem and vacuuming is a quick and effective method of keeping the dust and exposure rates under control. Under these conditions consideration shall be given to the use of dust removal by vacuuming.

4.4.2. Radiation Surveys

Radiation workers are often assigned tasks that conceivably could expose them to radioactive material. It is not sufficient to rely exclusively on equipment design to minimize contamination and exposure in the workplace. A radiation protection program shall include monitoring of both the workers and the conditions in the workplace. Both functions are essential to a good radiation monitoring program.

Monitoring the Workplace

Monitoring of the workplace is an essential element of every routine surveillance program. It can be effectively accomplished using any or all of the techniques that are discussed in this section. The rigor with which all of the various elements of a radiation monitoring program are applied shall be tailored to meet the needs of the individual work areas and shall depend on the kind and quantity of radioactive material present and its potential for dispersibility. Each program shall be designed to meet existing needs, but also shall be flexible to allow for incorporation of the possible advantages to be provided by the various available monitoring practices. Monitoring practices include, but are not limited, to the following:

- contamination surveys of the workplace
- release surveys
- external exposure surveys
- airborne contamination surveys
- routine surveillance by an RPT.

Contamination Surveys of the Workplace. To characterize the various aspects of a radiation monitoring program, the process area and controlled areas have been further classified into Class A, B, or C work stations as presented here:

- At a Class A work station, work with plutonium is conducted in one or more glove boxes and the annual exposures might exceed 10% of the dose equivalent limits.

- At a Class B work station, work is performed in open-face hoods and the annual exposure will exceed 1% but not more than 10% of the dose equivalent limits.

- At a Class C work station, normally found in a controlled area, work with very small quantities of plutonium (levels of activity on the order of environmental contamination levels) is performed in ordinary laboratory facilities.

The radiation monitoring program based on this classification of work stations is suggested only for reference. The values of 10% and 1% of the basic limit for occupational exposure are thus a reference level used for managing worker exposures; they are not a limit. Variations in the program should be based on the operating experience, training of the operating people, age of the facility, type of work that is involved, and any environmental conditions that could affect the contamination status (i.e., the presence of ozone, excess heat, chemical fumes that could deteriorate the barrier, or the presence of sharp objects or rough surfaces that could penetrate the barriers). Where a mixture of work stations exists in one room, the features of the program should encompass the requirements of the highest classification of work station in that room.

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

All workplaces shall be monitored for contamination levels on a regularly scheduled basis. The frequency of such surveys will depend on the potential for dispersibility of the radioactive material. As a minimum, all gloves, work surfaces, floors, equipment, etc., within the workplace should be surveyed according to the frequencies listed below.

- Class A - Daily survey if there is $\geq$1 mCi within the work station
  - Weekly survey if there is $\leq$1 mCi
- Class B - Weekly survey
- Class C - Monthly survey

The change room and other support facilities within the controlled area shall be surveyed for contamination daily. Continuous air monitors, step-off pads, and hand and shoe counters should be surveyed daily or once per shift in support of the weekly and monthly surveys. These frequent surveys are also part of the routine surveillance program and permit immediate follow-up if low-level contamination is detected to minimize major incidents. Some fixtures and support areas outside the controlled area, such as door knobs and the lunchroom, should also be surveyed daily. Other support areas should be surveyed monthly. If routine survey results are elevated for a given area, more detailed surveys should be performed to determine the extent of the contamination. The cause of elevated readings also should be investigated.

In order to preclude the possibility that contaminated waste would be disposed of as ordinary waste, 1) all process and controlled area waste shall be considered contaminated, and 2) mechanisms shall be established that prevent the mixing of contaminated and non-contaminated waste.

Release Surveys. DOE 5480.11 (DOE 1988) provides requirements for the release of materials and equipment for conditioned use in controlled areas.[a] Material and equipment shall not be released from radiation areas to controlled areas if either of the following conditions exist:

- Fixed or removable contamination levels on accessible surfaces exceed, on the average, 100 dpm/100 cm^2 or 20 dpm/100 cm^2, respectively.
- Contamination levels on inaccessible surfaces are likely to exceed, on the average, 100 dpm/100 cm^2 fixed contamination or 20 dpm/100 cm^2 removable contamination based on prior use of the material or equipment in the radiation area.

The requirements for unrestricted release of materials and equipment are found in draft DOE 5400.XX.[b] Material for unrestricted release shall have

(a) As defined in DOE 5480.11 (DOE 1988) a controlled area is "any area to which access is controlled in order to protect individuals from exposure to radiation and radioactive materials."

(b) U.S. Department of Energy (DOE). 1988. Radiation Protection of the Public and Environment. Draft DOE 5400.XX. U.S. Department of Energy, Washington, D.C.

less than 20 dpm/100 cm^2 removable transuranic contamination and, on the average, less than 100 dpm/100 cm^2 of fixed transuranic contamination. More detailed guidance on conditions for unrestricted release can be found in HPSSC N13.12-1988 (Health Physics Society Standards Committee 1988) and NRC Regulatory Guide 1.86 (1974).

External Exposure Surveys. Measurements of external exposure should be made at all locations where personnel exposure occurs at the time a program is established, to delineate the levels involved. Additional photon and neutron measurements should be made at the same frequency as the contamination surveys for a Class A facility to identify the buildup of plutonium in HEPA filters and glove boxes. Glove box exhaust filters should be changed with sufficient frequency to ensure that the resultant exposure from filter buildup will contribute less than 10% per year to the total exposure of operating personnel.

Airborne Contamination Surveys. Airborne contamination surveys should be performed for:

- prompt detection of airborne contaminants for worker protection
- personnel exposure assessment
- monitoring of trends within the workplace
- special studies.

Of primary importance is the prompt detection of airborne contaminants.

The rapid, early detection of airborne releases requires knowledge of the potential sources and characteristics of the airborne material, the locations of the personnel who are at risk, and the capabilities of the detection devices. Optimally, the samples shall be taken between the source and the person to intercept the airborne materials before they reach the individual. With the numerous sources and mobility of the workers, interception under all conditions is difficult, if not impossible, to achieve. Samples of airborne materials shall be taken as close to their points of origin as practicable to maximize the probability of their detection (airborne concentrations are at a maximum at their points of origin).

Fixed probes that are positioned to intercept releases from recognized major potential sources should be used along with portable air samplers for planned activities with known potentials for airborne release of contaminants and for temporary storage of contaminated materials in areas of low air flow. If the workplace exhaust system can be shown to provide rapid, essentially quantitative clearance of airborne contamination, fixed probes that sample the exhaust system may be adequate for routine coverage of nonplanned activities. If justified by documented studies, other sampling arrangements that provide improved "total" coverage of the workplace environment for the early detection of airborne contamination may be used.

Mishima et al.[a] provide additional guidance on airborne contamination surveys. The document indicates that those responsible for the rapid and reliable detection of airborne plutonium shall consider the following areas in evaluating monitoring systems and working environments:

- the source and characteristics of the airborne plutonium released to the workplace
- the airflow patterns and airborne transport of plutonium in the workplace
- the location of personnel within the workplace during various processing procedures
- the location at which the airborne plutonium sample should be intercepted before the sample is inhaled by workers
- the ability of the system to transport an undistorted sample to the collection media or measurement device
- the collection and retention efficiency of the collection medium
- the efficiency of the measurement device in measuring the plutonium collected and differentiating the plutonium from other materials present
- the accuracy and reliability of the system.

Guidance for each area listed above is provided in Mishima et al.[a]

(a) Mishima, J. et al. 1988. Health Physics Manual of Good Practices for the Prompt Detection of Airborne Plutonium in the Workplace. Draft Report, Pacific Northwest Laboratory, Richland, Washington.

Routine Surveillance by a Radiation Protection Technologist. Continuous radiation monitoring shall be provided during the periods of high or unusual risk associated with the work in the area. Periods of high or unusual risk include the potential or actual breaching of the integrity of the glove box or associated systems, including breaching for maintenance (replacement of panels, glove changes, bag-out operations, replacement of filters, or repair of vacuum systems). Work that involves the use of temporary enclosures (greenhouses) shall be provided with continuous coverage by an RPT.

Monitoring of the Worker

Monitoring the worker is necessary, not only to ensure that a potential intake is detected promptly and that the resulting internal dose is assessed, but to ensure the effectiveness of the overall radiation protection program and confirm the integrity of the engineered containment system.

There are several types of worker monitoring, some during and immediately following work with radioactive material and some scheduled for a later time at a preset frequency. This section addresses only the methods of monitoring the worker at the work station. Other methods are discussed in the section that deals with internal and external exposure controls.

Techniques to monitor the individual worker at the work site include:

- frequent/routine surveys of gloves
- exit surveys
- nasal swipes
- personal air sampling.

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

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

A contamination survey of shoes, clothing, and hands shall be required prior to leaving a plutonium work station. Following routine work, self-surveying upon exit usually shall be considered adequate, if the person has received proper training in the use of the instrument provided. The instrument shall clearly detect an unacceptable level of contamination.

After performing work that involves a high potential for intake of radioactive material (where continuous coverage by an RPT has been required), the RPT shall provide an exit contamination survey of the worker.

Nasal Swipes. After performing work that, in retrospect, involved a high potential for intake of radioactive material, each worker shall be required to provide a swipe of the nasal passages, which shall be counted immediately. If respiratory protection was worn there is no need for nasal swipes unless a breach of the respirator seal is suspected. If facial contamination is detected at the exit contamination survey, a nasal swipe should be counted immediately. Guidance on the actions to be taken if a nasal swipe is positive can be found in Sections 4.6 and 4.7.

Personal Air Sampling. Radiation protection programs at plutonium and uranium processing facilities in Europe have effectively used personal air sampling for the past decade (Jones et al. 1983). Experience in this country suggests that while this could be very useful, the low flow rate of currently available models limits their usefulness due to problems in obtaining representative samples. An evaluation of the relative merits of personal air samplers was performed by Ritter et al. (1984). The use of personal air sampling programs should be considered to monitor individuals workers for exposure to airborne plutonium.

4.4.3 Protective Clothing

Various types of protective clothing, including laboratory coats, shoe covers, gloves, coveralls, underwear, plastic or rubber suits, and air-purifying or atmosphere-supplying respiratory protective equipment, may be required for operations with transuranic radionuclides. The use of company-issue shoes and clothing for employees with work assignments in process areas can be a major aid in contamination control.

As a minimum, personnel who perform operations in controlled areas should wear coveralls and shoe covers. For inspections or visits, lab coats and shoe covers may be permissible. When contaminated wet areas are to be entered, water-repellent (plastic or rubber) clothing shall be worn. No personal outer clothing should be permitted under coveralls.

Shoe covers or company-issue shoes shall be worn except where wet operations are involved; in these areas, rubbers or waterproof shoe covers shall be provided.

Hands shall be protected by a minimum of two barriers, for example, at least one pair of surgeon's gloves and one pair of glove box gloves. Where manual dexterity is not required and the work involves a potential for piercing one or both layers of rubber gloves, leather gloves should be worn over the surgeon's gloves. Automated methods should be considered for replacing routine manual methods that have a high risk of piercing the gloves.

Protective clothing should be removed at the radiation zone step-off area and personnel monitoring for contamination shall be performed. If this is not practical, strict control of the movement of personnel from the step-off area to a location where protective clothing can be removed shall be maintained. personnel wearing protective clothing shall not be allowed to mingle with individuals in personal (street) clothing. Protective clothing shall not be allowed in non-controlled areas such as offices, lunchrooms, control rooms, etc.

Special Maintenance Requirements

For special maintenance work that involves significant quantities of plutonium, a double barrier concept shall be implemented. An example of minimum requirements for protective clothing is provided below:

- two pairs of coveralls (and sometimes a plastic suit)
- canvas boots taped to the inner pair of coveralls, with rubbers over the canvas boots
- one pair of surgical gloves taped to the inner coveralls, with a leather, cotton, or rubber outer pair of gloves
- respiratory protective device with hood taped to respirator.

Respiratory Protection

Respiratory protection shall be readily available. Respiratory protective equipment should be used for all bag-out operations, bag and glove changes, and any situation involving a potential or actual breach of confinement. Protection, in the form of air-purifying or atmosphere-supplying respirators, shall be used whenever concentrations of radionuclides in the air are likely to exceed the DACs. For good performance, the respirator shall fit closely on the facial contours and make an impenetrable seal so that all air enters through the filter or is supplied by the breathing-air supply. ANSI Z88.2-1980 (ANSI 1980b) describes qualitative and quantitative tests that shall be used to ensure that the respirator fits the individual; only the quantitative test shall be used for verification of respirator fit at plutonium facilities. Respirator fit tests shall be performed annually. Respirators shall be approved by the National Institute of Occupational Safety and Health (NIOSH).

The approved, full-face, high-efficiency-filter, air-purifying respirator may have a maximum protection factor of up to 1000 as measured using a quantitative fit test on each person. If only a qualitative test is performed, a maximum protection factor of 100 is permitted. Filter type respirators are considered ineffective against radioactive materials in gaseous form. In its Publication 87-11-6, Guide to Industrial Respirator Protection, NIOSH recommends that a protection factor for filter-type respirators be lowered from 100 to 50 (NIOSH 1987).

Full-face, supplied-air respirators have protection factors as measured on each person but generally limited to a protection factor of 10,000. The protection factor for full-face, supplied-air respirators shall be applicable for either particulate or gaseous materials, except for those gaseous materials that may be absorbed through the skin.

For air concentrations that exceed those permitted by the protection factor for the respirator used, either an air-supplied hood or a self-contained breathing apparatus shall be used. Self-contained breathing apparatuses (positive pressure closed-circuit type) have a protection factor of at least 10,000 (ANSI 1980b).

Air-supplied hoods are becoming more popular because a fitting is not required and facial hair does not prohibit their use. Protection factors greater than 1000 have been determined with air-supplied hoods. Air-supplied hoods require NIOSH approval.

Half-face respirators should not be used in plutonium facilities because of the difficulty in maintaining a seal during typical work activities.

The elements of a satisfactory respiratory protection program are described in ANSI Z88.2-1980 (ANSI 1980b). Some of the features that are important to plutonium facilities are described below.

- All respirators and filters shall be tested for adequate performance prior to their use.
- Individual fit tests shall be performed. If glasses are worn by the individual, respirator glasses shall be purchased and worn during the fitting.
- After each use, respirators shall be retested for proper operation, surveyed to ensure that they are not contaminated, and sterilized prior to reuse.
- Employees who use respirators shall receive training in the performance features of the respirators and how they shall be used. Emphasis shall be given to proper donning of the respirators and testing for a good face fit after donning the respirators.
- Care shall be taken to ensure that every air-purifying respirator is equipped with a filter and that every atmosphere-supplying respirator is used with approved air supplies. (Serious accidents have

resulted when respirators were used with filters not in place or when atmosphere-supplying respirators were connected to improper supplies.)

- Standard fittings that meet the recommendations of national standards groups shall always be used. However, fittings for personal breathing air equipment shall be unique to the personal breathing air equipment and shall not fit any other gas supplies in the facility.
- All respirators whose performance is suspected to be defective shall be removed from service.

4.5 CRITICALITY SAFETY

The health physicist staff should also have a basic understanding of program structure, engineering criteria, and administrative controls as related to nuclear criticality safety. However, the health physicist's primary responsibilities with regard to nuclear criticality safety include emergency instrumentation and emergency response. The health physics staff should be knowledgeable in the following areas:

- criticality alarm systems, including alarm design parameters, types of detectors, detector area coverage, alarm set-points, and basic control design
- postulated criticality accidents, including information on types of criticality accidents (e.g., burst, multi-burst)
- magnitude of accidents (number of fissions, neutron flux, energy distribution, fission gamma rates, and potential fission gas release)
- locations and scenarios for designing the fixed nuclear accident dosimetry program and formulating plans for emergency response.

The health physics staff shall maintain an adequate monitoring capability for a nuclear criticality accident. In addition to the criticality alarm systems and the fixed nuclear accident dosimeters, remotely operated high-range gamma instruments, personal alarming dosimeters for engineering response/rescue teams, neutron-monitoring instrumentation (in case of a sustained low-power critical reaction), and an air sampling capability for fission gases shall be maintained.

The health physics staff shall assure that emergency response personnel are trained in quickly identifying criticality situations and potential additional hazards. Response personnel must be able to quickly determine the type of criticality, the difference in exposure hazard, and the methods for quickly estimating exposure levels for rescue or mitigation of the incident. The health physics staff also shall provide support to medical personnel in treatment of exposed/injured workers.

For additional information on critical safety requirements and guidance, the reader is referred to DOE 5480.5, Safety of Nuclear Facilities (DOE 1986b), ANSI/ANS 8.3-1986, Criticality Accident Alarm Systems (ANSI 1986), ANSI/ANS 8.1-1983, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (ANSI 1983), and ANSI/ANS 8.19-1984, ANS Administrative Procedures for Nuclear Criticality (ANSI 1984).

4.6 INTERNAL DOSIMETRY AND EXPOSURE CONTROL

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

It is DOE policy that facilities are designed and operated to prevent intakes of radioactive materials. Radiological controls for the workplace should ensure that radionuclides are contained 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, however, as the result of inadvertent intakes of radioactive material. Experience has shown that the most common route for inadvertent plutonium deposition in man is by inhalation. Intakes may also occur by accidental ingestion or by wound contamination. Surveillance programs should be designed to rapidly detect a release in the event of a loss of radioactive material containment. Internal dosimetry programs should be tailored to the needs of each plutonium handling facility so that inadvertent intakes are discovered and quantified, and the worker's effective dose equivalent is determined by appropriate methods.

When workers are inadvertently exposed to radioactive material, appropriate corrective action should be taken to ensure that control and containment have been re-established. To control occupational exposure, an early assessment of probable severity, preferably within the first two hours of the intake, is desirable to facilitate a decision on decorporation therapy. Preliminary assessment of severity (based on estimates of probable intakes) is necessary to provide guidance to both the medical professionals involved and to 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 that may be of use in specific cases include:

- a description of the internal exposure scenario
- air concentration and exposure duration
- skin and clothing contamination
- nasal or saliva smears
- direct (in vivo) bioassay measurements
- indirect (in vitro) bioassay measurements.

4.6.1 Performance Capabilities for Internal Exposure Monitoring

The requirements in this section define the 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 containment 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 follow up with measurements of radioactive material in the body and in samples of excreta collected from workers with known or suspected intakes so that appropriate dose assessment may be performed. Retrospective monitoring also involves measurements to confirm a suspected intake of radioactive material, 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 from such intakes.

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 (Sections 4.2.2 and 4.4.2). However, prospective bioassay measurements may complement the air monitoring program.

In the design and evaluation of the monitoring required to meet DOE 5480.11 (DOE 1988), 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. The most current ICRP guidance on plutonium behavior in the body may be found in ICRP 48 (1986). The ICRP models were primarily designed for prospective radiation protection planning. The average systemic compartment partitioning and clearance half-times for plutonium given in ICRP 48 are shown in Table 1.7. No generalized model is appropriate for retrospective dose assessment when specific data are available.

If a release of radioactive material is detected that could result in an intake delivering a committed effective dose equivalent of 100 mrem or more, then workplace monitoring results should be used initially to estimate intakes by the workers involved.

It is difficult to meet the above performance capabilities for some radioactive materials such as enriched uranium oxides, thorium oxides, and insoluble transuranics. If the performance capabilities cannot be achieved, the following should be applied in lieu of the above requirements:

- The best practicable (state-of-the-art) prospective monitoring capability should be used.
- Internal exposures should be prevented by enhanced facility design, operation, controls, and personnel protection equipment.
- 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, and therefore 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
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.

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

- ensure that additional control measures are applied to prevent intakes of radioactive material
- document these additional control measures for auditing purposes

- upgrade bioassay measurement systems and air monitoring equipment to provide state-of-the-art measurements
- ensure that the best technology is available for internal dose assessments.

All confirmed occupational internal exposures, regardless of magnitude, should be assessed. The results of all bioassay and other measurements needed to support the quality of measurements and dose assessment should be recorded and maintained. Recording and reporting requirements for internal dosimetry data are set forth in DOE 5480.11 (DOE 1988) and DOE 5484.1 (DOE 1987a). Further guidance on recording and reporting is given in Guide to Implementation of DOE Orders for Internal Dosimetry.[a]

The assessed calendar year effective dose equivalent resulting from intakes of radionuclides during the year, based on confirmed bioassay measurements, should be recorded. With respect to evaluating conformance to occupational exposure limits, DOE 5480.11 (DOE 1988) requires provisions for assessing the annual effective dose equivalent for as long as the annual effective dose equivalent is 10 mrem or greater. Separate recording of the calendar year individual organ dose equivalent is not required if it is 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.

4.6.2 Protection of the 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 because the effective dose equivalent limits for these individuals are the same as for the general public.

4.6.3 Guidelines for Establishing Internal Dosimetry Programs

The need for an internal dosimetry program is determined by the potential for plutonium intakes and the associated risks to health. An internal

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(a) Currently in preparation by the DOE Expert Group on Internal Dosimetry.

dosimetry surveillance program is required for all work with unencapsulated plutonium. The type of surveillance procedures used to monitor the workplace and the type and frequency of bioassay measurements chosen should be based on the previously discussed performance capabilities for workplace monitoring programs and the probability of inadvertent releases that could result in intakes by workers.

The internal dosimetry program should include the following:

- internal exposure surveillance to ensure that engineered confinement (Section 3.4.3) and facility monitoring programs (Section 4.4) are effective and that there is no evidence of a loss of control
- intake detection and assessment capability for prompt evaluation of accidental intakes
- procedures to ensure that prompt therapeutic measures are taken following a significant intake of plutonium
- retrospective bioassay measurements to estimate the internal deposition and determine the effective dose equivalent.

These procedures should verify that engineered safeguards and administrative controls have been effective. A prospective monitoring program should be implemented even though work involves only encapsulated plutonium. A minimum program will involve an annual check of the integrity of encapsulated sources.

Bioassay Sample Collection

Bioassay samples or measurements should be classified according to their primary purpose, as follows:

1. Baseline (pre-employment) bioassay measurements to establish the individual's status relative to any previous radionuclide intakes. Such measurements should be made for newly hired employees, intra-company transferees, and other workers transferring from facilities where bioassay measurements were also required.
2. Prospective (or "routine") bioassay measurements to detect intakes of plutonium and other radionuclides. These measurements also may verify the effectiveness of containment and airborne contamination monitoring programs.

3. Retrospective (or "special") bioassay measurements for the assessment of plutonium and its decay products in the body. These analyses provide an estimate of the intake or deposition of radionuclides for dose calculations and for long-term follow-up and management of such intakes.

4. Termination bioassay measurements to document the status of plutonium deposition at time of termination of employment.

Bioassay sample classification is important because the purpose of a sample may affect the collection method chosen and the subsequent method of analysis. Single void urine samples, for example, are not adequate for plutonium exposure control purposes; samples representative of excretion over a 24-hour period should be collected for intake assessment. The date of sample collection is also important; however, the specific times when prospective bioassay samples are collected are usually not important for plutonium. Time of sample collection may be very important, however, in the interpretation of retrospective bioassay results.

Appropriate baseline samples and measurements should be obtained from all newly hired employees who will be working with plutonium and other radionuclides. The results of these measurements should be available and reviewed before the newly hired employee begins to work with plutonium. Measurements made by a previous facility upon termination of employment may be acceptable for this purpose.

Termination bioassay samples and measurements document the amount of radionuclides being excreted, and may be used in estimating the amounts deposited in internal organs of a terminating or transferring employee at the time of separation.

Air Sampling

Air sampling programs should be designed to rapidly detect any inadvertent release of plutonium (Sections 4.2.2 and 4.4.2). Air monitoring detection capabilities should be sufficient to detect releases of airborne plutonium, which, if inhaled by a worker, could result in a committed effective dose equivalent greater than 100 mrem.

Source Containment

Encapsulation or containment integrity should be verified periodically; the time interval between checks should be established in the facility's Quality Assurance and Radiation Protection plans. Containment integrity is verified by conducting periodic leak tests of the encapsulated source. Plutonium sources with removable alpha contamination greater than 10 pCi/100 cm^2 should be considered to be leaking. Appendix A of ANSI N542-1977 (ANSI 1977) describes several methods that may be used to test the integrity of encapsulated sources.

4.6.4 Prospective Internal Dosimetry Surveillance

Routine internal dosimetry surveillance consists of:

- baseline bioassay measurements to establish the status of employees relative to previous intakes
- routine measurements of surface contamination in the workplace
- continuous air monitoring to evaluate the potential for intakes of plutonium
- prospective bioassay measurements to detect inadvertent intakes of plutonium.

The first step in the design of a prospective surveillance program is to characterize the potential for intakes of plutonium and other radionuclides that may be present in the facility. Considerations should address the following:

1. radionuclide descriptions, including
   a. range of activities
   b. mixtures of plutonium isotopes
   c. plutonium daughters (such as ^{241}Am)
   d. other radionuclides
2. chemical and physical form descriptions, including
   a. chemical forms of plutonium
   b. expected lung clearance rates
   c. physical forms of plutonium
   d. particle size distributions

3. evaluation of intake mechanisms, including
   a. possible routes of intake
   b. activity dispersal mechanisms
   c. intake rate (acute or protracted).

The second step is to maintain a quality program for routine measurement of the workplace, such as surface contamination monitoring, air sampling, and continuous air monitoring. The results of these measurements should be used to indicate when retrospective bioassay measurements are needed. The routine bioassay measurements program should show that workplace measurements are effective in identifying conditions that result in confirmed intakes of plutonium.

Workplace Measurements

Levels of surface contamination, airborne radioactivity, skin contamination, and nose swipe activity that are expected to result in a committed effective dose equivalent of 100 mrem to a worker should be determined. When these levels are exceeded, the individuals involved should be placed on a retrospective bioassay program. Surface contamination monitoring, air sampling, continuous air monitoring, and surveys of skin and clothing of workers should be performed routinely or whenever necessary to determine the effectiveness of radiological control efforts.

Surface Contamination Monitoring. Rooms and work areas in plutonium facilities occupied by workers should be surveyed routinely for removable and fixed plutonium contamination using alpha survey meters. Section 4.4.2 provides detail on surface contamination monitoring, including classification of work stations, survey procedures, survey frequencies, and release surveys.

The amount of removable contamination may be related to airborne contamination using resuspension factors. Resuspension factors are dependent on the type of surface contaminated, such as smooth paint, rough concrete, or porous wood, and on the nature of activities taking place in the room or area measured. There are presently no well-established contamination-control levels that, when exceeded, will require follow-up bioassay measurements.

Each facility should determine the relationship between surface contamination levels and the probability of a resulting intake to determine trigger levels for initiating retrospective bioassay.

Air Sampling and Continuous Air Monitoring. Rooms or work areas that contain plutonium in unsealed form should have air sampling and continuous air monitoring systems if there is a potential for internal exposure that could result in a committed effective dose equivalent exceeding 100 mrem, considering all radionuclides to which the individual is potentially exposed during the year.

The results of air sampling and continuous air monitoring should be used to initiate retrospective bioassay to assess intakes of plutonium (greater than 40 DAC-h for the year from all radionuclides). Analyses should be made and documented to determine the degree to which air sampling and continuous air monitoring results are representative of air breathed by individuals (Mishima et al. 1987) in that area.[a]

Continuous air monitoring and air sampling systems for particulates should be evaluated to determine the appropriate type of collection filter to be used, the minimum detectable air concentration, the frequency of filter change, and the dust-loading correction factor. This evaluation should be documented. Room-air exhaust samplers should be isokinetic to insure that a representative sample enters the inlet of a sampling tube when sampling from a moving aerosol stream; an isokinetic sampler is one that samples at the same velocity as the air stream passing the sampler and is aligned parallel to the stream. Air sampling system calibration and quality control procedures should be maintained and reviewed periodically.

If air sampling and continuous air monitoring cannot provide results that are reasonably representative of the air concentrations to which individuals are exposed, a program of breathing-zone air sampling should be initiated.

(a) Mishima, J. et al. 1988. Health Physics Manual of Good Practices for the Prompt Detection of Airborne Plutonium in the Workplace. Draft Report, Pacific Northwest Laboratory, Richland, Washington.

The air sampling program may include both fixed-location and lapel breathing-zone air samplers. Breathing-zone air samples are more representative of actual exposure conditions than are samples collected at fixed locations. Sections 4.2.2 and 4.4.2 provide further detail on air sampling and monitoring and include discussion of measurement performance criteria, appropriate monitoring locations, and use of breathing-zone samplers.

Personnel Contamination. Measurements of contamination on workers may be useful as indicators of when to perform bioassay measurements. The criteria used for initiating these retrospective measurements should be documented.

Plutonium contamination on skin surfaces, particularly on the face, indicates that an intake may have occurred. Contamination in the nostrils may be detected by gently wiping a cotton swab or other absorbent material on the inside of each nostril. A separate swab is necessary for each nostril. The amount of plutonium contamination removed can be determined by alpha counting.

Contamination on protective clothing, particularly the layer of clothing nearest the skin, may also be useful as an indicator that an intake of plutonium has occurred. If plutonium contamination is found on the inside of respirators worn by workers, then bioassay should be initiated immediately. Procedures for monitoring plutonium contamination on protective clothing and the inside of respirators should be documented.

Personnel Monitoring and Bioassay Measurements

The prospective bioassay program should attempt to meet the previously discussed performance capabilities for workplace monitoring programs. Bioassay measurements may be classified as "direct" in vivo counting or "indirect" excreta analysis. The prospective bioassay program should be documented.

In Vivo Counting. Direct bioassay (in vivo counting) is the measurement of radiations emitted from radioactive material taken into and deposited in the body. Lung, whole body, and wound counting are examples of direct bioassay. Direct bioassay is appropriate for detection and measurement of photons emitted by plutonium and its decay products.

Some low-energy x rays emitted by plutonium decay products are energetic enough to escape the body. When direct bioassay is used, the detection system should be calibrated for the radionuclides to be measured and the appropriate organs. All calibration procedures, calibration records, and quality control data should be maintained. Retrospective lung counting for plutonium is discussed further in the section on lung counting.

In Vitro Analyses. Indirect bioassay is the measurement of radioactivity in biological material removed from the body, such as urine, feces, blood, mucus, and biopsy tissue samples. Radioactive material in bioassay samples is measured by a variety of techniques: liquid scintillation counting, fluorescence measurements, gamma spectrometry, chemical separation followed by electrodeposition, and counting with radiation detectors. Urinalysis and fecal analysis for plutonium are further discussed in ensuing sections.

Bioassay results should be used with biokinetic models to estimate internal dose. Procedures describing details of the bioassay program should be documented. These procedures should include a description of sample collection, analysis, calibration techniques, quality control, biokinetic modeling, and dose calculational methods used.

Selection of Individuals for Prospective Bioassay Programs. All individuals who work with unencapsulated plutonium should participate in the prospective bioassay program. Other workers who have potential for inadvertent intakes by inhalation, ingestion, or entry through the skin should also participate. The criteria for selection of employees for internal dosimetry monitoring should be documented.

Bioassay Measurement Frequency. The prospective bioassay measurement frequency should be based on 1) the potential "missed dose," and 2) the potential risks associated with internal exposure. The missed-dose concept refers to the dose that could be "missed" at a given measurement interval. The pattern of retention of activity in the body, the minimum detectable amount (MDA) for a bioassay measurement technique, and the frequency with which that technique is applied define a quantity of intake that could go undetected by the bioassay program. An intake of such a magnitude would not be detected if it occurred immediately after a bioassay measurement and if it were eliminated from the body at such a rate that nothing was detected during the next scheduled measurement. The dose resulting from such an intake would be the "missed dose" for that particular measurement technique and frequency.

Estimates of missed committed effective dose equivalent for each measurement technique should be documented for each measurement frequency and MDA per sample. Retention functions specific to the various chemical forms and particle size distributions found in the facility should be used. In addition, the estimated annual effective dose equivalent missed from intakes of plutonium that might not be confirmed by retrospective bioassay measurement program should be documented.

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

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

4.6.5 Retrospective Internal Dosimetry Surveillance

Direct (in vivo) and indirect (in vitro) retrospective bioassay capabilities are important for evaluating the worker who has received a potential intake of plutonium. Retrospective bioassay measurements are made to confirm positive measurements and to allow for making estimates of the amount of radioactivity taken into the body or deposited in organs of concern. Because estimates of initial deposition are based on time-dependent retention functions, the dates and times of sample collection are important information.

The retrospective internal dosimetry program should ensure that all combined intakes resulting in an annual effective dose equivalent in excess of 100 mrem from all radioactive materials are quantified and recorded. The dose assessment should be based on the isotopic composition, quantity of plutonium taken into the body, and the chemical and physical properties of the material.

For acute intakes, direct bioassay measurements may be taken before and after the period of rapid clearance of activity. Urine and fecal samples collected after known or suspected inhalation incidents may also be used to estimate the magnitude of the intake. Initial assessment of intakes from contaminated wounds are based primarily on urinalysis data.

Retrospective bioassay measurement programs for workers with known or suspected acute inhalation intakes of plutonium or other alpha-emitting radionuclides should include both urine and fecal sampling. Retrospective bioassay measurements should be initiated for each employee in a contaminated work area when surface contamination is detected by routine surveillance if it is possible that the contamination resulted in a committed effective dose equivalent of 100 mrem or greater. Excreta samples should not be collected where they may be contaminated by external sources of plutonium.

All employees suspected of having received an intake of plutonium should be referred for retrospective bioassay measurements. Because a fraction of an intake by inhalation may be retained in the nasal passages for a few hours after exposure to airborne radioactive materials, any level of contamination on a nasal swab is indicative of an intake that should be followed up by a retrospective bioassay measurement program. Retrospective bioassay should also be initiated if plutonium contamination is found on the worker in the vicinity of the nose or mouth.

If a significant intake is indicated, the worker should not be further exposed to plutonium. This is particularly important because a small or recent new intake may confound the interpretation of bioassay measurements for previous intakes of plutonium. The bioassay measurement program should be capable of meeting the performance criteria for bias, precision, and MDA required by draft ANSI N13.30.[a]

Lung Counting

A plutonium facility should have the capability to detect and assess depositions of plutonium in the lungs of radiation workers. The major objective of lung counting is to provide retrospective measurements of suspected

(a) Health Physics Society Standards Committee. 1987. Performance Criteria for Radiobioassay. Draft ANSI N13.30, Health Physics Society Standards Committee, McLean, Virginia.

intakes triggered by workplace monitoring results. Lung measurements should be made to provide an early estimate of the magnitude of the intake and resulting lung deposition.

Two methods have been used to detect plutonium in the lung: the L x-ray method and the americium-tracer method. The L x-ray method is based on the measurement of L x rays following the decay of plutonium. This method provides a direct measurement of plutonium. The detection capability of the method may be on the order of tens of nanocuries for ^{239}Pu and requires an accurate measurement of the chest wall thickness (because of the large attenuation of the low-energy x rays by the rib cage and overlying tissues). Other problems that complicate the measurement of L x rays are 1) the difference in attenuation in muscle and fat, 2) the possibility of nonuniform distribution of the plutonium in the lung, and 3) interferences from radionuclides in other organs or from other radionuclides in the lung.

The americium-tracer method has the advantage of better detection capability for some mixtures of plutonium. The typical MDA for ^{241}Am lung counting is 0.1 nCi. The americium-tracer method depends on the plutonium/americium ratio, which must be independently determined or estimated for each intake. The detection level for this method with a plutonium/americium ratio of 15 is typically 2 nCi plutonium in the lung. The americium-tracer method also has the advantage of being less affected by attenuation in the chest wall or by variations in the muscle/fat ratio. It has the disadvantage of requiring an estimate of the plutonium/americium ratio, both initially and at long times post intake. This ratio may change over time due to ingrowth of ^{241}Am as the decay product of ^{241}Pu, or because americium may naturally clear from the lungs and translocate among internal organs at a rate different than that for plutonium.

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

Urinalysis

Urine sampling provides useful information about the amount of plutonium excreted following an intake. After chemical isolation, the plutonium in urine samples may be determined by alpha spectrometry (gas-flow proportional or surface-barrier detection), alpha counting (zinc sulfide or liquid scintillation counting), or track counting. Analytical procedures for in vitro measurement of plutonium and other radionuclides have been published (Volchok and dePlanque 1983; Gautier 1983).

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

Fecal Analysis

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

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

Fecal sampling is primarily a retrospective monitoring procedure for confirming and evaluating suspected intakes, but is used at some plutonium facilities for prospective monitoring as well. Workers may find fecal sampling unpleasant or objectionable, and laboratory technicians may also have aversion to fecal sample analysis. Some of these problems may be minimized if commercial fecal sample collection kits are used for convenient collection and handling of samples (Fisher et al. 1982). Collection kits also provide a means for collecting uncontaminated samples. Fecal samples may require additional sample preparation prior to analysis.

Wound Counting

Measurement equipment to detect and measure plutonium contamination in wounds should be available at all plutonium facilities. Instrumentation used for this purpose may include thin-crystal NaI(Tl), intrinsic germanium, or Si(Li) detectors. The detection level for plutonium wound measurements is typically 0.1 nCi for ^{239}Pu. Correction for depth due to absorption of photons in the overlying tissues should be considered. Collimated detectors are useful for determining the location of the plutonium in wounds.

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

4.6.6 Indicators of Intake

The health physicist must make important decisions for prompt action at the site of an accidental or suspected intake of plutonium or other radioactive materials. Often these decisions must be based on limited data. Information that may be available for initially estimating the amount and type of intake may include the following:

- levels of measured contamination in the work area
- skin contamination levels, the affected skin areas, and whether or not the skin is damaged or punctured
- wound contamination levels
- chemical form of the material involved
- results of air monitoring
- nasal smear activity levels
- sputum and/or mouth contamination.

The retrospective monitoring program is initiated following a known or suspected intake. This information is needed for dose assessment and future exposure management. The intake is confirmed if follow-up bioassay measurements indicate positive measurement results. Additional bioassay measurements may be needed to quantify the intake and provide data for determining the effective dose equivalent. The frequency of bioassay monitoring will depend on the specific case to be evaluated. Selection of the appropriate sampling frequency is based on the previously discussed performance capabilities for workplace monitoring program, consultations with internal dosimetry specialists, and the cooperation of the affected employee.

4.6.7 Estimating Intake from Bioassay Measurements

Biokinetic models may be used with bioassay data to evaluate the intake, uptake, and retention of plutonium in the organs and tissues of the body. Intake assessments may then be used to calculate committed and annual effective dose equivalent.

Methods for Estimating Intake

There are several published methods for estimating intake from bioassay data (Skrable et al.[a]; Skrable 1983; King 1986; Johnson and Carver 1981). These methods each employ an idealized mathematical model of the human body showing how materials are retained in and excreted from the body over time following the intake. An intake retention model is a simplified mathematical description of the complex biokinetics of a radioactive material in the human body. These models are used to predict the fraction of an intake that will be detected in whole body or chest measurements and in urine or fecal samples. Intake retention models consist of an uptake retention model that relates uptake to bioassay data, and a feed model that relates intake to uptake and bioassay data. Results predicted by the model may then be compared with the observed bioassay data. The model may be iteratively adjusted until the best fit between theory and the observed data is achieved.

(a) Skrable, K. W., G. E. Chabot, C. S. French, and T. R. LaBone. 1986. "Intake Retention Functions and Their Applications to Bioassay and the Estimation of Internal Radiation Doses." Health Physics (submitted for publication in 1986).

Ideally, one should obtain as much bioassay information as possible to determine the intake and track the retention of plutonium in the body to reduce the uncertainty associated with the daily variation in the measurements. A regression analysis should be used to fit the measurement values for estimating the initial intake and clearance half-times.

4.6.8 Estimating Effective Dose Equivalent from Intakes of Plutonium

Plutonium in the body was previously reported in terms of the maximum permissible body burden (MPBB) or maximum permissible organ burden (MPOB). Maximum permissible concentrations were derived for air and water for each radionuclide. Annual limits on intakes of plutonium by workers are now based on the committed effective dose equivalent (ICRP 26 and 30, 1977 and 1979). The committed effective dose equivalent is assumed to be proportional to the risk associated with intake of radioactive materials (ICRP 1979).

The committed effective dose equivalent resulting from an intake of plutonium may be estimated from tabulated data in the Supplement to Part 1 of ICRP 30 (ICRP 1979), or calculated directly using computer programs. The newer ICRP 48 (1986) model parameters may be substituted for ICRP 30 assumptions in the codes.

Annual effective dose equivalent and committed effective dose equivalent values from an acute intake of various plutonium isotopes are given in Table 4.1. The first-year, the second-year annual effective dose equivalent, and the 50-year committed effective dose equivalent that will be received by Reference Man from an acute inhalation of class W or class Y plutonium, and for plutonium directly entering the blood stream by injection are shown. Table 4.1 may be used to estimate the first-year effective dose equivalent that could be received following a postulated intake. These numbers agree reasonably well with estimates by other investigators such as Dunning (1985).

Table 4.2 shows the acute intake expected to result in a first-year or a 50-year committed effective dose equivalent of 100 mrem.

TABLE 4.1. Effective Dose Equivalent Expected Following an Acute Intake of Various Plutonium Radionuclides[(a)]

	REM per Nanocurie Intake								
	Inhalation Class W			Inhalation Class Y			Injection		
Nuclide	1st yr	2nd yr	H(E,50)	1st yr	2nd yr	H(E,50)	1st yr	2nd yr	H(E,50)
^{238}Pu	0.0200	0.0130	0.400	0.0323	0.0232	0.292	0.111	0.108	3.26
^{239}Pu	0.0189	0.0124	0.440	0.0305	0.0221	0.314	0.104	0.102	3.61
^{240}Pu	0.0189	0.0124	0.440	0.0305	0.0221	0.313	0.104	0.102	3.61
^{241}Pu	1.40E-5	3.12E-5	0.00849	2.62E-5	5.48E-5	0.00508	9.49E-5	2.59E-4	0.0706
^{241}Am	0.0202	0.0132	0.455	0.0324	0.0234	0.327	0.111	0.109	3.70

(a) Dose equivalent values calculated using the computer code GENMOD (Johnson and Carver 1981) based on the ICRP Respiratory Tract Model (ICRP 1979) and ICRP recommendations for fractional deposition and retention in organs (ICRP 1986).

TABLE 4.2. Acute Intake Expected to Result in a First-Year or Committed Effective Dose Equivalent of 100 mrem[(a)]

	Intake in Nanocuries					
	Inhalation Class W		Inhalation Class Y		Injection	
Nuclide	H(E,1)	H(E,50)	H(E,1)	H(E,50)	H(E,1)	H(E,50)
^{238}Pu	4.99	0.250	3.10	0.342	0.904	0.0307
^{239}Pu	5.29	0.227	3.28	0.319	0.958	0.0277
^{240}Pu	5.28	0.227	3.28	0.319	0.958	0.0277
^{241}Pu	7130	11.8	3810	19.7	1050	1.42
^{241}Am	4.96	0.220	3.08	0.306	0.901	0.0270

(a) Dose equivalent values calculated using the computer code GENMOD (Johnson and Carver 1981) based on the ICRP Respiratory Tract Model (ICRP 1979) and ICRP recommendations for fractional deposition and retention in organs (ICRP 1986).

4.7 MANAGEMENT OF INTERNAL CONTAMINATION

Experience has shown that most internal exposures to plutonium are accidental. Plutonium facilities and operating procedures are designed to prevent intakes. Nonetheless, it is important for management to prepare for the possibility that workers might receive an intake of plutonium--even though the probability of an incident may be very small. Prompt and appropriate action following an accidental intake of plutonium will allow for therapeutic measures to be taken to minimize the internal contamination and lessen the potential for harmful effects. The health physicist and medical staff should work closely together to ensure that the proper course of action is followed.

The management at the plutonium facility should be prepared to follow an emergency action plan for response to a plutonium intake. If a worker accidentally inhales or ingests plutonium or is injured by a plutonium-contaminated object, the action plan should be initiated immediately. A rapid response is important because any delay in implementing appropriate action could lessen the effectiveness of decorporation therapy and increase the probability for internalized plutonium to deposit on bone surfaces.

4.7.1 Medical Response Plan

The health physicist and medical staff must establish an emergency action plan for the appropriate management of an accidental intake of plutonium. The elements of the plan should include the following:

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

The sequence and priority of the emergency action plan may vary with the magnitude and type of accidental conditions and their severity. An initial early assessment of the incident should focus first on treatment of life-threatening physical injuries and second on the radioactive contamination involved. Minor injuries should be treated after decontamination.

A rapid estimate of the amount of internal contamination by plutonium or other alpha-emitters may not be possible. If a significant intake (meaning one that exceeds ten times the ALI) is suspected, medical staff should proceed with decorporation therapy after first treating major injuries.

Decision Level for Initiating Medical Response

The medical response plan should be activated if 1) a worker requires immediate medical attention, or 2) a worker is contaminated and there is potential that an intake has occurred that is at least ten times the ALI for the radionuclides involved.

Monitoring results may provide an early indication that an immediate medical response is necessary. Following an incident, the health physicist should immediately perform nasal smears, facial contamination surveys, and surveys of the worker's respirator and protective clothing. An example of action levels that might be established as a guide to the health physicist for notifying medical staff is shown in Table 4.3.

Decorporation therapy should be administered immediately following any suspected intake or accidental internal contamination in excess of established action levels. The extent and magnitude of an internal plutonium contamination usually cannot be determined quickly; however, the usefulness of therapy will diminish if plutonium is allowed to translocate to bone where DTPA is ineffective.

TABLE 4.3. Example Action Levels for Prompt Notification of Medical Staff

Measurement	Action Level
Nasal Smear	4,000 dpm
Facial Contamination	250,000 dpm
Chest Count	any positive result
Wound Contamination	10,000 dpm (about 4 nCi)

There is currently no consensus among health physics professionals as to acceptance of decision levels for initiating decorporation therapy with chelating agents. Table 4.2 shows that an intake of about 5 nCi of Class W ^{239}Pu may result in a first-year effective dose equivalent of 100 mrem, and therefore 50 times this amount (250 nCi) could result in a first-year effective dose equivalent of about 5 rem. The probability of harmful effects increases with increasing dose equivalent. Early decorporation therapy will significantly decrease the potential long-term radiation dose.

Decision levels are not based on internal dose because there will not be sufficient time immediately following an accident to determine the effective dose equivalent. Dose evaluation will require retrospective bioassay measurements, obtained periodically after the accident, and careful study of the details of the exposure incident.

Responsibilities for Management of Internal Contamination

Responsibilities should be assigned for action in response to an accidental internal plutonium contamination. The affected worker has the responsibility to inform the health physicist, RPT, or his immediate supervisor as soon as an intake is suspected. The health physicist or RPT should make an initial survey of the extent of the contamination and immediately contact his supervisor and, when action levels are exceeded, contact a member of the medical staff. He should continue to provide monitoring and radiation safety support to the medical staff and supervisors during the management of the contamination incident. Care should be taken to limit the spread of radioactive contamination.

The health physicist should immediately begin to gather data on the time and extent of the incident. Contamination survey results should be recorded. Radionuclide identity, chemical form, and solubility classification should be determined. Nasal smears should be obtained immediately if an intake by inhalation is suspected. When action levels are exceeded, all urine and feces should be collected and labeled for analysis. Personnel dosimeters should be sent to the laboratory for processing. Decontamination should proceed with the assistance of the medical staff. Contaminated clothing and other objects should be saved for later analysis.

Immediate Medical Care

The medical staff should provide immediate emergency medical care for serious injuries to preserve the life and well-being of the affected worker. Minor injuries may await medical treatment until after an initial radiation survey is completed and the spread of contamination is controlled. However, the individual should be removed from the contaminated radiation area as soon as possible. Chemical contamination and acids should be washed immediately from the skin to prevent serious burns and reactions.

A chelating agent should be administered immediately following an accidental intake of plutonium. Both the zinc or calcium salts of DTPA are approved for human use and are available under Investigational New Drug (IND) permits for treating internal plutonium contamination. The worker to be treated must first be informed of the proposed use of an experimental drug, instructed on the purpose of administering the chelating agent, and warned about the possible side-effects of the drug. The worker must then give signed consent before DTPA chelation therapy may be initiated. Even though DTPA therapy is the only method available for reducing the quantity of plutonium or americium retained in the body, the affected worker has the right to refuse its use.

The recommended therapy for decorporation is 1 g Ca-DTPA or Zn-DTPA in 250 ml normal saline or 5% glucose in water, infused intravenously over one hour (NCRP 1980). Treatment may be repeated on five successive days per week, and longer, if retrospective bioassay indicates that decorporation therapy continues to enhance the urinary excretion of plutonium.

Ca-DTPA should not be administered to potentially fertile female workers. Instead, Zn-DTPA should be used for internal decorporation of plutonium and other transuranic materials.

External Decontamination

All transferrable contamination should be removed if possible by cleansing contaminated skin and showering. Wounds should be decontaminated and treated by the medical staff.

Personnel Decontamination Kit. A personnel decontamination kit should be stored in a suitable location near the work area. The kit should be labeled, fully supplied with items such as those shown in Table 4.4, and sealed to prevent others from taking supplies for non-emergency use. Its contents should be inventoried on a periodic basis, and any missing items should be replaced.

Chemical reagents used for skin decontamination may be prepared as follows:

- Na_4-EDTA--A stock solution may be prepared in advance and kept available in the kit. Dissolve 16 g of 70% active Na4-EDTA chemical salt in 100 ml of distilled water for a solution of 10% active ingredient (EDTA [ethylenediaminetetraacetic acid]).
- $KMnO_4$--A saturated solution of KMnO4 is prepared by dissolving crystals in water. The solution is usable as long as the crystals remain dissolved. KMnO4 may be abrasive if used in crystaline form.
- NaHSO3--Sodium bisulfite should be prepared in a 4% solution just prior to its use by dissolving 4 g in 100 ml of tap water. Sodium bisulfite should be stored as a powder and prepared as a solution just prior to its use.
- Household bleach--bottle strength, such as commercial Chlorox bleach will suffice for this purpose.

TABLE 4.4. Typical Items in a Personnel Decontamination Kit

Abrasive soap	Nasal swabs (1x2-in. of 2.5x5-cm filter paper strips)
Cotton-tipped applicators	Paper cups (1 and 4 oz)
Cleansing tissue	Potassium permanganate ($KMnO_4$)
Cotton balls or rolls	Scissors
Flushing tube	Sodium bisulfite ($NaHSO_3$)
Gauze sponges	Surgical gloves
Hand brush	Tongue depressors
Hand cream	Tourniquet kit
Household bleach	Water
Masking tape	Waxes bags (for waste disposal)
Mild detergent containing lanolin and antiseptic	
Na_4-EDTA solution	

Monitoring instruments to detect and locate skin contamination should be available near the personnel decontamination station.

A decontamination sink connected to a contaminated liquid waste system, and containers for sealing solid or moist waste materials, such as cotton swabs, should be available for personnel decontamination.

Procedures for Skin Decontamination. Skin decontamination should be performed by the health physicist. The treatment and decontamination of wounds should be performed by medical staff.

Non-abrasive methods should be used for skin decontamination to protect the tissues from deeper contamination. Masking tape should be used to remove dry contamination. Wet decontamination should be used to remove residual contamination. The skin should be gently scrubbed with soap and water. Household bleach may be applied for additional decontamination effectiveness. The following procedures are recommended:

1. Monitor to determine the contaminated areas of the skin. Have the medical staff treat and decontaminate breaks in the skin.
2. Wipe loose contamination with a gauze sponge or cotton applicators dipped in mild antiseptic detergent. Do not spread contamination to uncontaminated areas.
3. Rub the skin with the applicators to produce good sudsing.
4. Dry the skin area with cleansing tissue. After the skin is thoroughly dry, survey it for any remaining contamination.
5. If no contamination is detected, apply a quality hand cream to prevent chapping.

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

If contamination persists on the skin, a more abrasive decontamination method may be necessary. The decision to proceed with a more abrasive decontamination method should be based on decontamination effectiveness. The decontamination factor is the ratio of the initial contamination level to the contamination level after decontamination methods are applied, as determined by survey instrument readings. Non-abrasive methods should be repeated until the decontamination factor between washes drops below 2 or 3 with significant contamination still remaining. At that point, a more abrasive decontamination method should be considered. An abrasive soap should be applied with a moist gauze sponge or soft handbrush while rubbing the skin to develop a soapy lather. Care should be exercised to prevent damage to the skin surface. If contamination persists after using using the abrasive soap, potassium permanganate and sodium bisulfite should be used. One should paint the contaminated skin with $KMnO_4$ using cotton-tipped applicators, allow the solution to dry, and paint it again two or three more times allowing the solution to dry thoroughly between each application. The skin will then appear almost black. Applicators should be discarded after each use to avoid spreading contamination to the solutions. One should then rub the treated area with $NaHSO_3$, using cotton applicators, until the brown discoloration is removed. Rinse the skin with water to remove the remaining $NaHSO_3$, and dry the area thoroughly and survey it for contamination.

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

4.8 EXTERNAL EXPOSURE CONTROL

Measuring the external exposure and resultant dose equivalent for personnel involved with plutonium production is one of the most difficult dosimetry problems because of the types of radiation encountered. An examination of the radioactive decay schemes in Table 1.1 for the various plutonium isotopes shows the wide distribution of photon energies that require measurement. This energy distribution and the varying energy response of most dosimeters make the interpretation of dosimeter results extremely dependent on proper calibration of the dosimeter and on the design of the dosimeter itself. The presence of neutrons from either spontaneous fission or (α,n) reactions is a further complication. The type of shielding material and its thickness are also critical because a good gamma shield is usually a poor neutron shield and vice versa. The neutron shield thickness for glove boxes is limited to about 6 in. (15 cm) due to the physical limitations of most workers.

This section provides guidance to assist the practicing health physicist in controlling external exposure at a reasonable level.

4.8.1 Dose Rates

Photon dose rates, neutron dose equivalent rates, and shielding materials are discussed in the following subsections.

Photon Dose Rates

One of the problems in handling plutonium oxide or plutonium metal in a glove box is the contact handling of the bare material using a glove. Because of the intensity (yields) of low-energy x rays, the photon dose rates through gloves can be very high. Equation (4.1) has been shown to be useful for predicting the surface dose rate of plutonium through a 100-mg/cm^2 shield[(a)]:

$$D_s(\text{rad/h}) = 171f_{238} + 0.51f_{239} + 2.4f_{240} + 8.7f_{241} + 0.15f_{242} + (0.074f_{241})t \qquad (4.1)$$

where D_s is the surface dose rate of plutonium dioxide or plutonium metal (rad/h), f_i is the weight fraction of the i^{th} isotope of plutonium, and t is the time since chemical separation (days). This equation is valid over a period from three months to five years after separation of the plutonium.

Equation (4.1) is valid for sources that are "infinitely thick" (about 1/4 in. [0.63 cm] for plutonium oxide powder, or 0.040 in. [0.1 cm] for plutonium metal). Note that a term in this equation involves time. This term accounts for the buildup of ^{241}Am, which is produced from the decay of ^{241}Pu. Americium-241 shall be accounted for because of its yield of 60-keV photons and x rays, which readily penetrate some glove box gloves.

If the plutonium oxide is mixed with uranium oxide, the gamma dose rate is reduced by self-shielding. Measurements made with 12% and 19% ^{240}Pu in

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(a) From Faust, L. G., and G. W. Endres. Summary Report on Plutonium Dosimetry. BNWL-1887, Pacific Northwest Laboratory, Richland, Washington (unpublished).

mixed oxide samples from 2 to 50% plutonium oxide in mixed oxide indicate a linear relationship between gamma surface dose rates and plutonium oxide concentration, as indicated by the following equation:

$$D_{MO_2} = D_{PuO_2}\, x + 0.1 \text{ rad/h} \tag{4.2}$$

where D_{MO_2} is the surface dose rate from mixed oxide (rad/h), D_{PuO_2} is the surface dose rate from plutonium oxide (rad/h), and x is the weight fraction of plutonium oxide in mixed oxide. Equation (4.2) has been verified for both new plutonium oxide and old plutonium oxide samples in which a large fraction of the photon dose is due to ^{241}Am.

An equation similar to Equation (4.1) has been derived for use with a lead-loaded rubber glove having the attenuation properties calculated by the computer code PUSHLD (Strode and VanTuyl 1970). The lead-loaded neoprene glove is nominally 0.076-cm thick and contains 0.01 cm of lead equivalent. The surface dose rate, D_{PbGl}, is given by:

$$\begin{aligned} D_{PbGl}(t) = {} & 2.83 f_{238} e^{-0.00789t} + 0.104 f_{239} + 0.0315 f_{240} \\ & + 6.35 \times 10^{-5} f_{242} + f_{241}\,(158.5e^{-0.0016t} - 152.5e^{-0.0457t}) \end{aligned} \tag{4.3}$$

where $D_{PbGl}(t)$ is the surface dose rate (rad/h), f is the weight fraction of the particular plutonium isotope, and t is the time since chemical separation of the plutonium from americium (years). Equation (4.3) includes only the radiation from plutonium, and the ^{237}U and ^{241}Am decay products from the decay of ^{241}Pu. The expression is valid for times of between 50 days and 5 years after the chemical separation of the plutonium. The equation gives results that are within +0% and -20% of PUSHLD calculations.

Neutron Dose Equivalent Rates

The neutron dose equivalent is important in any glove box operation that involves kilogram quantities of plutonium. Neutrons originate from three sources: 1) spontaneous fission, 2) (α,n) reactions with ^{17}O and ^{18}O and low atomic number impurities, and 3) neutron-induced fission. Experience has shown very little neutron-induced fission from loose plutonium oxide powder; induced fission seems to be significant only with plutonium metal or very large arrays of plutonium oxide.

Most neutrons from spontaneous fission originate from the isotopes ^{238}Pu, ^{240}Pu, and ^{242}Pu; in plutonium oxide, neutrons result from $^{17}O(\alpha,n)$ and $^{18}O(\alpha,n)$ reactions. Table 4.5 gives the theoretical yields from various isotopes and compounds. The approximate neutron yield from a plutonium compound (Y in neutrons per second) can be established by summing the products of the weight fraction of the isotope (W_i), multiplying the sum by the spontaneous fission (Q_{si}) and (α,n) yields ($Q_{\alpha i}$) (as shown in Equation [4.4]) for that particular isotope as listed in Table 4.5.

$$Y = \sum_i W_i \left(Q_{si} + Q_{\alpha i}\right) \tag{4.4}$$

In addition, a small contribution of neutrons usually comes from (α,n) reactions in low atomic number impurities found in the plutonium or plutonium compound. The neutron yield from impurities (Y_{imp} in neutrons per second) can be determined from the data in Table 4.6 and Equation (4.5):

$$Y_{imp} = A_\alpha \sum_j P_j I_j \tag{4.5}$$

where A_α is $\sum_i A_i$, which is the total alpha activity from all alpha-emitters, including ^{241}Am ingrowth from ^{241}Pu; P_j is the neutron yield from j^{th} low Z element (neutrons/α-ppm); and I_j is the impurity concentration in plutonium (ppm).

Using a precision long counter, measurements were made of the neutron yields of more than 100 samples of plutonium and plutonium compounds. The plutonium compounds had different isotopic compositions ranging from 6% ^{240}Pu to 23% ^{240}Pu. In these measurements, nuclear fuel grade plutonium metal usually had a neutron yield at least 20% greater than the calculated value, probably because of light element impurities in the parts-per-million range

TABLE 4.5. Neutron Yield from Plutonium and Uranium Compounds

Nuclide	Half-life Years	Alpha Activity, α/sec-g of Isotope	Q_s (a) Spontaneous Fission Yield, n/sec-g of Isotope	$Q_s + Q_\alpha$ (b) Oxide Yield, n/sec-g of Compound	$Q_s + Q_\alpha$ (b) Fluoride Yield, n/sec-g of Compound
^{236}Pu	2.85	1.97×10^{13}	3.7×10^{4}	4.63×10^{5}	6.5×10^{7}
^{238}Pu	8.78×10^{1}	6.34×10^{11}	2.62×10^{3}	1.63×10^{4}	2.1×10^{6}
^{239}Pu	2.44×10^{4}	2.27×10^{9}	3×10^{-2}	4.5×10^{1}	4.3×10^{3}
^{240}Pu	6.54×10^{3}	8.44×10^{9}	1.02×10^{3}	1.07×10^{3}	1.7×10^{4}
^{241}Pu (c)	1.5×10^{1}	9.01×10^{7}	---	1.8	1.70×10^{2}
^{242}Pu	3.87×10^{5}	1.42×10^{8}	1.7×10^{3}	1.5×10^{3}	1.67×10^{3}
^{241}Am (d)	4.33×10^{2}	1.27×10^{11}	1.6	3.1×10^{3}	4.0×10^{5}
^{232}U	7.2×10^{1}	7.95×10^{11}	0.3 (e)	1×10^{4} (e)	3×10^{6} (f)
^{233}U	7.04×10^{8}	8.1×10^{4}	---	6.8	4×10^{3} (e)
^{234}U	2.47×10^{5}	2.29×10^{8}	5.7×10^{-3}	4.7	5.76×10^{2} (g)
^{235}U	7.1×10^{8}	7.93×10^{4}	6.0×10^{-4}	2×10^{-3}	0.122 (g)
^{236}U	4.47×10^{9}	1.23×10^{4}	4.3×10^{-3} (e)	---	3.95 (g)
^{238}U	4.51×10^{9}	1.23×10^{4}	1.12×10^{-2}	1.9×10^{-4}	2.79×10^{-2} (g)

(a) Neutrons/sec-g of metal.
(b) Neutrons/sec-g of compound including spontaneous fission and (α,n) neutrons.
(c) ^{241}Pu is a beta emitter with an alpha-branching ratio of 2.46×10^{-5}. The (α,n) yields were calculated from the ratio of alpha activities and yield of ^{240}Pu, which has a similar alpha energy.
(d) The amount of ^{241}Am can be calculated from $f_{Am} = 1.034\, f_{Pu}\, (e^{-0.0016t} - e^{-0.0481t}$, where f_{Am} is the weight fraction of ^{241}Am at time t in years and f_{Pu} is the weight fraction of ^{241}Pu at the time of separation.
(e) Estimated.
(f) Estimated for ^{232}U alone; increase by a factor of 5 if in equilibrium with its decay products.
(g) From Sampson (1974).

TABLE 4.6. Approximate Neutron Yields of Low Atomic Number Impurities in Plutonium

Impurity Element	(α,n) Neutron Yield, n/α/ppm	$\frac{n}{(sec)\ (Ci\ of\ Pu)\ (ppm\ Impurities)}$ [(a)]
Li	6.3×10^{-12}	2.3×10^{-1}
Be	2.0×10^{-10}	7.4
B	4.6×10^{-11}	1.7
C	2.8×10^{-13}	1.0×10^{-2}
O	1.6×10^{-13}	5.9×10^{-3}
F	2.4×10^{-11}	8.9×10^{-1}
Na	3.0×10^{-12}	1.1×10^{-1}
Mg	2.7×10^{-12}	1.0×10^{-1}
Al	1.5×10^{-12}	5.6×10^{-2}
Si	3.3×10^{-13}	1.2×10^{-2}

(a) Excluding the beta activity of ^{241}Pu.

and, possibly, some neutron multiplication. Nuclear fuel grade plutonium oxide usually had a measured neutron yield of about 10% greater than the calculated value, based on (α,n) reactions.

In plutonium compound mixtures, the neutron yield is about the same as that from the pure plutonium compound. In plutonium oxide-uranium oxide mixtures, the size of the plutonium oxide particles is usually large enough so that α particles from the plutonium lose most of their energy inside the plutonium oxide particle, and therefore the O(α,n) yield is essentially the same as that from pure plutonium oxide.

The neutron dose equivalent from plutonium cannot be neglected. Large masses of plutonium can generally be treated as a point source of neutrons located at the center of the mass if the distance between the center of the mass and the measurement point exceeds the dimensions of the plutonium source array. The neutron dose equivalent rate H (in millirem per hour) at a distance r (in centimeters) from the center of a large mass of plutonium, plutonium oxide, or mixed oxides can be found from Equation (4.6) (Brackenbush, Brown, and Faust 1974):

$$H = 0.0097 \frac{S}{r^2} \qquad (4.6)$$

where S (the total neutron emission rate in neutrons per second) is the product of the mass of the plutonium (g) times Y (the total neutron yield per gram of plutonium in neutrons per second per gram, from spontaneous fission [α,n] reactions and fission-induced neutrons).

4.8.2 Shielding Materials

The types and thicknesses of a shield required to achieve a specific dose rate are a function of the quantity and isotopic composition of the plutonium present, the type of operation, and the chemical and physical properties of the plutonium. The data in the following subsections, with the equations in the previous section, can be used to estimate what exposures can be expected and what type and thickness of shielding will be necessary to achieve a desired dose rate. The results presented are only estimates; if more refined results are desired detailed calculations or measurements should be performed. The buildup of plutonium contamination on glove box surfaces shall be considered. After a few months work in a glove box with quantities of plutonium, the radiation from the decay products of ^{241}Pu, ^{237}U and ^{241}Am, can dominate.

Photon Shields

The attenuation of the photon dose rates of a finite thickness of material is highly dependent on the isotopic composition of the plutonium. Generally, the greater the fraction of ^{239}Pu, the less the attenuation per unit of thickness of shielding material; however, the total photon dose rate is normally lower. Tables 4.7 and 4.8 contain some measured attenuation factors for several materials and two isotopic compositions of plutonium.

Neutron Shields

Calculating the effectiveness of neutron shields requires the use of computer codes such as PUSHLD, ANISN (Engle 1967), and BMC/MG (Zimmerman and Thompson 1975). However, for completeness some data included here give dose reductions as a function of several shielding materials and thicknesses. These data can be used with Equations 4.8 through 4.10 to estimate the neutron dose.

TABLE 4.7. Plutonium Oxide Sources

Shield Material	Thickness, in.	Photon Attenuation Factors[(a)] for: Source A	Source B
None		1.00[(b)]	1.00[(b)]
Stainless Steel (7.9 g/cm^3)	0.005	0.48	0.93
	0.015	0.38	0.77
	0.0295	0.29	0.58
	0.058	0.18	0.38
	0.1305	0.08	0.18
	0.187	0.06	0.13
	0.250	0.05	0.11
	0.532	0.02	0.05
	0.854	0.011	0.03
	1.015	0.007	0.02
Lead (11.4 g/cm^3)	0.002	0.58	
	0.006	0.04	0.22
	0.132	0.014	0.051
	0.198	0.008	0.030
	0.262	0.005	0.021
	0.528	0.0013	0.008
	0.660	0.001	0.005
	0.990	0.0007	0.003
0.1875-in. Steel + Lead of Thickness:	0.066	0.026	0.079
	0.132	0.011	0.040
	0.262	0.004	0.026
Leaded Glass (4.9 g/cm^3)	0.25	0.023	0.171
	0.50	0.009	0.040
	0.75	0.005	0.017
	1.01	0.003	0.010
Safety Glass (wire grid)	0.271	0.35	0.68
	0.542	0.25	0.50
Neoprene Glove (0.015-in. nominal)	1 layer	0.84	0.90
	2 layers	0.72	0.89
	3 layers	0.62	0.88
Lead-Loaded Neoprene (0.030-in. nominal)	1 layer	0.26	0.51
	2 layers	0.14	0.30
Acrylic Plastic	0.059	0.935	0.90
	0.179	0.80	0.89
	0.250	0.75	0.88
	0.375	0.66	0.81
	0.533	0.60	0.80
	0.78	0.52	0.79
	1.07	0.48	0.75

(a) The attenuation factor is the ratio of the shielded dose rate to the unshielded dose rate.

(b) The unshielded dose rates for sources A and B are 0.5 rad/h and 1.5 rad/h, respectively, as measured by an extrapolation chamber with the plutonium dioxide powder encapsulated in 0.03-in.-thick polyvinylchloride plastic bags.

TABLE 4.8. Isotopic Composition of the Plutonium Oxide Sources Used in Table 4.7

	Isotopic Concentration, wt%					
	^{236}Pu	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
Source A (2.9 years since separation)	4.5×10^{-6}	1.92	63.3	19.2	11.7	3.88
Source B (5.0 years since separation)	9.1×10^{-7}	0.399	70.0	21.6	6.31	1.73

The dose reduction data shown in Figures 4.1 through 4.4 are based on measured values using PuF_4, plutonium oxide, and ^{252}Cf sources. Data showing the quality of the "fit" between plutonium oxide and ^{252}Cf are also given so that the latter can be used in place of plutonium oxide for neutron studies.

All neutron dose rate (i.e., kerma rate) measurements were made with a tissue-equivalent proportional counter (TEPC). This counter directly measures the neutron dose to a small volume of tissue equivalent material, and thus eliminates the energy correction problems inherent in other types of neutron dosimeters. The TEPC measurement for these studies does not include any gamma radiation secondaries from (n,γ) reactions in the shielding materials. All measurements were made at a 30-cm centerline-to-centerline separation, with the slab shields placed in between and cylindrical or spherical shields placed over the source. The relative dose reduction is the ratio of measured neutron dose rates with and without the shield.

Figure 4.1 shows the dose reduction for a ^{252}Cf, spontaneous-fission, neutron source (with an average neutron energy of 2.3 MeV) shielded by polyethylene (0.93 gm/cm^3). The circles indicate TEPC measurements made with 1-ft by 2-ft (0.3-m by 0.6-m) slab shields. The triangles denote TEPC measurements with spherical shields of polyethylene that match the one-dimensional ANISN computer code geometry. The solid dots indicate the relative reduction in kerma, calculated from fluence values using the computer code ANISN and the fluence-to-kerma factors of Ritts, Solomito, and Stevens (1969). ANISN is a

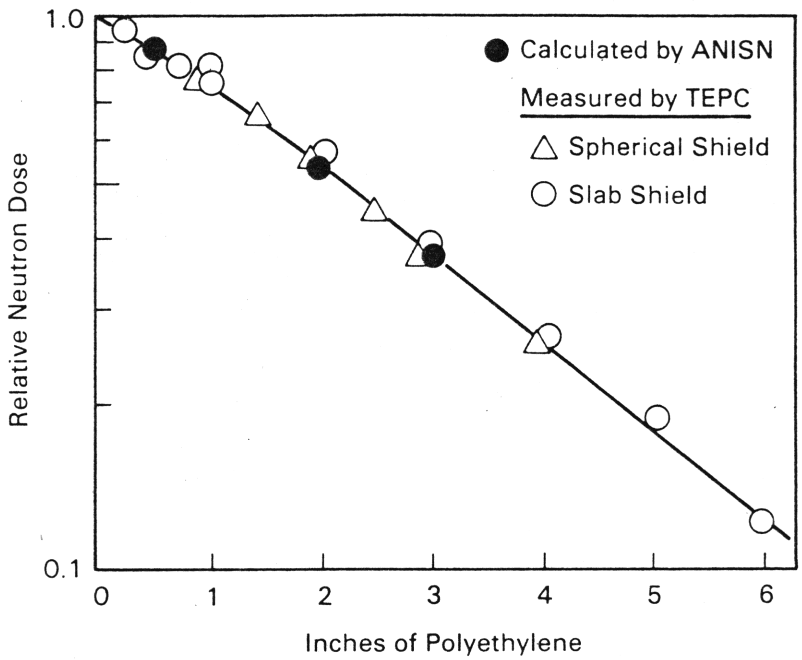

FIGURE 4.1. Shielding Effectiveness of Polyethylene Shields for a ^{252}Cf Fission Neutron Source

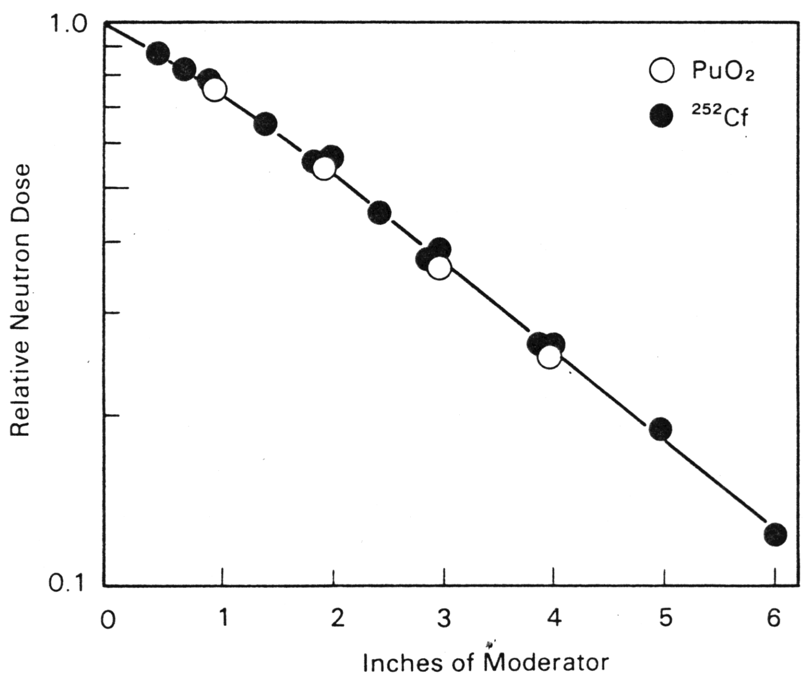

FIGURE 4.2. Shielding Effectiveness of Polyethylene Slab Shields for a ^{252}Cf Fission Neutron Source and High-Exposure Plutonium Oxide Neutron Source

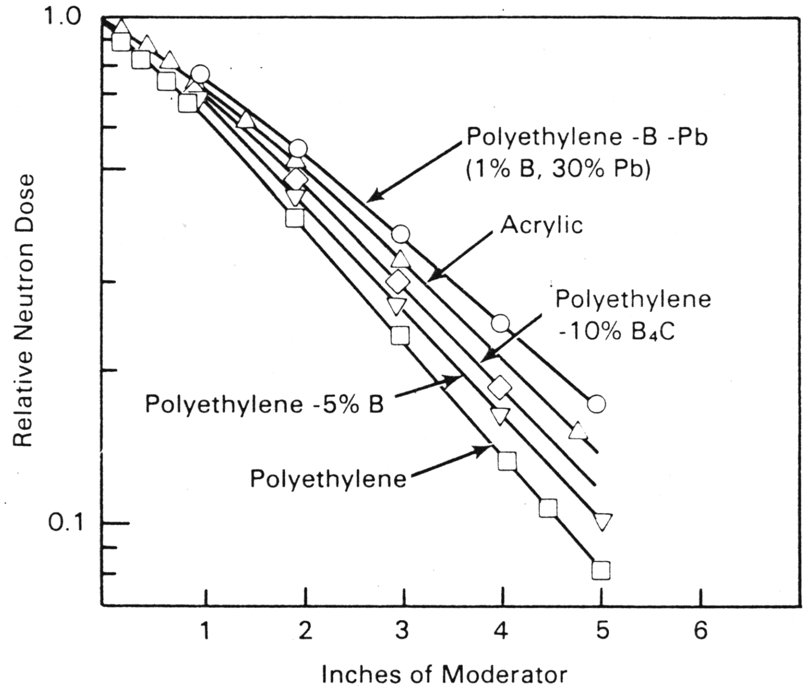

FIGURE 4.3. Shielding Effectiveness of Various Slab Shields for a PuF_4 Neutron Source

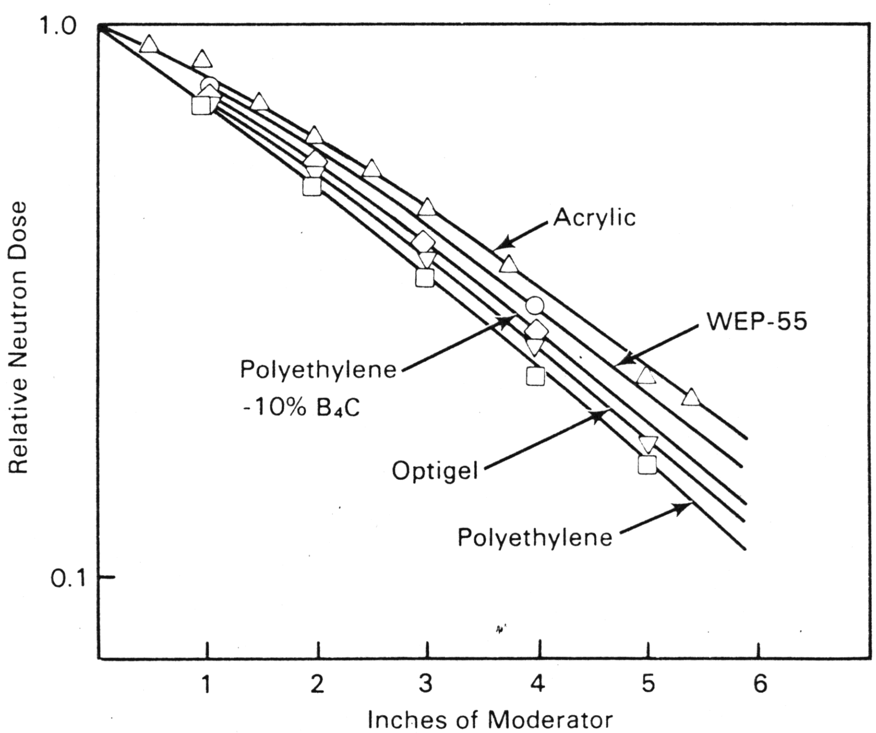

FIGURE 4.4. Shielding Effectiveness of Various Slab Shields for a ^{252}Cf Fission Neutron Source

one-dimensional, discrete ordinate transport code that yields results that are in good agreement with the values measured by the TEPC.

Figure 4.2 presents the results of neutron shielding measurements using polyethylene shields with a ^{252}Cf spontaneous-fission source (with an average neutron energy of 2.3 MeV) and a high-exposure, plutonium oxide (>20% ^{240}Pu) neutron source. The average energy of the PuO_2 source was calculated to be about 2.1 MeV, assuming that 68% of the neutrons came from spontaneous fission with an average energy of about 2 MeV and 32% of the neutrons came from O(α,n) reactions with an average energy of about 2.3 MeV. These results indicate that plutonium oxide can be simulated by ^{252}Cf sources.

Figure 4.3 shows a set of shield-effectiveness measurements using a PuF_4 neutron source with an average neutron energy of 1.3 MeV, and 1-ft by 2-ft (30-cm by 60-cm) slab shields. The slab shields used in this experiment were regular polyethylene, acrylic plastic, and three types of polyethylene loaded with boron. The boron-loaded polyethylenes were polyethylene with 5 wt% boron in the form of borates, polyethylene with 10 wt% boron carbide, and polyethylene with 1 wt% borate and 30 wt% lead for gamma shielding. The data points in Figure 4.4 give an idea of the precision that can be achieved with the TEPC. In this experiment, polyethylene was the most effective shield material. In general, the results verify the maxim that the best fast-neutron shield is the one with the greatest hydrogen density. The addition of boron or other thermal neutron-absorbing materials seems to have little effect in shielding fast neutrons and may, in fact, reduce the effectiveness of the shields by reducing the hydrogen density.

Figure 4.4 shows the results of the shielding measurement using ^{252}Cf to simulate plutonium oxide, and 2-ft by 2-ft (60-cm by 60-cm) slab shields of various materials. The shielding materials used were regular polyethylene, acrylic plastic, Optigel® (a proprietary shield containing 90% water), polyethylene with 10% carbon, and water-extended polyester (WEP) containing 55% water by weight (WEP-55®).

® Optigel and WEP-55 are registered trademarks of Viox Corporation, Seattle, Washington.

Figure 4.5 shows the shielding effectiveness of concrete using a ^{252}Cf source and ANISN calculation. Measurements were made using relatively new concrete (<2 years old) and concrete that was approximately 10 years old.

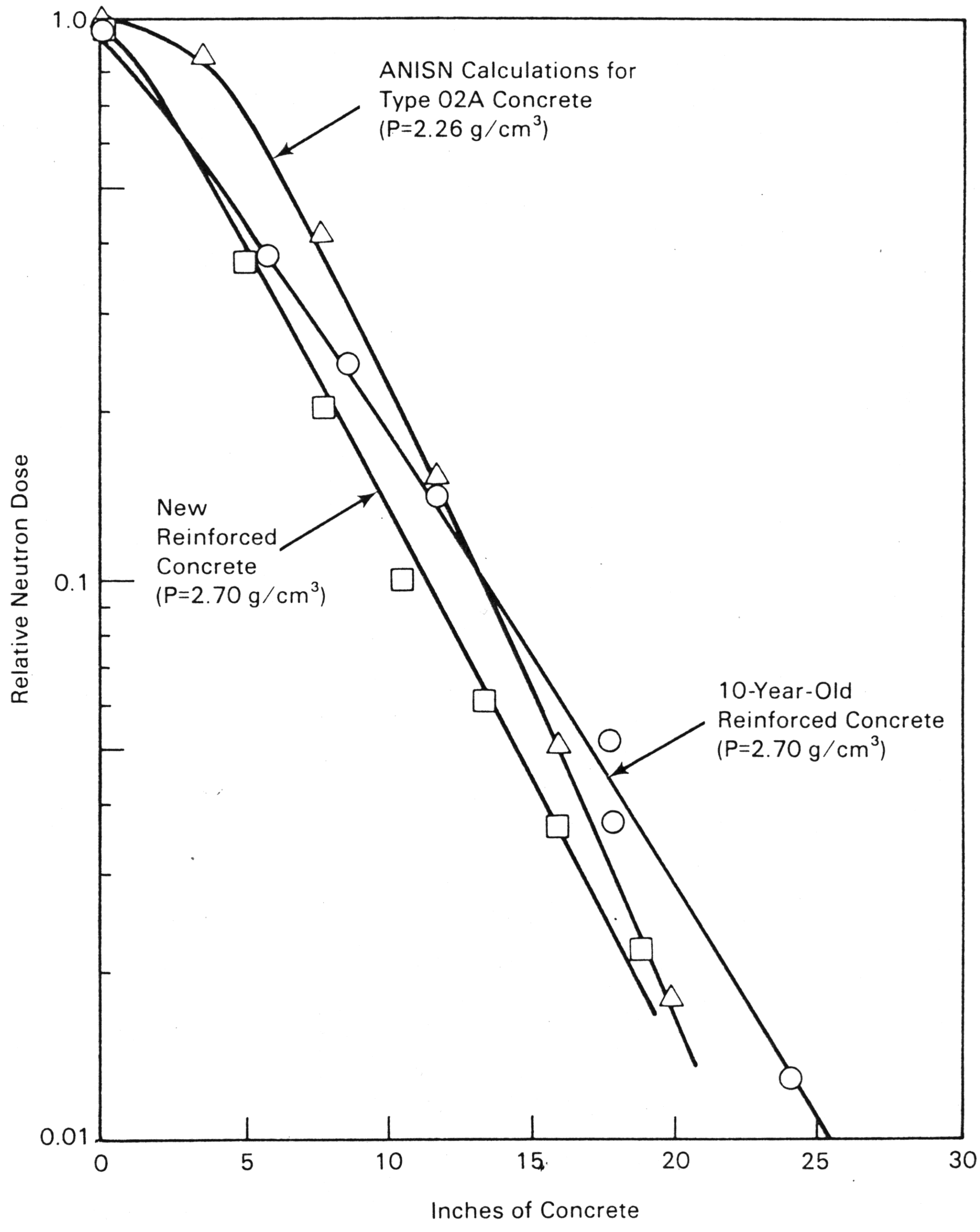

FIGURE 4.5. Shielding Effectiveness of Concrete for ^{252}Cf Fission Neutrons

Both concretes were provided by the same supplier and were of the same composition. However, the attenuation curves in Figure 4.5 show that the newer concrete is a better neutron attenuator than the older concrete due to loss of water content through dehydration. The rate of dehydration can increase if the cement is subjected to moderate-to-high radiation levels over long periods of time, such as in storage vaults.

Because no gamma contributions are included, small differences may be noted when these results are compared with computer calculations. In most computer calculations, a small dose contribution is included from $H(n,\gamma)$ reactions. This difference is most apparent for thick shields with a large number of thermal neutrons, because of the increased cross-section of the $H(n,\gamma)$ reactions.

4.8.3 Example of Calculating the Dose Rate from Handling Plutonium in a Glove Box

The equations and dosimetry data described in the previous sections can be applied to estimate the dose rate to a worker handling plutonium in a glove box. As an example, assume that the worker is handling 1.131 kg of plutonium oxide (1.0 kg of plutonium) with the isotopic compositions specified in Table 4.9 (the expected average compositions of LWRs producing plutonium in 1975 and 1980). The plutonium is contained in a steel can 1.75 in. (4.45 cm) in radius and 3.78-in. (9.59-cm) high with 0.020-in.- (0.051-cm-) thick walls. The plutonium oxide is in the form of a loose powder with a density of 1.9 g/cm^3.

Three locations are of interest: 1) the surface of the can where the worker's hands contact it; 2) approximately 6 in. (15 cm) away from the can where the dose to the forearm is important; and 3) about 14 in. (36 cm) away from the can where the dose to the body is important. Fourteen inches is an average working distance for an operator at the side of glove box with his arms in the gloves. The doses at these points can be compared with the design criteria for DOE plutonium facilities, which are contained in the DOE 6430.1 (DOE 1983). For purposes of designing shielding and equipment layout, DOE's design limit for exposure to the whole body is 1 rem/yr, which is equivalent to an average dose-equivalent rate of 0.5 mrem/h at the work station for continuous exposure. An operational alternative of remote handling,

TABLE 4.9. Isotopic Composition of LWR-Produced Plutonium and Low-Exposure Plutonium

Year	Plutonium Isotope, wt%					
	^{236}Pu	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
1975	5×10^{-6}	0.9	65.5	21.8	9.0	2.7
1980	7.3×10^{-6}	1.5	58.1	24.1	11.4	4.9
1985	6.5×10^{-6}	1.6	54.7	24.7	12.2	6.7
Low exposure	--[a]	0.128	93.7	5.81	0.138	0.030

(a) Below 10^{-7} wt%.

automation, or worker rotation shall be used where exposures to the hands and forearms are likely to approach 10 rem/yr. The limits on dose equivalent for design purposes are as follows:

Whole body	1 rem/yr
Forearm	10 rem/yr
Hands	10 rem/yr

However, a lower design goal should be considered.

Equations (4.5), (4.6), and (4.7) were used to calculate the photon dose rates at the three locations of interest one year after chemical separation of americium from the plutonium. The neutron dose equivalent rates were estimated from Equation (4.10), with neutron yields of 450 n/s-kg for 1975-plutonium oxide, and 602 n/s-kg for 1980 LWR-plutonium oxide at one year after separation of the plutonium. At the surface of the can, the average neutron flux per unit area (source strength divided by surface area) was multiplied by a flux-to-dose-equivalent conversion factor of 0.122 mrem/h per n/cm^2-s to determine the average neutron dose equivalent rate on the surface of the can.

Several shielding materials were considered. In glove box operations with low-exposure plutonium, 0.015-in.- (0.038-cm-) (nominal) thick neoprene gloves are used. For plutonium oxide in a 0.020-in.- (0.051-cm-) thick steel can, the dose rate is almost the same as it is using 0.015-in.- (0.038-cm-)

thick neoprene gloves or 0.030-in- (0.076-cm-) thick polyvinyl chloride (PVC) plastic. Two types of lead-loaded neoprene gloves were also considered: a nominal 30-mil- (0.076-cm-) thick glove containing 4 mil (0.010 cm) of lead equivalent, and a nominal 60-mil (0.152-cm) glove containing 16 mil (0.041 cm) of lead equivalent. The glove box was considered to be 1/4-in- (0.635-cm-) thick stainless steel. The lead shields were considered to be 1/8-, 1/4-, and 1/2-in- (0.32-, 0.64-, and 1.3-cm) thick and the effects of composite steel-lead shields were not included. The neutron shield used for these calculations was Optigel, a commercial shielding material composed of water and an organic compound that form a solid colloid. Optigel has the advantages of being fireproof, transparent, and effective as a neutron shield. However, care must be exercised when using Optigel to ensure the integrity of the colloid container. Air leaking into the container may cause the colloid to dry out and deposit as an opaque layer on the container surfaces. Table 4.10 shows the results of these calculations.

Based on the data in Table 4.10 the following situation could occur. The worker could actually have his hands in contact with the can of plutonium oxide from about 3 to 8 hours before he receives 5 rem to his hands. If 1.31 kg of plutonium oxide are left in a steel glove box without any additional shielding, at about 1 ft (30 cm) from the wall, the worker will receive his design exposure of 1 rem to the whole body in 3 or 4 weeks. If a 1/2-in.-thick lead gamma shield and a 6-in. thick Optigel neutron shield are added, 1975 LWR-produced plutonium can be processed for the entire year and 1985 LWR-produced plutonium can be processed for 42 weeks before the design exposure limit of 1 rem is reached. The worker's activity in or near the glove box is assumed to occur for about 6 h/day, with 2 hours spent away from the work area.

Restricting the amount of plutonium near the operator in the glove box or placing neutron shields in the floor of the glove box generally would be more effective than putting massive 6-in.- (15-cm-) thick neutron shields around the outside of the glove box. Inside the gloves, the gamma dose is more restrictive than the neutron dose. Outside the glove box, the neutron dose is more restrictive, especially if small amounts of shielding are used.

TABLE 4.10. Dose Equivalent Rates from Handling a Can Containing 1.13 kg of Plutonium Oxide (1.0 kg of Plutonium) in a Glove Box

	1975 LWR Plutonium Oxide				1980 LWR Plutonium Oxide			
	Dose Equivalent Rates, mrem/h			Time to Reach Annual Limit(b)	Dose Equivalent Rates, mrem/h			Time to Reach Annual Limit(b)
	Gamma	Neutron	Total		Gamma	Neutron	Total	
Surface of Glove Touching 0.020-in.-Thick Steel Can with Plutonium Oxide								
0.033-in.-thick PVC plastic or neoprene glove	1600	160	1800	17 h	2000	210	2200	13.5 h
0.030-in.-thick lead-loaded glove	980	160	1150	26 h	1200	210	1400	21 h
0.060-in.-thick heavy lead-loaded glove	400	160	560	53 h	490	210	700	42 h
Forearm at 6 Inches from Can Containing Plutonium Oxide								
0.033-in.-thick PVC plastic or neoprene glove	87	13	100	5 wk	110	17	126	4 wk
0.030-in.-thick lead-loaded glove	71	13	83	6 wk	89	17	105	4.8 wk
0.060-in.-thick heavy lead-loaded glove	30	13	43	12 wk	38	17	54	9.3 wk
Outside of Glove Box with Surface of Can 14 Inches from Wall								
1/4-in.-thick steel	5.3	3.1	8.4	4 wk	6.6	4.1	10.7	3.1 wk
1/8-in.-thick lead	1.3	3.1	4.5	7.4 wk	1.6	4.1	5.7	6 wk
1/4-in.-thick lead	0.63	3.1	3.8	9 wk	0.77	4.1	4.8	7 wk
1/2-in.-thick lead	0.22	3.1	3.4	10 wk	0.29	4.1	4.4	7.6 wk
1/2-in.-thick lead + 4 in. of Optigel in lucite box	0.22	0.81	1.0	33 wk	0.29	1.1	1.4	24 wk
1/2-in.-thick lead + 6 in. of Optigel in lucite box	0.22	0.39	0.61	(c) wk	0.29	0.50	0.79	42 wk

(a) The plutonium is one-year old; i.e., one year has passed since isotope separation.
(b) For shielding and equipment layout design purposes only, the upper limit exposures are 1 rem/whole body, 15 rem/yr to forearms, and 30 rem/yr to the hands (Bistline and Tyree 1967). It is also assumed that the worker is actually at the glove box 6 h/day or 30 h/wk (assuming 2 h/day for changing clothes, lunch, coffee, and smoke breaks, etc.).
(c) Below design exposure limits.

4.8.4 Other Control Methods

The example given in Section 4.7.3 illustrates the effects of shielding to achieve exposure control. However, shielding alone may not be adequate and other means of control must be considered, among them physical design, housekeeping, inventory controls, remote handling, occupancy factors, etc. The data in Table 4.19 do not include dust layers on the surface of the gloves and the interior of the glove box, which increase the exposure rate. For instance, exposure rates as high as 30 mR/h inside neoprene gloves and 10 mR/h inside 30-mil-thick, lead-loaded, rubber gloves have been measured inside a glove box that was used to process plutonium powder with 12 wt% ^{240}Pu, 6% ^{241}Pu. Plutonium with 15 to 25% ^{240}Pu would yield even higher exposure rates. The measurements were made with thermoluminescent dosimeters (TLDs) taped onto an arm phantom, and the exposure rates varied by a factor of 3 from point to point depending upon the thickness of the glove and the amount of dust on the glove (the largest amount of dust being on the region by the hand). At 10 mR/h, a person would approach 5 rem to the forearm in about 8 weeks with only the plutonium oxide dust on the leaded rubber glove as the source. This fact illustrates the necessity of cleaning the interior surface of the glove box, including the gloves, at least weekly and minimizing operations that generate dust.

The plutonium should be processed as quickly as possible after it is chemically separated from ^{241}Am. In the case of plutonium with the average isotopic composition expected from LWRs in 1980, the gamma dose rate can increase by a factor of 20 over a 10-year period and by a factor of 10 over a 4-year period from the buildup of decay products, notably ^{237}U and ^{241}Am from the decay of ^{241}Pu. The dose rate increases rapidly from the buildup of ^{237}U in the first 50 days, then increases more gradually from the buildup of ^{241}Am and other decay products. Plutonium that contains more than about 5% ^{241}Pu and is several years old should be reprocessed to remove the ^{241}Am before processing the plutonium.

4.8.5 Neutron Spectrometer

Brackenbush et al. (1988) describe the development and use of a new field neutron spectrometer suitable for making measurements of neutron energy spectra in plutonium facilities. One version of the field spectrometer uses a

^{3}He proportional counter to measure the neutron energy spectrum between 50 keV and 5 MeV and a TEPC to measure absorbed neutron dose and estimate quality factors for neutrons with energies between thermal and 20 MeV. These detectors were selected because they are self-calibrating and do not require exposure to calibrated neutron sources. The spectrometer also contains an analysis module with a computer to control the data acquisition and analysis of data in the field. The spectrometer can measure energy spectra and determine average quality factors and dose equivalent rates.

4.8.6 Dosimetry

An important part of any radiation protection program includes making adequate measurements and maintaining records of radiation exposure. The operation and types of radiation present in a process area are crucial in determining the type of dosimeter needed to ensure adequate measurements, which then become part of the individual's exposure record.

Dosimetry Assignment

According to DOE 5480.11 (DOE 1988), personnel dosimeters are required for all workers who have a potential to exceed an annual effective dose equivalent to the whole body of 100 mrem. The dosimeters need to be routinely calibrated and maintained to meet the requirements of the DOE Laboratory Accreditation Program for personnel dosimetry as required in DOE 5480.15 (DOE 1987b).

Dosimeters capable of measuring beta, gamma, and neutron exposures should be assigned to every individual who routinely works in the process area. The exchange period should be based on facility experience. Normally monthly to quarterly exchange periods should be the preferred frequency. However, a less frequent exchange (e.g., annual) may improve the detection capabilities and reduce the amount of false dose equivalents recorded.

Whole Body Dosimeters

Plutonium facilities shall use a combination thermoluminescent/track-etch neutron dosimeter. Implementation of this combination dosimeter is imminent at DOE facilities where the potential exists for significant neutron exposures to some portion of the workforce. Except for neutron energy ranges experienced at accelerators, the neutron sources to which personnel are exposed typically range in energy from thermal to 5 MeV. Most TLD albedo systems in use at DOE facilities can detect exposures in this range, but their detection capability to neutron energies above 100 keV decreases rapidly. The use of the CR-39 track-etch dosimeter (TED) (responsive to energies from 100 keV to at least 18 MeV) in combination with the TLDs, covers the energy range of most occupational exposures and provides additional protection against missing a dose. Track-etch dosimeters have several advantages over TLDs because they

- are not effected by beta and gamma radiation up to about 1000 rad
- are stable over a long period of time exhibiting no significant fading
- do not require backscatter from the body to detect exposure
- do not over-respond, leading to overestimations of neutron dose equivalents.

The combination TLD/TED also will provide basic energy information that may be useful in assigning dose equivalent.

Calibration of the TED components of the combination TLD/TED shall be based on a technique that is traceable to the NBS and on the response of the dosimeter to the field environment according to the manner in which it is used. Calibration or correction factors shall be determined that relate the response of the dosimeter in the field environment to the response of the NBS-traceable exposure. The response of the TEDs to energies between 150 keV and 3 MeV shall be essentially constant. However, if the facility has numerous exposure sources of differing energies, a variety of energy calibrations should be performed to assist in interpretation of the TEDs. The neutron spectrometer discussed in Section 4.7.5 could be used for supplying spectral data for calibrating the TEDs.

The whole body dosimeter should be placed between the waist and the neck. For albedo types of dosimeters, the dosimeter shall be secured firmly against the torso. Generally radiation workers in process areas or at glove boxes face the radiation source so that the dosimeter should be placed on the front of the torso. If an operator is working in an environment in which the radiation dose is greater to his back than his front a dosimeter should be placed on his back. The RPT needs to adequately survey all of the work areas and study the work procedures to properly determine the correct placement and number of personnel dosimeters required for the operators. The dosimeter that yields the highest valid reading should be used for the dose of record.

Extremity Dosimetry

Extremity dosimeters shall be considered for all personnel who perform operations that involve hand contact with plutonium. If hand contact is only momentary and would result in less than 1 rem annually to the extremities, the need for extremity dosimeters should be evaluated. Extremities are defined as the hands and forearms below the elbows, and the feet and legs below the knees. In a plutonium facility the contact dose to the hands may be the most limiting of the extremity doses. For non-glove box operations, the extremity dose from any type of radiation shall be considered limiting over the whole body dose when the dose gradient is greater than 6:1 over the distance of 1 m (the approximate distance between the tip of the fingers on an outstretched hand and the trunk of the body). In most cases the source is not held at arm's length and the dose gradient needs to approach 10:1 or 20:1 at 1 m for the extremity dose to be limiting (Reece et al. 1983). It should be noted that glove box shielding may reduce the radiation exposure to the whole body but not to the hands. Therefore, the ratio of extremity dose to whole body dose should be measured in the actual field geometry in order to determine the limiting conditions.

Extremity dosimeters are usually only gamma sensitive and are worn on the wrists or fingers. Their exchange period is impacted by the process and requirements or conditions unique to the facility. Some means of routinely estimating the neutron exposure should also be used and included in the record of total extremity exposure. Neutron-to-gamma ratios shall not be used without supporting evidence that such ratios are reasonably constant.

Personnel Nuclear Accident Dosimeters

Personnel nuclear accident dosimeters (PNADs) shall be worn by all workers assigned to work areas where a nuclear accident is possible. The dosimeters shall be capable of determining gamma dose from 0.1 to 1000 rad with an accuracy of ±20% and an neutron dose from 0.1 rad to 1000 rad without dependence upon fixed nuclear accident dosimeters data (see Section 3.6.5) to an accuracy of ±30% in accordance with DOE 5480.11 (DOE 1988).

4.9 RADIATION SAFETY TRAINING

A thorough radiation protection training program shall be established at plutonium facilities. Separate training programs should be established for general employees, radiation workers, and RPTs. The training of all staff members shall be carefully documented. DOE 5480.11 (DOE 1988) provides guidance on the information that needs to be presented during the training programs. A summary of that information is presented in the following subsections. Moe (1988) provides additional guidance on operational health physics training.

DOE requires at least biennial training. Individuals who work with unencapsulated plutonium should be retrained annually.

4.9.1 Radiation Safety Training for All Employees

All employees who work in radiation areas shall receive an orientation in radiation safety within one month of their initial employment. Retraining shall be provided at least every two years or when there are significant changes to radiation protection policies and procedures that affect general plant employees. The initial orientation shall include, but is not limited to:

- the risk of low-level occupational radiation exposure, including cancer and genetic effects (NRC [1981] and Moe [1988] are excellent references)
- the risk of prenatal radiation exposure
- facility orientation

- posting requirements
- emergency response signals
- basic radiation protection concepts
- DOE and company radiation protection policies and procedures
- employee and management responsibilities for radiation safety
- emergency procedures.

4.9.2 Radiation Safety Training for Radiation Workers

Programs for initial training and periodic retraining (at least every 2 years) for radiation workers shall be established and conducted to familiarize the worker with the fundamentals of radiation protection and the proper procedures for maintaining exposures ALARA. Training should include both classroom and applied training. This training shall precede or be concurrent with each worker's assignment as a radiation worker provided they work under the supervision of a trained individual. Each radiation worker's knowledge of radiation safety fundamentals should be certified by examination (both written and practical) prior to an unsupervised assignment. The level of training in the following topics shall be commensurate with each worker's assignment:

- radioactivity and radioactive decay
- characteristics of ionizing radiation
- manmade radiation sources
- acute effects of exposure to radiation
- risks associated with occupational radiation exposures
- special considerations in the exposure of women of reproductive age
- dose equivalent limits
- modes of exposure--internal and external
- dose equivalent determinations
- basic protective measures--time, distance, shielding
- specific plant procedures for maintaining exposure ALARA

- radiation survey instrumentation--calibration and limitations
- radiation monitoring programs and procedures
- contamination control, including protective clothing and equipment and workplace design
- personnel decontamination
- emergency procedures
- warning signs and alarms
- responsibilities of employees and management
- interaction with radiation protection staff
- operational procedures associated with specific job assignments.

4.9.3 Radiation Safety Training for Radiation Protection Technologists

Programs for initial training and periodic retraining (at least every 2 years) for RPTs shall be established and conducted to familiarize the technologists with the fundamentals of radiation protection and the proper procedures for maintaining exposures ALARA. Radiation protection technologists should be certified by the National Registry of Radiation Protection Technologists [NRRPT]. The training programs shall include both classroom and applied training and shall precede or be concurrent with each worker's assignment as an RPT while under the supervision of a trained individual. Each RPT's knowledge of radiation safety fundamentals should be certified by examination prior to an unsupervised work assignment. The training program shall include but not be limited to the topics listed in Section 4.8.2 and should emphasize procedures specific to the facility where the RPT is assigned. The level of training in each topic should be commensurate with the RPT's assignment.

4.9.4 Training Records

The training records of plant employees, radiation workers, and radiation safety personnel shall be retained to document the level of understanding and proficiency of personnel who work with radioactive materials. Training records shall include information about the level and type of training and retraining. Outlines of course content, copies of handouts, lecture materials, and tests represent typical information that shall be included in training files. Records that demonstrate the successful completion of training programs and the performance of the participants shall also be retained.

A written test should be given after the training program to ensure that the information is understood by the employees. Personnel should successfully complete initial training in order to be allowed to work independently. Retraining should be completed successfully for independent work to continue. Audits shall be performed to ensure that the objectives of the training program are fulfilled. An audit should include an assessment that the training program is conducted effectively, proper records are maintained, the training instructors are qualified and competent, the desired level of competence in trainees is achieved, and that retraining is effective and timely.

4.10 ALARA AND OPTIMIZATION

Limiting radiation exposures to the lowest levels commensurate with economics and the work to be accomplished has long been a part of health physics and radiation protection programs of DOE and its contractors. DOE 5480.11 (DOE 1988) establishes the policy to maintain exposures of workers and the public to radiation from DOE operations to within the limits of ALARA. This order also requires that plans and programs be prepared and implemented and that records be maintained to demonstrate the implementation of ALARA. DOE/EV-1830-T5, Guide to Reducing Radiation Exposures to As Low As Reasonably Achievable (ALARA) (DOE 1980a) may be used as a guide in developing an ALARA program.

A significant facet of an ALARA program is the practice of optimization and the use of cost-benefit analysis. While implementation of ALARA evaluation, optimization, and cost-benefit analyses are most cost-effective in the design of a facility, analyses should be applied to all phases of the facilities activities. Detailed descriptions of the use and application of optimization and cost-benefit analyses can be found in ICRP 37, Cost Benefit Analysis in the Optimization of Radiation Protection (ICRP 1982), Application of the Dose Limitation System for Radiation Protection (IAEA 1979), and The Dose Limitation System in the Nuclear Fuel Cycle and in Radiation Protection (IAEA 1982).

4.11 RADIATION PROTECTION PROGRAM RECORDS

The systematic generation and retention of records relating to occupational radiation protection program are essential to describe the occupational radiation exposure received by workers and the conditions under which the exposures occurred.

DOE 5480.11 (DOE 1988) establishes minimum record requirements. Records of the following shall be maintained:

- ALARA programs
- individual occupational dose (i.e., internal exposure, external exposure, summation of internal and external dose equivalent, and programs to determine individual doses)
- monitoring and area control (i.e., records that establish the conditions under which individuals were exposed)
- monitoring methods (i.e., records to document the appropriateness, quality and accuracy of monitoring methods)
- training records.

Additional guidance on internal dosimetry records and training records can be found in Sections 4.6 and 4.8, respectively. Guidance on monitoring and area control records and monitoring method records can be found in ANSI N13.6-1972, Practice for Occupational Radiation Exposure Records Systems (ANSI 1972), DOE 1324.2 (DOE 1980b), and DOE 5484.1 (1987a).

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SECTION 5.0

EMERGENCY PREPAREDNESS

5.0 EMERGENCY PREPAREDNESS

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, DOE, in DOE 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. Procedures for implementation provide specific instructions to facility 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) must 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) must 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 plutonium facilities might not include any members of the general public.

This section addresses the requirements and good practices pertaining to emergency planning at plutonium facilities, concentrating on these practices as they pertain to the facility-specific emergency plans and procedures. Although specific to plutonium facilities, much of the guidance provided may apply to any facility handling radioactive or other toxic and hazardous materials.

5.1 EMERGENCY PLAN

The organization and assignment of responsibility, emergency response support and resources, emergency response levels, notification methods and procedures, emergency communications, emergency facilities and equipment, accident assessment, protective response, radiological exposure control, medical and health support, recovery and re-entry, and maintenance of emergency preparedness are integral components of an emergency plan for a plutonium facility. These components are discussed in the following subsections.

5.1.1 Organization and Assignment of Responsibility

The emergency plan shall clearly define the onsite emergency organization for all shifts in order to maintain continuous (24 h/day) response capability. The plan shall designate the individual that has the responsibility and authority to immediately initiate emergency response actions. This individual is referred to by different titles depending on the facility and DOE field office. At the facility this individual is typically referred to as the emergency director. A line of succession for the emergency director should be identified. The emergency director's duties should specify the responsibilities that should not be delegated during an emergency. Among them 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. The last two responsibilities should typically rest with the DOE field office emergency director with the DOE contractor (facility) emergency director providing support in these areas.

Organizations or individuals that support the emergency director should be described to identify specific functions, interrelationships with other support groups, and the individuals and alternates responsible for the functions. Emergency duty assignments should correspond as closely as possible to normal duty assignments. Typical support functions should include

- Technical Support
- Communications
- Notifications
- Data Distribution
- Administration and Logistics
- Documentation
- Quality Assurance
- Design and Construction
- Plant System Assessment
- Corrective Actions
- Procedure Support
- Maintenance and Repair
- Operational Support
- Health Physics Support
- Radiological Assessment
- Criticality Safety
- Chemical Analysis Support
- Public and Media Information
- Security
- Fire Control
- Medical Services.

Organizational charts that show the primary responsibilities and reporting lines may be useful in ensuring clear delineation of functions.

Emergency plans shall identify the local, state, and federal organizations that are part of the overall response organizations responsible for the emergency planning zones. The involvement, responsibilities, and authorities of each group shall be specified as well as their methods of contact. The individuals and alternates assigned to supervise each response group should be specified by job title.

5.1.2 Emergency Response Support and Resources

As part of the support functions listed above and in addition to onsite organizations, 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

- firefighting
- law enforcement
- medical services
- radiological monitoring
- 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 must identify the organizations expected to provide emergency response support, give a brief description of their capabilities, and specify the items of agreement. These agreements should be formalized in Memoranda of Understanding or Letters of Agreement.

5.1.3 Emergency Response Levels

To be consistent with other government agencies, DOE, as defined in DOE N5500.2 (DOE 1987b), 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 that normally would not constitute an emergency but that indicates a potential reduction of safety of the facility. No potential exists for significant offsite release of radioactive or other toxic 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 that involves an actual or potential substantial reduction of the level of the nuclear safety of the facility. Limited offsite releases of radioactive material may occur. For other toxic materials, offsite releases are not expected to exceed applicable, permissible limits. The purpose of an Alert level is

to assure that onsite and offsite emergency response personnel are promptly 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 process or having occurred that involves actual or likely major failures of facility functions needed for the protection of onsite personnel, the public health and safety, and the environment. Releases of radioactive material offsite, not exceeding protective response recommendations (PRRs), are likely or are occurring. For other toxic materials, offsite releases have the potential to exceed applicable permissible limits. The purpose of the site emergency level is to assure that emergency control centers are staffed, 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 that involves actual or imminent substantial reduction of facility safety systems. Releases offsite are occurring or are expected to occur and exceed PRRs. Offsite releases of other toxic materials are expected to exceed applicable permissible limits. The purpose of the general emergency level is to initiate predetermined protective actions for onsite personnel, the public health and safety, and the environment, and to 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 depending on present and anticipated plant conditions. The conditions that would determine the classification of an emergency response level are specific to each site and plant and are called emergency action levels (EALs). The EALs are specific plant instrument readings or plant conditions that, if met, would result in the declaration of the appropriate emergency response level (Unusual Event, Alert, Site Emergency, or General Emergency). EALs should be developed and formalized in procedures for use in guiding emergency response personnel.

5.1.4 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 for notifying personnel, both onsite and offsite, of an event must be developed and described in the emergency plan. Notification of onsite personnel may be accomplished by the plant paging system, plant alarm, telephone, portable radio, pagers, or combinations of these. Typically, a primary method is established and other means are used for back-up. The plant evacuation alarm should meet the criteria in ANSI N2.3-1979, Immediate Evacuation Signal for Use in Industrial Installations (ANSI 1979). Offsite agencies may be notified by dedicated telephone, normal telephone, radio, teletype, or combinations of all of these methods. Because of the importance of accurate data and message transmission, especially in the transmission of emergency classification, release data, and protective action recommendations, common terminology and a standard format for information should be established.

5.1.5 Emergency Communications

The emergency plan must identify and describe the communications systems to be used by the response organizations, emergency personnel, and federal, state, and local agencies. A primary communications system and at least one back-up 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 exchange of information. Routine periodic testing of communications devices is necessary to ensure that they are available and functioning in an emergency. Frequency of testing will depend on the equipment used and should be specified in the emergency plan. Documentation of equipment testing must be maintained.

5.1.6 Emergency Facilities and Equipment

Emergency facilities and equipment must be available 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 and the operations performed at the site. While specific site needs will vary, four essential functions must be included. These are 1) the command, 2) technical support, 3) operational support, and 4) alternate command functions.

The command function is centered in an in-plant location designated as the Emergency Control Station (ECS). The ECS provides the primary location from which designated management personnel can assume immediate direction of the response to an emergency and from which efforts can be made to control the processes and establish safe shutdown conditions. The ECS may also be the central location for communications and information on plant status. Consideration should be given to ensuring habitability of the ECS or establishing an alternate ECS during an emergency that requires general evacuation so that appropriate control and mitigating actions can be taken.

Management and support personnel carry out emergency response activities in the Emergency Operations Center (EOC). The EOC may be a dedicated facility, office, conference room, or other predesignated location that has appropriate communications and informational materials to support the emergency. It should be located in a secure and protected location.

The technical support function provides engineering and technical support to the emergency response effort. This function should be staffed by the technical experts for the facility processes. The experts should have ready access to all information about plant operations, maintenance, and design that would assist them in diagnosing and correcting the plant problems. The technical support staff should provide technical information and recommendations to the ECS and to personnel concentrating on problem correction and mitigation of the accident. To permit necessary consultation and interaction between operators and engineers, the technical support function should be located as close as possible to the personnel it supports, or it should have dedicated communication links with them. The technical support function would likely be centered in the EOC.

The operational support function provides the staff that may be required to perform repairs and corrective actions 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 that is not affected by the event and that has access to any tools and equipment needed. Typically, operational support would be provided from the ECS.

Equipment must 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 firefighting personnel. Tables 5.1 and 5.2 give example contents of these kits. Because emergency kits may serve more than one type of plant within a facility, kit contents could require items not applicable to a particular plant.

TABLE 5.1. Typical Items in an In-Plant Monitoring Kit

- Dosimetry
  - Dosimeter charger
  - Self-reading dosimeters (0-200 mR, 0-5 R, 0-100 R)
  - TLDs
- Instruments
  - Administrative supplies
    - Log books
    - Paper, pens
    - Procedures
  - Alpha survey meter
  - Flashlights
  - GM with thin window (0-100K cpm)
  - High-volume air sampler (particulate)
  - Instrument check sources
  - Ion chamber (0-100 R/h with beta/gamma distinction capability)
  - Knife and scissors
  - Personnel decontamination kits
  - Prefilter and iodine cartridges
  - Portable radios
  - Posting signs and step-off pads
  - Rope
  - Sample bags
  - Spare batteries for all electronic devices
- Protective Clothing
  - All-weather jackets
  - Cloth and plastic overalls
  - Cloth foot covers
  - Gloves (rubber and work)
  - Head covers
  - Masking tape
  - Rainsuits
  - Respiratory protection (SDBAs and full face respirators)
  - Rubber boot covers
  - Spare air bottles (charged)

TABLE 5.2. Typical Items in an Offsite Monitoring Kit

- Dosimetry
  - Dosimeter charger
  - Self-reading dosimeters (0-20 mR, 0-5 R)
  - TLDs
- Instruments
  - Administrative supplies
    - Credit card for gasoline
    - Log books
    - Maps
    - Paper, pens
    - Procedures
    - Roll of quarters (for telephone)
    - Sample bags and labels
  - Air sampler(s) (particulate)
  - Alpha survey meter
  - Analytical instrument to measure 10^{-7} μCi/cc iodine and 10^{-9} μCi/cc particulate
  - Decontamination solution
  - Flashlights
  - Generator or adaptor to run air sampler from vehicle battery
  - GM with thin window (0-200 mR, 0-5 R)
  - Instrument check sources
  - Ion chamber (with beta/gamma distinction)
  - Knife and scissors
  - Portable radios
  - Prefilter and iodine cartridges
  - Rope
  - Spare batteries for all electronic devices
- Protective Clothing
  - Cloth or paper overalls
  - Full-face respirators with particulate and charcoal cartirdges
  - Gloves (rubber)
  - Head covers
  - Masking tape
  - Rain gear
  - Rubber boot covers

Equipment contents and packaging, for both onsite and offsite emergency kits, should be made to support the team size designated for the function. Emergency equipment such as in-plant monitoring kits and offsite monitoring kits should be located in areas that typically would be accessible during an emergency. It may be advantageous to have equipment located in multiple locations so that at least one would be accessible during an emergency.

5.1.7 Accident Assessment

Methods and equipment for monitoring and assessing the actual or potential offsite consequences of a radiological emergency condition must be available and operational. There is a variety of ways to monitor an actual release, anticipate a possible future release, and calculate its consequences. While complete coverage is not appropriate for this document, a brief discussion of the capabilities associated with good practices may be useful.

Actual releases are best determined by installed effluent instrumentation. Isokinetic stack samplers and monitors with alpha-detection capabilities are preferred. These give a fairly accurate readout of the activity in effluent gases that are being released to the atmosphere. Other instruments, such as portable or area monitors, can also be useful in determining source terms.

Anticipated release projections may be made based on plant parameters and failed systems. Procedures should provide estimated source terms at various release points for different system failure scenarios. This provides prompt availability of source term data when needed.

Emergency action levels should be developed for projected and actual offsite dose rates and/or doses. For example, when actual or projected dose rates at the site boundary exceed a given emergency action level, a certain emergency response level (Unusual Event, Alert, Site Emergency, General Emergency) would be declared.

Near-site and offsite consequences of a release are usually determined by calculations from a plume model and compared with actual field monitoring data. In order to maintain this capability, appropriate equipment, instruments, and procedures must be in place and operable.

Any computer program chosen to calculate radiation doses from plutonium should have the capability to determine the 50-year committed dose equivalent from the uptake of plutonium. The calculational method should be consistent with the methods of ICRP 30 (ICRP 1979). The short time required to perform a dose calculation using a computer is very desirable; however, a back-up method independent of normal electric power should be available. The back-up method could be performed on a hand-held calculator or portable computer. A hand calculation method would also be acceptable as a back-up.

If different dose calculation methods are used by the facility, DOE field office, and any state emergency response organization, an evaluation should be performed to determine if the methods yield equivalent results. Reasons for any differences should be documented so that this is not an unnecessary source of confusion during an actual emergency.

Basic information, guidance, and detailed references for radiological assessment may be found in Radiological Assessment by Till and Meyer (1983), in IAEA Safety Series 152 (Evaluation of Radiation Emergencies and Accidents, 1974), IAEA Safety Series 86 (Techniques and Decision Making in the Assessment of Off-Site 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 plutonium source term can be rather difficult in most situations, especially at an unmonitored release point, it is recommended that field monitoring be implemented. A comparison of field data and calculated values can be very valuable in confirming the accuracy of the source-term estimate. It may also confirm or invalidate the need for protective actions for changing emergency response levels.

Field monitoring teams must be well equipped to fulfill their mission. As a minimum, kits should have the items listed in Section 5.1.6, and they should be located in an area that is accessible during any type of accident. The number of teams that may be required is dependent upon the facility and should be designated in the emergency plan.

5.1.8 Protective Response

Protective responses taken to avoid or minimize personnel and public exposures to a plutonium 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 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 office 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.

5.1.9 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 (DOE 1988) 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.

In order 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 plutonium facilities, high levels of external radiation exposure could only result from an accidental criticality. In a criticality event, special precautions must be exercised to ensure that radiation doses to individuals exposed to neutrons are considered, that the reaction has ceased, and that emergency actions do not reinitiate the criticality.

5.1.10 Medical and Health Support

Medical services for personnel who are injured during an emergency shall be provided. Medical support services for plutonium facilities should work closely with the health physics staff to ensure that appropriate medical procedures such as excision and administration of chelating agents are performed when necessary. Special training may be required to ensure the maximum protection from intake of plutonium and the minimum trauma to the injured. Medical attention at a plutonium facility may be somewhat complicated because of potential plutonium contamination. For this reason, special preparations must be made to treat, handle, transport, and decontaminate patients (Wick 1967).

Skin contamination in the area of an open minor wound should be cleaned so as not to introduce material into the wound. This prevents external contamination from entering the body. Skin contaminants in areas away from open wounds may be cleaned using normal decontamination procedures, if the wounds have been cleaned, dressed, or otherwise protected from contaminants.

Major injuries that require patient transport to a medical facility and involve contamination must be treated differently. Decisions may be required about how to maximize decontamination efforts without hindering the medical treatment required for the health of the patient. The situation may not allow for decontamination efforts before transport because of the seriousness of the injury. Procedures must provide guidelines for these decisions.

In the event of a confirmed or suspected uptake of plutonium material in the respiratory system or through a wound, a chelating agent may be valuable in reducing the deposition. The chelating agent DTPA is a commonly used agent that aids in ridding the body of plutonium. The administration of DTPA is most effective if given soon after the intake occurs. However, administration of DTPA has been shown to be effective in reducing permanent deposition even if given days after the uptake. Procedures must ensure that chelating therapy be initiated as soon as possible by qualified personnel after authorization from a physician is obtained.

It is highly desirable to use a site vehicle to transport injured and contaminated persons offsite. The site-controlled vehicle can be equipped for contamination control at all times. Vehicles from offsite used to transport contaminated patients may need extra time for contamination control preparation. However, medical treatment normally takes precedence over contamination control and extra time for contamination control preparation of vehicles should only be taken if the patient's medical condition permits. The floor and sides of the ambulance interior should be covered with a plastic sheet and absorbent pads. Ambulance drivers and attendants should be supplied with light protective clothing, and, in severe cases, respiratory protection.

Each site should have access to a medical facility that is capable of treating a full range of medical emergencies with the contaminants involved. If offsite public facilities are used, advanced planning should ensure that sufficient supplies, equipment, and preparations are in place to minimize medical personnel exposures to contaminants. Similarly, advanced planning should be done for any onsite medical facilities.

5.1.11 Recovery and Re-Entry

After emergency conditions have stabilized and the plant is in a safe shutdown status, recovery of the facility can begin. This effort consists of working 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 before re-entry to ensure that adequate precautions and controls are established to protect the health and safety of workers.

5.1.12 Maintenance of Emergency Preparedness

In order 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:

- purpose of emergency planning
- emergency organization
- interrelationships between organizations
- training on specific duties and procedures
- training on protective actions
- 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:

- state, local, and municipal agencies (law enforcement, fire protection, and public health)
- state, local, and municipal government officials
- private medical doctors, hospital staffs, and the staffs of emergency rescue organizations and ambulance services
- volunteer fire department personnel
- military personnel (where appropriate)
- private industry emergency services personnel.

The training for offsite groups should include

- some plant specifics (what are the processes, potential source terms, type of possible accidents, etc.)
- discussion of emergency response levels and emergency planning zones
- site organization and offsite organization structure and responsibilities
- 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 improvement 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.

5.2 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 plutonium 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.

5.2.1 Organizational Procedures

Written procedures are required to describe the lines of authority and responsibility for the staff assigned to emergency response functions. These procedures should include identification of responsibilities that 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 communications requirements shall be coordinated with appropriate headquarters organizations as required by DOE 5320.1A, Telecommunications: Spectrum-Dependent Services (DOE 1980a).

Emergency action levels should be developed and formalized in procedures for use in guiding emergency response personnel. Emergency procedures should define the facility-specific conditions under which each of the four emergency response levels shall be declared. The procedures should also describe the criteria and process for escalation and de-escalation of the emergency response levels.

5.2.2 Personnel Evacuation and Accountability Procedures

Written procedures shall be developed for determining the conditions under which evacuation is recommended. They shall also include designation and use of staging areas and assembly points, determination of sheltering 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 ensuring the security of classified matter and all source material, byproduct material, and special nuclear material, and for the evacuation of disabled persons.

5.2.3 Emergency Notification

Written procedures shall be established for the 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. Emergency notification procedures should be established for each of the DOE emergency response levels.

Procedures shall be developed and coordinated with appropriate agencies for handling of alerting notifications received via the National Warning System (NAWAS).

5.2.4 Readiness Actions

Emergency procedures shall include those procedures required for readiness actions by security personnel in response to emergency conditions, security and safeguards alerts, and emergency information. Radiation protection for the security personnel should be included. The procedures shall provide for automatically placing in effect an appropriate security and safeguards alert upon receipt of notification of an emergency condition. Because the security system and response actions may involve national security, some of the emergency procedures may require a classified status in accordance with DOE 5630.1, DOE 5630.2, DOE 5632.2, and DOE 5632.4 (DOE 1979a, DOE 1980b, DOE 1979b, and DOE 1985).

5.2.5 Personnel Monitoring and Decontamination

Procedures shall be established to monitor personnel exposed to toxic and radioactive materials and to decontaminate personnel and equipment. As much as possible, normal facility procedures should be used with provisions for emergency conditions so that personnel are familiar with their implementation.

Procedures should provide for authorizing radiation exposures in excess of the normal limits and for the control and recording of doses received.

5.2.6 Medical Emergency Requirements

Procedures shall be established for the search and rescue of missing or injured persons. For plutonium facilities, special consideration must be given to protecting search and rescue personnel from inhalation of radioactive material.

Procedures should include provisions for the transport of injured personnel to offsite hospitals and the contamination control measures necessary for protection of transport personnel and medical staff and facilities.

5.2.7 Emergency Equipment and Supplies

Procedures shall be established that specify the location of emergency equipment and supplies and the system for ensuring that they are available when needed. The procedures should include a list of the contents of emergency kits and the inventory responsibility and frequency. A check list 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 time of limited-life components.

5.2.8 Re-Entry Procedures

Re-entry into a facility after an emergency shall be controlled to the extent practicable by written procedures. The procedures should define the authority and responsibility for the decision to re-enter. A method 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.

After emergency conditions have stabilized and the plant is in a safe shutdown condition, recovery of the facility may begin. This phase consists of efforts to return the facility to its preaccident condition. While the plant may be in a safe shutdown condition, extra precautions may be needed during restoration work because of potential or actual damage to safety systems, process equipment, and/or structures.

Procedures must be available that designate the personnel with the authority and responsibility for re-entering the damaged facility. The individual(s) with that authority must ensure that all re-entry criteria have been satisfied (including an accountability system) before entry approval is given. Procedures should also require

- establishment of exposure guidelines for repair workers
- accurate determination of facility damage
- securing adequate stocks of protective equipment and clothing.

5.3 GENERAL INFORMATION

This section discusses the administrative aspects of an emergency program, the need for a periodic review and audit of the emergency management system (emergency plans and procedures), and addresses the special support functions within the system that are available as potential emergency resources.

5.3.1 Program Administration

Emergency plans and procedures must be reviewed annually. The facility emergency planning coordinator should have the responsibility for reviewing the facility emergency plan and procedures. This review should ensure that all equipment identified in the plan and procedures is still in place and functioning, that all Memorandums of Understanding and Letters of Agreement are current, and that lists of emergency response personnel are current. Changes identified from drills and exercises should also be made at this time. If major deficiencies are identified as a result of a drill or exercise, they must be corrected promptly and the plan or procedure reissued. The emergency planning coordinator shall not wait until the annual review to incorporate major changes.

Other administrative functions include routine testing of emergency communication systems and routine inventories of emergency equipment supplies. Documentation of equipment testing and inventories must be maintained.

5.3.2 Audit and Review

The preparation of emergency plans and procedures and the maintenance of a condition of readiness to respond to an emergency condition must be supported by a program to ensure that all needs have been addressed and that each need is satisfied. All facets of the facility's emergency management system should be periodically reviewed and audited. The review and audit should be conducted by individuals who are knowledgeable of the processes and operations of the facility and of emergency response requirements but who are independent of the responsibility for plant performance. This function is typically performed by the QA organization for the facility or the DOE field office emergency preparedness coordinator. Audits of facility emergency preparedness programs are also conducted during DOE-HQ Technical Safety Appraisals. The review and audit should include verification of compliance with requirements, adequacy of plans and procedures, compliance with procedures, and quality of performance. The program for ensuring the quality of the emergency management system should provide for tracking, follow-up, and closeout of findings from the reviews and audits.

5.3.3 Potential Emergency Resources

Each DOE field element is responsible for ensuring that an effective emergency management system is established and maintained for facilities, operations, and activities under its jurisdiction. The emergency management system must be consistent with the responsibilities and functions assigned by DOE orders in the 5500 Series, including DOE 5500.1A, DOE N5500.2, DOE 5500.3, and DOE 5500.4 (DOE 1987a, DOE 1987b, DOE 1981a, and DOE 1981b). Included in the emergency management system are provisions for liaison and coordination with other federal agencies and state and local governments as appropriate.

While each field element has its own plans and procedures, there are special support functions within the system that are available during an emergency. Operators of plutonium facilities should be aware of these support functions.

Headquarters Emergency Management Team

The Headquarters Emergency Management Team was established to manage and coordinate DOE-HQ emergency response and support for each type of emergency. The team membership consists of individuals at the Deputy Assistant Secretary/Office Director level who have authority to commit the resources of their respective organizations.

Headquarters Operational Emergency Management Team

The Headquarters Operational Emergency Management Team has the authority and responsibility for management of operational emergency situations. The head of the field element that has primary jurisdiction over the affected facility or activity is responsible for onsite direction and control of specific activities. The Headquarters Operational Emergency Management Team coordinates DOE-HQ support of the field element response and maintains DOE-HQ oversight and interface with federal agencies at the national level, congressional delegations, Washington media, and the White House. The team also coordinates use of DOE field element resources outside the area of responsibility of the affected field element, and may authorize deployment of resources to support a field element in an emergency or in assisting another federal agency or state or local government.

Accident Response Group

The Accident Response Group (ARG) provides a special resource capability for response to emergencies involving nuclear weapons or strategic quantities of government-owned SNM in the transportation safeguards system, space nuclear systems, defense nuclear energy systems, and isotope applications. This group is the responsibility of the Albuquerque Operations Office.

Nuclear Emergency Search Team

The Nuclear Emergency Search Team (NEST) provides a special resource capability that has specialized equipment for conducting radiation survey and detection, field communications, explosive ordnance disposal support, bomb/weapons diagnostics, hazard prediction, damage mitigation, and decontamination. This group is the responsibility of the Nevada Operations Office.

Joint Nuclear Accident Coordinating Center

The Joint Nuclear Accident Coordinating Center (JNACC) is a combined Defense Nuclear Agency and DOE centralized agency for exchanging and maintaining information about radiological assistance capabilities and coordinating assistance activities. This center is the responsibility of the Albuquerque Operations Office.

Radiation Emergency Assistance Center/Training Site

The Radiation Emergency Assistance Center/Training Site (REACTS) provides a special resource capability for medical treatment and consultative services for emergencies, and also provides training courses. This resource is the responsibility of the Oak Ridge Operations Office.

Regional Response Teams

Each DOE Regional Coordinating Office has a Regional Response Team that is maintained in a state of readiness to provide effective and immediate assistance as needed to cope with radiological incidents.

Atmospheric Release Advisory Capability

The Atmospheric Release Advisory Capability (ARAC) is a special resource capability that provides computer model estimates of the distribution of contamination resulting from a nuclear weapons accident. The model includes projections of the location and levels of contamination patterns deposited on the ground and projections of the possible radiation dose to people in the area. While the service is directed to nuclear weapons accidents, it could be useful for other accidents involving plutonium at locations where site-specific computer models are not available. This resource is directed by LLNL.

Aerial Measuring System

The Aerial Measuring System (AMS) is a special resource that is designed to survey large areas rapidly for ground and airborne radiation. Major components of the system include gamma and neutron radiation detectors, data formatting and recording equipment, positioning equipment, meteorological instruments, direct readout hardware, and data analysis equipment. A variety of aircraft is provided to support this capability. Contact for use of this resource may be made through the DOE-HQ EOC or JNACC.

Mobile Accident Response Group Unit

The Mobile Accident Response Group Unit is an element of the ARG that provides a readily mobile system for analysis, identification, and documentation of radioactive contamination. The unit consists of two trucks and two trailers designed to be transported by aircraft. This system is the responsibility of LLNL.

Department of Defense

The Department of Defense maintains substantial emergency capability, primarily for response to nuclear weapons accident. However, these resources could be made available in other situations if necessary. Contact for these resources would be made through the DOE-HQ EOC and JNACC to the National Military Command Center. Some of the specialized resources that are established include the U.S. Army Radiological Control Team (RADCON), U.S. Army Radiological Advisory Medical Team (RAMT), U.S. Air Force Transportable RADIAC Package (ATRAP), Headquarters Air Force Communications Command (HAMMER ACE), and Headquarters Tactical Air Command (HARVEST EAGLE). These resources provide readily transportable personnel and equipment for radiological surveillance and decontamination, medical advice and assistance, secure communications systems, and field operations support sets.

Federal Emergency Management Agency

The Federal Emergency Management Agency (FEMA) is responsible for the establishment of federal policies for civil emergency planning and the coordination of all civil emergency planning, management, mitigation, and assistance functions of executive agencies. The primary role of FEMA during an emergency at a DOE facility is to coordinate requests for assistance from federal agencies and to ensure that offsite actions and response activities of federal, state, and local officials are mutually supportive and coordinated with the onsite actions of DOE.

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SECTION 6.0

DECONTAMINATION AND DECOMMISSIONING

6.0 DECONTAMINATION AND DECOMMISSIONING

The term "decontamination and decommissioning" (D&D) is typically used to describe the activities at the end of the useful life of a facility that are needed to restore it to noncontaminated status and permit its unrestricted use.

When plutonium facilities are no longer useful and operational activities are no longer conducted, measures must be continued to ensure that control of residual radioactivity is maintained. The decision may be made to undertake a D&D program to minimize or eliminate long-term institutional control. This may be done in a variety of ways, all of which (except converting the facility to some other nuclear use) may be termed D&D. Planning for D&D should be initiated in the facility design phase.

A description of different D&D options may be found in the DOE Decommissioning Handbook, DOE/EV/10128-1 (DOE 1980). This handbook also includes guidance on decontamination techniques, assessment of environmental impacts, disposition of wastes, and preparation of decommissioning cost estimates.

6.1 DECONTAMINATION AND DECOMMISSIONING CRITERIA

The general criteria for D&D of DOE surplus facilities are found in DOE 5820.2, Chapter V (DOE 1984), which defines the DOE policy and the protocol for preproject and project activities. The residual radioactivity levels permitted and the potential need for EISs for D&D activities are discussed in the following subsections.

6.1.1 Residual Radioactivity Levels

A primary concern in the D&D of any nuclear facility is the level of residual radioactivity that may be permitted. The facility may be decontaminated to levels where no controls are needed to limit the exposure of the public to acceptable levels. There is currently little firm guidance as to what acceptable levels are. The NRC in Regulatory Guide 1.86 (AEC 1974) and HPSSC N13.12 (HPSSC 1988) provide definitive values for acceptable surface contamination levels for termination of operating licences for nuclear reactors and for materials, equipment and facilities.

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 formerly contaminated or controlled radioactive facilities for unrestricted release. Such guidelines will likely be based on the radiation dose to the maximum exposed member of the general population. They may also include limits for population dose. Although Vaughan (1985) establishes the maximum dose that may be received by a member of the general population from the operation of a DOE facility at 100 mrem/yr, the EPA limits will almost certainly be lower. Values of 50, 10, 1, and 0.1 mrem/yr are being considered by the EPA as the "de facto de minimis" levels for the disposal of contaminated material.

In Subpart H of 40 CFR 61 (CFR 1986a), the EPA established dose limits for air emissions from DOE facilities. Emissions from air shall not exceed 25 mrem/yr to the whole body or 75 mrem/yr to the critical organ of any member of the public. In 40 CFR 190 (CFR 1986b), the EPA has established annual dose equivalent limits of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials (radon and its decay products excepted) to the general environment from uranium fuel cycle operations and to radiation from these operations. In 40 CFR 141, "National Primary Drinking Water Regulations" (CFR 1986c), the EPA provides a limit of 4 mrem/yr, annual dose equivalent to the whole body or any internal organ from manmade radionuclides in drinking water.

While none of these guidelines is directly applicable to the D&D of DOE facilities, they do provide some perspective of the magnitude of residual radioactivity that may be permitted when final guidance is provided. The EPA is developing public radiation protection criteria for residual radioactivity following cleanup of contaminated lands and facilities. However, statements by EPA staff in February 1986 indicated that the analyses may take several years to complete (HPSSC 1986).

6.1.2 Decommissioning Regulations

The decommissioning of a DOE plutonium facility will require a determination about whether or not the action is a "major or significant government action adversely affecting the environment." If it qualifies as such an action an EIS will be required. If the action does not require an EIS, either because the possible adverse impacts are insignificant, or because decommissioning was adequately addressed in a preoperational 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 contained in DOE 5440.1C, National Environmental Policy Act (DOE 1985), and 40 CFR 1500 through 1508 (CFR 1986d through 1). The EIS will need to address the amount of material that will remain onsite and its impact in addition to addressing the alternatives. The alternatives will include retaining radioactive material onsite under DOE control, cleaning the site to a level that would be acceptable for unrestricted release, and the null or no action alternative of "walk away."

The regulations that address waste transportation and disposal also 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. Criteria of Class A, B, and C waste are established in 10 CFR 61 (CFR 1985), but the code does not specify any procedures or disposal possibilities for less than Class A waste.

Facilities must also comply with the Resource Conservation and Recovery Act (RCRA) and the Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (CERCLA) (GI 1986). The objectives of RCRA are to promote the protection of health and the environment and to conserve valuable material and energy resources. CERCLA provides for liability, compensation, cleanup, and emergency response for hazardous substances released into the environment and the cleanup of inactive hazardous waste disposal sites.

6.2 DECOMMISSIONING DECISION-MAKING

Currently, decisions to clean a facility to a level that permits unrestricted release are extremely difficult to make because the levels are only established by precedent and by the EIS process. There is no firm guidance. Such guidance would greatly facilitate decommissioning.

Even with such guidance there will still be important decisions to be made regarding decommissioning, such as when it can be assumed that the government will maintain control of the site. A detailed discussion of the decommissioning planning process can be found in the DOE Decommissioning Handbook (DOE 1980). 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 addressed separately in Section 6.3.

6.2.1 Background Radiation Levels and Facility Records

The determination of the natural background levels and the levels of natural background and fallout radionuclides is the most critical step in decommissioning, which is best begun before the facility becomes operational. These levels need to be determined so that the incremental dose occurring from material left onsite at the termination of operations can be assessed.

The contamination control practices and records maintained during facility operation will be important. If paint is used in contamination fixation (a seldom optimum, but sometimes necessary practice) it should be of a distinctive color, and the location should be permanently recorded. Other records are also helpful in planning and executing final decontamination for dismantlement. Operational incidents and practices, such as spills, pipe and tank leaks, and ventilation failures, burial of low-level radioactive or potentially radioactive materials onsite, or other actions that may impact decommissioning, shall become part of the permanent record of the facility and be considered in decommissioning planning.

6.2.2 Criteria

The first step in planning decommissioning is the development of a series of required criteria. These criteria will necessarily include such items as compliance with DOE and EPA regulations and other requirements. They may also include commitments to states, landowners, or others, and provisions of the original EIS.

As these criteria are developed, other high-value criteria should also be established. These are likely to include such considerations as maximizing the aesthetic and recreational value of the site, performing decommissioning within allocated funds, lowest worker dose, lowest population dose, lowest cost, lowest future surveillance commitment, and least impact in case of probable accidents. Depending on the visibility of the decommissioning action, the decision-making process that has been established, and the level of public concern, an EIS may be required. When it is, a notice of a scoping meeting may be issued and published in the Federal Register. Scoping meetings may assist in determining the applicable criteria and the alternatives that are considered.

Whether or not a formal scoping meeting and an EIS are used, it is necessary to define the decommissioning options. Most effort should be expended analyzing those options that fulfill the required criteria so that they can be ranked relative to the other high-value criteria listed above.

6.2.3 Cost-Benefit Analysis

The options or alternatives should then be assessed relative to the required and high-value criteria. A formal or informal cost-benefit analysis should be performed to ensure that each of the alternatives is appropriately considered. The methodology for estimating decommissioning costs is given in the DOE Decommissioning Handbook (DOE 1980).

6.2.4 Special Equipment

The volume of waste and the associated cost of decommissioning the waste will be greatly reduced if equipment can be cleaned and disposed of as nonradioactive waste or cleaned and disposed of as nontransuranic waste. Numerous techniques have been developed for decontamination of equipment and materials. Established techniques and the latest technology should be considered in minimizing the quantity of contaminated equipment that requires disposal and the waste generated from the decontamination processes.

6.2.5 Independent Verification

Once a disposition is determined, detailed project planning, staffing, and arrangements for waste transportation and disposal can be initiated. A decommissioning readiness review is recommended before beginning major decommissioning jobs.

At the completion of planned decommissioning activities, independent verification and/or third-party measurements should be performed to ensure that the established criteria are met, especially if unrestricted access is contemplated. While not a regulation or requirement, third-party verification may be prudent as an independent verification of the radiological status of the facility.

6.3 DESIGN FEATURES

There are certain facility design features that will greatly affect the cost of decommissioning and the volume of radioactive waste that will require disposal. The following general guidelines apply.

6.3.1 Building Materials

In general, the design features that aid in contamination control during operation also facilitate decommissioning. The inclusion of all the building materials suggested in this section may be cost prohibitive, but they should be considered if the budget allows. The maintenance procedures that are used during operation are also important in controlling the spread of contamination to clean areas, and therefore facilitate decommissioning.

Less permeable building materials are more easily decontaminated. Any concrete with uncoated surfaces that comes in contact with plutonium solutions or plutonium-contaminated air will require surface removal and disposal as radioactive waste at the end of its life. If there are cracks through which contaminated solutions have penetrated, the entire structure may require disposal as radioactive waste.

Metal surfaces may also require decontamination. In general the more highly polished the surface, the easier it will be to decontaminate. If feasible, all stainless steel that will come into contact with plutonium should be electropolished before being placed into service. If HEPA filtration has failed at any time during facility operation, roofs may require decontamination. Metal roofs are easiest to decontaminate, but even these may contribute to the volume of radioactive waste unless unusual measures are taken to clean them.

Interior surfaces are most capable of being cleaned if they were completely primed and painted prior to the introduction of radioactive material into the facility. If interior surfaces are repainted during operation, their disposal as clean waste is likely to require removal of the paint. However, if the paint has deteriorated, cleaning for unrestricted use may be as difficult as if the material had never been painted. Wood will almost certainly become contaminated as will plasterboard and other such materials.

Floor surfaces are likely to be a problem. Concrete should be well sealed and covered with a protective surface. Single sheet, vinyl flooring with heat-sealed seams is preferable to asphalt or vinyl tile because it is more easily cleaned. If the floor needs resurfacing, it is preferable to overlay new flooring material rather than remove the old material and expose the underlying floor.

Carpets are not recommended because they are difficult to clean and survey. (In some areas, such as control rooms, their use may be justified by noise control requirements; however, their contamination control limitations should be considered.) Small rugs are often helpful in picking 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.)

6.3.2 Ventilation Systems

In addition to decommissioning considerations, the design of the ventilation system will depend on the operations that will be conducted in the facility. Adequate air flow for all operations and good design practices will help keep the facility clean during operations and will facilitate decommissioning. Fiberglass duct work may present a fire hazard and may be more difficult to decontaminate than stainless steel, especially stainless steel that has been electropolished.

6.3.3 Piping Systems

Potentially contaminated piping systems that are imbedded in concrete are a common and relatively expensive decommissioning problem. Most often they must be sealed and removed last after all other radioactive material has been removed and the building is being demolished by conventional methods. Often they provide the major impetus for demolishing a building rather than converting it to some non-nuclear use. For this reason it is best to run pipes in chases or tunnels that have been lined (usually with stainless steel) to prevent contamination from penetrating building surfaces. To minimize hand jackhammer work required during decommissioning, floor drains should not be enclosed in concrete.

6.3.4 Soil-Contamination Considerations

Depending on the activity levels found, locations where contaminated effluents have penetrated the ground may require excavation during decommissioning. Because there is no current guidance on acceptable ground contamination levels, the levels will probably be determined on a case-by-case basis. The facility design should minimize such areas. Particular attention should be paid to storm runoff from roofs, storage areas, contaminated equipment storage, and liquid waste treatment impoundments (including sanitary sewage systems if they may receive some small amount of contamination during the life of the facility.)

6.3.5 Other Features

Installed decontamination and materials-handling equipment that facilitates operation and maintenance generally facilitates decommissioning in two ways. First, it can be used for its intended purposes of cleaning and moving equipment during the decommissioning phase. Even more important, it usually contributes to a cleaner, better maintained facility, where nonfunctional equipment is moved out when it is no longer needed and work surfaces are kept free of spreadable contamination.

Additional guidance can be found in Chapter I of DOE 6430.1, General Design Criteria Manual (DOE 1983), for design features to facilitate decontamination and decommissioning.

6.4 PAST EXPERIENCE IN DECONTAMINATION AND DECOMMISSIONING

Considerable experience has been gained in D&D of commercial plutonium facilities as discussed in Hoovler, Myers, and Caldwell (1986), Denero et al. (1984) and Adams et al. (1982). Hoovler, Myers, and Caldwell (1986) discusses the decommissioning programs carried out at two Babcock and Wilcox's buildings in Lynchburg, Virginia, that housed plutonium/uranium fuel development laboratories. Included are information on decommissioning and quality assurance plans, conducting D&D work, performing radiological surveys before and after D&D work, and disposing of the waste. Denero et al. (1984) discusses the D&D of the Westinghouse Nuclear Fuel Facility at Cheswick, Pennsylvania. Described are the facility and its operations, non-destructive assay techniques, equipment required for dismantling and packaging the waste, and management of the transuranic waste. Adams et al. (1982) discusses the complete D&D of the Westinghouse Advanced Reactors Division Fuel Laboratories at the Cheswick, Pennsylvania site. The report discusses the D&D plans, the Environmental Assessment written for the operation, the quality assurance plan, and the health physics, fire control, and site emergency manuals written for the operation.

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