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A GUIDE TO GOOD PRACTICES
AT PLUTONIUM FACILITIES

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A GUIDE TO GOOD PRACTICES AT PLUTONIUM FACILITIES

FOREWORD

The manufacture and use of plutonium means many things to many people and is a complex subject. Plutonium itself is 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, both good and bad, pro and con; it is feared by some and respected by all. Some have (incorrectly) labeled it the most toxic element known. Its toxicity and, more important, its radiotoxicity are the reasons this manual of good practice has been compiled.

The manual establishes guidelines and principles for use in setting up a sound radiation protection program for work with plutonium. The guidance presented is based on the experiences of Energy Research and Development Administration (ERDA) contractors and those portions of private industry concerned with the operation of plutonium facilities, specifically with the fabrication of mixed oxide reactor fuel. The manual is directed primarily to those facilities which have as their sole purpose the handling of large quantities of plutonium for military or industrial uses. It is not intended for use by facilities 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 addresses good practices that should be observed by management, staff and designers, since the benefits of a good radiation protection program are the result of their joint efforts. The establishment of this joint effort cannot be stressed enough and should occur as early in the design stage as possible. No amount of good management can correct for basic problems, including safety-related problems caused by poor

design; nor can 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.

The practices presented in this manual are not mandatory; several alternatives may be available to accomplish a specific task safely, and any one of them may be as acceptable as that presented here. However, those who lack experience in health physics and radiation protection should find the prescribed practices useful.

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PROPERTIES OF PLUTONIUM

SECTION 1
PROPERTIES OF PLUTONIUM

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SECTION 1

PROPERTIES OF PLUTONIUM

Trace amounts of plutonium occurring naturally in uranium ores result from the capture in ^{238}U of neutrons produced by: 1) spontaneous fission, 2) alpha-neutron reactions in light elements and 3) cosmic rays. Plutonium in nature was also produced by a natural reactor that existed near Oklo in Gabon over a billion years ago. Plutonium was first synthesized by Glen Seaborg, Arthur Wahl and Joseph Kennedy, who bombarded uranium with 16-MeV deuterons at the University of California's 60-inch (152-cm) cyclotron. Plutonium-238 was the first isotope to be identified, in 1941; plutonium-239 was identified the same year.

1.1 NUCLEAR PROPERTIES

The single most important nuclear property of plutonium that makes it useful to man is its fissility. Under neutron bombardment a plutonium nucleus fissions and releases about 200 MeV of energy and between 2 and 3 neutrons. This tremendous release of energy makes plutonium useful for nuclear weapons and for reactor fuel.

All forms of plutonium are highly radioactive. Fifteen isotopes of plutonium, ^{232}Pu to ^{246}Pu , have been identified. Isotopes with even mass numbers (except mass number 246) are primarily alpha emitters. Isotopes of mass number 232, 233, 234, 235 and 237 decay by electron capture; isotopes of 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⁽¹⁾ and in ANSI Standard N-317, "Performance Criteria for Instrumentation Used for Implant Plutonium Monitoring."⁽¹⁾

1.1.1 Decay Schemes

Table 1.1 shows the decay mode of the most important plutonium isotopes. Most of these isotopes are alpha emitters that have very little gamma radiation associated with decay. In some instances the decay daughters become significant, especially for isotopes with short half lives. Where more than 1 cm of lead or steel shielding is present, the daughters of ^{236}Pu may be extremely significant, especially ^{208}Tl which emits a highly penetrating gamma ray of 2.615 MeV. In high-exposure plutonium containing a few weight percent ^{241}Pu , the ^{241}Am daughter is important because it emits a large number of 60-keV photons. For brevity, only the most abundant radiations have been included in this table.

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 Table 4.6 for pure plutonium and plutonium compounds. The approximate neutron yield from a substance with a known isotopic composition can be determined by adding up the contributions from each component. This procedure is described in detail in the section that discusses neutron dose equivalent rates (pp 4-44 ff.).

The energy spectra of neutrons emitted by plutonium metal and plutonium compounds are given in Figure 1.1. Pure 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: plutonium tetrafluoride 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 plutonium dioxide emits neutrons with an average energy of slightly more than 2 MeV.

TABLE 1.1. Radioactive Decay Properties of Selected Plutonium Isotopes and Daughters Excluding Spontaneous Fission

Isotope	Half Life	Particle	Mode of Decay		X-Ray		Gamma Ray	
			Energy, MeV	Yield, %	Energy, MeV	Yield, %	Energy, MeV	Yield, %
^{236}Pu	2.85 y ^(a)	α	5.77	69	L's 0.011-0.021	10 ^(b)	0.047	3.1×10^{-2}
		α	5.72	31			0.110	1.2×10^{-2}
							0.165	6.6×10^{-4}
^{238}Pu	87.7 y	α	5.49	72	L's 0.011-0.021	10.5	0.473-0.645 ^(c)	5.2×10^{-4}
		α	5.45	28			0.0435	3.9×10^{-2}
		α	5.35	0.13			0.0999	7.4×10^{-3}
							0.1528	1.0×10^{-3}
^{239}Pu	2.44×10^4 y	α	5.147	73.3	L's 0.0116-0.0215	4.8	0.201-1.085	3.8×10^{-5}
		α	5.134	15.1			0.0387	5.9×10^{-3}
		α	5.095	11.5			0.0516	2.1×10^{-2}
							0.129	6.2×10^{-3}
							0.375	1.58×10^{-3}
^{240}Pu	6.6×10^3 y	α	5.159	75	L's 0.0115-0.0215	10.8	0.414	1.51×10^{-3}
		α	5.115	25			0.142-1.057	4.93×10^{-3}
							0.0452	4.5×10^{-2}
							0.104	7.0×10^{-3}
^{241}Pu	15 y	β	0.0222	99.99 +			0.160	4.2×10^{-4}
		α	4.896	83 of $2.3 \times 10^{-3}\%$			0.0442	4.5×10^{-6}
		α	4.853	12 of $2.3 \times 10^{-3}\%$			0.0443	1.1×10^{-6}
		α	4.797-5.055	5 of $2.3 \times 10^{-3}\%$			0.0663	3.7×10^{-6}
							0.077	2.41×10^{-5}
							0.1037	1.04×10^{-4}
							0.121	8.9×10^{-7}
							0.148	1.9×10^{-4}
^{242}Pu	3.79×10^5 y	α	4.903	76	L's 0.0116-0.0215	4.1	0.160	6.45×10^{-6}
		α	4.863	24			0.0447	5×10^{-2}
^{241}Am	458 y	α	5.486	86	L's 0.0119-0.0222	37.6	0.0434	9.1×10^{-2}
		α	5.443	12.7				
		α	5.389	1.3			0.0595	35.9
							0.0699	2×10^{-3}
							0.099	2×10^{-2}
							0.103	1.95×10^{-2}
							0.123	1×10^{-3}
							0.125	3.95×10^{-3}
							0.146	4.6×10^{-4}
							0.150	7.2×10^{-5}
							0.159-0.2927	1.17×10^{-3}
^{237}U	6.75 d	β	0.248	96	L's 0.0119-0.0208	62.9	0.301-0.4863	1.39×10^{-3}
							0.514-0.863	7.2×10^{-4}
							0.0264	2.4
							0.0332	1.9×10^{-2}
							0.0434	7.4×10^{-2}
							0.051	0.22
							0.0595	36
							0.208	23.3
							0.0648-0.371	6.3

(a) y = years; d = days

(b) Total for all x-rays

(c) Gamma rays of many energies are in these groups. Yield is for the gammas included in the range exclusive of those listed separately.

FIGURE 1.1. 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 argon the metal rapidly oxidizes producing a mixture of oxides and hydrides. With this coating the metal is highly pyrophoric, so pure metal is usually handled in an inert, dry atmosphere of nitrogen or argon.

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 1.2. 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 19.2 g/cm³. Large changes in volume can occur from heating caused by high specific activity or machining operations. For this reason many forms of metallic plutonium are stabilized with gallium alloys in the δ phase, which has a density of about 15.9 g/cm³.

TABLE 1.2. Allotropic Forms of Plutonium Metal

<u>Phase</u>	<u>Stability Range, °C</u>	<u>Density, g/cm³(a)</u>
α	Stable below 115	19.86
β	~115 to 200	17.70
γ	~200 to 310	17.14
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.

At room temperature, the most stable oxide is plutonium dioxide. Loose PuO₂ powder, as formed by calcination, has a density of about 2 g/cm³. If the oxide is pressed and sintered into pellets, it has a density of 10.3 to 11.0 g/cm³.

The chemistry of plutonium is very 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(II) is oxidized into Pu(IV), which is the most stable state. The compounds PuF_4 , $\text{Pu}(\text{IO}_3)_4$, $\text{Pu}(\text{OH})_4$ 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.⁽²⁾

The most important compounds in the processing of plutonium are PuF_4 , $\text{Pu}(\text{NO}_3)_4$ and PuO_2 . The presence of plutonium fluoride is an important intermediate step in most conversion processes. 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. Plutonium nitrate is a convenient compound to use in processing, since it is readily soluble in aqueous solutions. Because of its chemical stability, plutonium dioxide is the preferred form for shipping and storing plutonium.

Plutonium has been used in reactor fuels in the form of PuAl 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.

1.3 BIOLOGICAL PROPERTIES

Most of our knowledge of the biological behavior of plutonium comes from animal studies of dogs, rats, rabbits and baboons. Relatively little comes directly from man. Up to the present time, no fatality or serious disease has been directly attributable to plutonium. Several groups of human beings have appreciable amounts of plutonium in their bodies, but few effects have been observed. Among those people who have internal plutonium are 25 Manhattan Project plutonium workers who received accidental exposures of up to 400 nCi over 30 years ago.⁽³⁾ Groups of individuals are also being studied at Los Alamos and several thousand plutonium workers are being studied by the U.S. Transuranium Registry (USTR).^(a) The Registry, operated by the Hanford Environmental Health Foundation, works on a voluntary basis with individuals who authorize the release of medical and health physics data. Permission may also be granted for post-mortem analyses of tissues of interest.

The metabolism of plutonium is very complex. The manner in which plutonium is incorporated into body tissues depends on the chemical and physical form of the plutonium and the mode of entry into the body, as discussed in the following subsections.

1.3.1 Modes of Entry into the Body

The major health effects from plutonium are caused by the radiation dose from alpha particles. The radiation dose delivered to the various organs is determined by the chemical and physical properties of the plutonium and the route of entry into the body. There are four methods by which plutonium may enter the body: 1) absorption through the skin; 2) absorption through wounds; 3) ingestion of contamination and subsequent absorption from the gastrointestinal tract; and 4) inhalation and deposition in the respiratory tract. Inhalation of plutonium aerosols is perhaps the most serious health hazard.

(a) P. O. Box 100, Richland, WA 99352.

Absorption Through the Skin

Relatively small amounts of plutonium are absorbed through the skin into the body. Animal experiments and studies of human contamination show that skin is an effective barrier against the entry of plutonium. W. J. Bair⁽⁴⁾ reports that plutonium in acid solutions can enter through the skin, as shown in Table 1.3.

TABLE 1.3. Absorption of Pu Through Intact Skin^(a)

<u>Compound</u>	<u>Duration of Exposure</u>	<u>Percent Absorbed</u>
Pu in 9% HCl + EDTA	(b)	0.01
Pu(NO ₃) ₄ in 0.4 N HNO ₃	1 hr.	0.002

(a) Bair, p. 231.

(b) No data.

Absorption from Wounds and Translocation Through the Blood

Little data is available on the absorption of plutonium from wounds. Deposition in a wound may provide a reservoir for continuous absorption into the blood unless the plutonium is excised.

Once plutonium enters the bloodstream, it binds to the blood protein, which is transported to three main organs: the liver, the bone and the kidney. Plutonium is eliminated from the body by the liver and kidney but is firmly retained by the bones. Only complexes like diethylenetriamine-pentaacetic acid (DTPA) escape being trapped in bone. After local administration of plutonium into the blood, Lafuma found the final distribution to be 50% remaining in the skeleton, 20% eliminated in urine and 30% eliminated in the feces after passing through the hepatic system.⁽⁵⁾

Absorption from Gastrointestinal Tract

Plutonium is not readily absorbed by the gastrointestinal tract. The absorption of plutonium in rats is given in Table 1.4 from the data of

Bair.⁽⁴⁾ The International Atomic Energy Agency (IAEA) gives the overall average absorption of $\text{Pu}(\text{NO}_3)_4$ through the human gut as 0.003%.⁽⁵⁾

TABLE 1.4. Gastrointestinal Tract Absorption
of Plutonium in Adult Rats^(a)

<u>Pu Compound</u>	<u>Percent of Administered Dose Absorbed</u>
^{238}Pu nitrate	0.03
^{239}Pu nitrate	0.003
^{239}Pu chloride	0.007
^{239}Pu oxide	0.0001

(a) Bair, p. 230.

Absorption and Translocation from the Lung

Airborne plutonium behaves much like other inhaled particles in that the deposition in the lung depends primarily on the physical size of the particles. Plutonium particles are nonuniformly deposited throughout the lung. Studies of inhaled plutonium nitrate in rats and dogs show that immediately after inhalation exposure, plutonium is present in both particulate and nonparticulate forms. Dogs exposed to $^{239}\text{PuO}_2$ show relatively diffuse distribution of particles throughout the entire lung.⁽⁴⁾

From the moment plutonium particles are deposited in the respiratory tract, biological and physical forces are at work to remove them. A fraction of the amount of plutonium deposited in the lung may be dissolved and absorbed into the blood. Some of the remaining plutonium particles may be engulfed by macrophages. The alveolar macrophage appears to be capable of transporting plutonium particles and aggregates from the alveoli to the ciliated epithelium lining the bronchioles. These phagocytic cells containing particles and aggregates can then be removed from

the lung in the mucous blanket which is propelled up the respiratory passage by ciliary action. Plutonium removed from the lung by this route is swallowed and excreted in the feces.

Both soluble and insoluble plutonium not immediately cleared from the lung tend to become further aggregated. A few days after inhalation of plutonium nitrate and other relatively soluble compounds, single tracts in autoradiographs decrease, and after several weeks nearly all of the radioactive material appears to be aggregated.

Particles of plutonium are also transported via lymphatic vessels in the lung and collected in the thoracic lymph nodes. Autoradiographs suggest that radioactive particles transported to lymph nodes are eventually sequestered in the "hot spots" of scar tissue and do not appear to be mobile.

In summary, while the rates of movement may be low, plutonium particles do move by several mechanisms within lung tissue as the lung attempts to expel the particles and other foreign material that may have been inhaled. The migration of particles in the lung partially compensates for the non-uniformity of the radiation exposure caused by the initial diffuse deposition of the particles. The retention half-times for organic complexes of plutonium, plutonium nitrate and plutonium fluoride range from less than 100 days to about 300 days in dogs. The retention half-times for PuO_2 are much longer, ranging from 300 to 1000 days in dogs and 250 to 300 days in human beings. The wide range of values for dogs is largely due to extensive experimentation with a wide variety of plutonium oxides with different particle-size characteristics (larger particles are retained longer).

Translocation from the Lung to Other Tissues

The translocation of plutonium from the lung to other tissues in the body is highly dependent upon the form of the plutonium. Most of the data available are from animal experiments.

In beagles that inhaled relatively soluble plutonium nitrate, the fraction of plutonium remaining in the lungs decreased within several months to 40% of the amount initially deposited in the lower respiratory

tract. About 30% of the plutonium translocated to the bone and about 10% to the liver, and a small percentage was found in the spleen, lymph nodes and other soft tissues. The remainder was excreted in urine and feces.

When plutonium dioxide is inhaled, most of the plutonium ends up in the lymphatic system. Figure 1.2, reproduced from the work of Bair,⁽⁴⁾ shows the rate at which plutonium dioxide is removed from dogs. After 11 years, the amount of plutonium in the lung decreases to about 10% of the original amount. About 40% of the plutonium translocates to the thoracic lymph nodes, about 10% to the liver, about 5% to the bone and about 7% to the abdominal lymph nodes. Small amounts are also found in the spleen and kidney. In general, the translocation of plutonium occurs very slowly after the inhalation of plutonium fluoride and plutonium dioxide.

FIGURE 1.2. Distribution of Plutonium in Dogs After Inhalation of $^{239}\text{PuO}_2$

1.3.2 Deposition in the Gonads

Because of possible genetic effects the amount of plutonium that accumulates in the gonads is of considerable interest. Bair⁽⁴⁾ states

that about 0.05% of the plutonium circulating in the blood is deposited in the testes and only about 0.01% is deposited in the ovaries of experimental animals.

1.3.3 Deposition in the Fetus

The amount of plutonium that may reach the fetus of a pregnant female has also been studied. Experiments with animals show that the placenta is an effective barrier against plutonium and that extremely high levels of contamination would have to occur in the pregnant female before appreciable quantities of plutonium would occur in the fetus. Experiments with rats show that only 0.01 to 0.0004% of the plutonium injected into female rats reaches the fetus.⁽⁴⁾

1.3.4 Biological Effects on Various Organs

The biological effects that may occur depend upon the radiation exposure and the relative sensitivity of each tissue to radiation. The tissues exposed are primarily blood, bone, liver, lung and the lymphatic system. Animal experiments have identified neoplasia as the most sensitive response to the long-term effects of small amounts of plutonium in the body.

Blood

Plutonium is cleared from the blood within a few days after the blood absorbs it from the site of deposition (e.g., the GI tract or lung); the blood cells are, therefore, exposed for only a short time. As a result, most of the hematologic effects observed after a plutonium deposition occur at dose levels higher than those resulting in cancer. Although plutonium can deposit in bone and lymph nodes, the development of leukemia is uncommon in animal experiments.

Bone

Osteogenic sarcomas appear to be the most sensitive response to the skeletal deposition of plutonium. The most informative experiment on the effects of plutonium in bone is a study done on dogs at the University of

Utah.⁽⁶⁾ The experiment was begun in 1952 to compare the long-term effects of injected plutonium and radium. The objective was to determine the toxicity ratio of plutonium and radium in dogs, so that the available data on radium toxicity in humans could be used to infer the toxic effect of plutonium in man. At a plutonium exposure of 0.016 $\mu\text{Ci/kg}$ ingested intravenously, about 20% of the dogs developed osteosarcoma (bone cancer). This experiment is being extended to expose the dogs to 0.0006 $\mu\text{Ci/kg}$, the permissible plutonium body burden for man. Bair estimates the tumor risk for bone to be 5×10^{-4} per nCi of plutonium.⁽⁷⁾

Liver

Liver is comparable to bone and lung in terms of plutonium content and radiation exposure. However, the liver appears to be less radio-sensitive. Bair estimates the risk of liver tumors to be 1.2×10^{-3} per nCi in man.⁽⁷⁾

Lung

The inhalation of relatively soluble compounds of organic complexes or plutonium nitrate results in primary lung cancer in rats, rabbits and dogs. In the formation of both lung and bone tumors, the lowest dose at which a significant effect has been observed is about 30 rads. Bair estimates that the values for the risk of lung cancer resulting from inhaled plutonium range from about 1 to 2×10^{-4} per nCi/g lung·days.⁽⁷⁾

Lymph Nodes

The lymph nodes are not susceptible to cancer induced by plutonium, even though a large fraction of a lung deposition can translocate to the lymph nodes, sometimes in very high concentrations. Data taken 7 to 9 years after dogs inhale PuO_2 indicate that the concentration of plutonium in the lymph nodes is 1400 times that in the lung tissue.⁽⁷⁾

1.3.5 Summary

In summary, the metabolism of plutonium is complex and depends upon the mode of entry into the body and the chemical and physical form of the

plutonium. Inhalation of plutonium aerosols into the lung presents the greatest health hazard. The tissues most sensitive to plutonium are bone and lung and the most common reaction is cancer. A large amount of data has been collected from studies on dogs and rats but little has been collected on human beings. More data from humans will become available as the USTR continues its autopsy program.

1.4 MANUFACTURE AND USES OF PLUTONIUM

In the past most plutonium 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 by-product in the operation of research reactors and nuclear-powered ships.

1.4.1 Future Sources and Uses

Much of the plutonium produced in the future will come from commercial light water reactors (LWRs) used to generate electric power. If recycling is allowed, thousands of kilograms of plutonium per year will be used as a fuel for other LWRs or for fast breeder reactors. Because they produce more plutonium "fuel" than they consume, liquid metal fast breeder reactors (LMFBRs) will probably produce most of the plutonium in the more distant future.

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 are being used as a power source for medical prosthetic devices such as cardiac pacemakers and a prototype artificial heart. Figure 1.3 shows the main modes of production of the plutonium isotopes of interest.

1.4.2 Expected Changes in Isotopic Composition

If planned goals for fuel exposure are met, the plutonium available from commercial LWRs will be considerably different in isotopic composition from production-grade plutonium. For present purposes, the newer reactor grade or "high-exposure" plutonium is defined as any plutonium containing up to 80 wt% ^{239}Pu .

The calculated⁽⁸⁾ isotopic compositions of low-exposure and LWR-produced plutonium for the years 1975, 1980 and 1985, and the decay heats of these materials, are summarized in Table 1.5. These values are

considered typical although they are averages over all LWRs. Moreover, because they are averages, these exact compositions are not likely to be encountered. The values in Table 1.5 are subject to change caused by startup delays, unscheduled outages, fuel performance, more or less frequent recycling, etc., and should therefore be considered primarily illustrative.

TABLE 1.5. Isotopic Composition and Quantity of LWR-Produced Plutonium

Year	Pu Isotope, wt%						Decay Heat, W/kg
	^{236}Pu	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	
Low Exposure	--- (a)	---	93	6	1	---	3
1975	5×10^{-6}	0.9	65.6	21.8	9.0	2.7	9
1980	7.3×10^{-6}	1.5	58.1	24.1	11.4	4.9	11.6
1985	6.5×10^{-6}	1.6	54.7	24.7	12.2	6.7	12.4

(a) None.

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

1.5 REFERENCES

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SECTION 2

SITING OF PLUTONIUM FACILITIES

SECTION 2
SITING OF PLUTONIUM FACILITIES

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SECTION 2

SITING OF PLUTONIUM FACILITIES

Substantial quantities of unsealed plutonium should be processed only in a facility solely devoted to, and specially equipped for, work with highly radiotoxic materials. Such a facility should be located so that normal operations, including anticipated minor contamination spreads, do not adversely affect adjacent buildings. Some factors to be considered are:

- natural site characteristics
- transportation
- decontamination and decommissioning
- environmental, safety and health considerations.

In addition, applicable federal standards must be met.

2.1 APPLICABLE STANDARDS

The general requirements for the site of an ERDA plutonium facility are covered in the appendices of ERDA Manual Chapters (ERDAM) 6202, "Site Selection"⁽¹⁾ and 6203, "Site Development Planning."⁽²⁾ Other references include 10 CFR 100, "Reactor Site Criteria"⁽³⁾ and the Nuclear Regulatory Commission's (NRC's) Regulatory Guide 3.26, "Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants."⁽⁴⁾ The ERDAM appendices outline the methods used for site selection and the responsibilities and authorities of personnel involved. 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 map survey; the remaining few sites are carefully analyzed by a site selection committee using the criteria and information sources listed in ERDAM-6202. After the survey is complete, the site selection committee gives its report to the directors outlined in ERDAM-6201, who make the final site selection.

One important aspect in selecting an ERDA site is coordination with state, regional and local development plans and programs, as required by the Intergovernmental Cooperation Act of 1968. At the earliest practicable stage of site selection, the Office of the Director, ERDA Division of Construction, acts as headquarters for consultation with state officials and agencies, state, regional and metropolitan planning and development clearing houses, and local officials.

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 Impact Statements (EISs), and System Design Descriptions (SDDs). Natural phenomena that should be considered in site selection and plant design include earthquakes, lightning, tornadoes or hurricanes, missiles, flooding, loss of water supply, and snow and ice load.

2.2.1 Meteorology

The wind patterns (speed, direction and stability) at a site must be tabulated. This data is needed to estimate doses to populations downwind from possible releases of radioactive material. Data should also include the frequency and intensity of heavy rain, snow and ice storms, thunderstorms and lightning strikes, which may affect a facility's power supplies and ventilation or other safety features.

Nuclear facilities must be built to withstand a standardized design basis tornado unless it can be demonstrated that such a tornado is not likely to occur (see ERDAM-6301, Appendix, Part II, I, p. 73).⁽⁵⁾ A complete history of the magnitude and frequency of reported tornadoes, hurricanes or other wind storms in the region of the site and of the probability of high dewpoint temperatures should be compiled. Tornado information can also be obtained from Thom, "New Distribution of Extreme Winds in the United States."⁽⁶⁾

2.2.2 Hydrology

Precautions should be taken to avoid flood damage, erosion and water pollution. The water flow of streams, rivers, and reservoirs should be documented, and the maximum precipitation and water levels that might adversely affect plant safety or the storage of radioactive waste should be considered. This data can be obtained from the National Oceanic and Atmospheric Administration (NOAA) of the U.S. Department of Commerce.

The history of floods (water levels and peak discharges) in the region of the site should be compiled to help in the design of flood protection. The effects of seismically-induced dam failures on the upper limit of flood controls at the site should also be considered. The U.S. Army Corps of Engineers can supply runoff, water distribution, and "worst probable flood" data, and guidance can be obtained from "Flood Hazard Evaluation Guidelines for Federal Executive Agencies" published by the United States Resources Council.⁽⁷⁾

Finally, the population groups that use water which could be contaminated by plant effluents under both accident and normal conditions must be identified.

2.2.3 Geology and Seismology

Geologic and seismic data for the site of the proposed plutonium facility should be gathered. Earthquake data and maps can be obtained from the National Ocean Survey, NOAA. The geologic conditions underlying all structures, dams, dikes and pipelines should be examined for the possibility of earth movement that could damage the facility. Natural conditions such as caverns or potential landslide areas, and man-made conditions caused by mining or the withdrawal or addition of subsurface fluids, should be absent.

The earthquake history (e.g., magnitude or highest intensity, and epicenter) that could cause a maximum ground acceleration of 0.1 g at the foundations of the proposed plant must be determined. The effects of tectonic structures and active faults that could produce a major earthquake with an epicenter within 200 miles (322 km) from the plant should be estimated. The possible effects of earthquakes from any fault more than 1000 feet (305 m) long within 5 miles (8 km) of the plant should be considered in the plant design.

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

2.3 TRANSPORTATION

Although a major plutonium facility should be isolated from areas with a large population, the facility should be reasonably close to major transportation networks.

As a safeguard, routes should be available for the shipment of plutonium to and from the facility, including alternative routes for emergency evacuation. Changes in packaging and transportation standards may require that the location of future facilities and transportation networks receive more careful scrutiny than they have to date.

2.4 DECONTAMINATION, DECOMMISSIONING AND DISMANTLING

The decontamination, decommissioning and dismantling of a plutonium facility may influence its location. Some facilities could conceivably be so contaminated that the property could never be released to the public or used for other purposes. However, if an entire facility is eventually to be dismantled, it should be located near transportation so that heavy equipment and contaminated structures can be shipped by railroad or barge to a federal depository for disposal.

2.5 ENVIRONMENTAL, SAFETY AND HEALTH CONSIDERATIONS

A plutonium facility should be located where its construction and operation will comply with the provisions of ERDAM-0510, "Prevention, Control and Abatement of Air and Water Pollution,"⁽⁸⁾ and ERDAM-6301, "General Design Criteria,"⁽⁵⁾ and will not have an adverse environmental impact.

Facilities that emit gaseous effluents should be located where favorable wind distributions will minimize the levels of contaminants at site boundaries and in nearby populated areas as directed by ERDAM-0513, "Effluent and Environmental Monitoring and Reporting,"⁽⁹⁾ and ERDAM-0524, "Standards for Radiation Protection."⁽¹⁰⁾ Consideration of wind factors could prevent the serious consequences of any accidental loss of radiation control at the facility.

The disposal or storage of wastes, particularly radioactive wastes, requires careful attention to federal, state, regional and local standards on pollution control. The potential for the release of radioactive materials to surrounding communities and possible resultant effects should be thoroughly evaluated; a hazard analysis should be performed for both the environmental conditions and the work that will be done at the proposed disposal site. Where possible, several facilities that handle radioactive materials should be grouped to allow the use of common effluent treatment systems and facilities.

In locating waste storage or disposal areas, precautions should be taken to prevent liquid or solid wastes from adversely affecting surface or ground waters, both under normal conditions and as a result of flood runoffs or accidental spills. In many states, waste containing more than 10 nCi of plutonium per gram of waste must be shipped out of the state to a federal depository for disposal. All radioactive wastes that leave a plant must be in solid form.

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FACILITY DESIGN

SECTION 3

SECTION 3
FACILITY DESIGN

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SECTION 3

FACILITY DESIGN

The philosophy behind the design criteria of the Atomic Energy Commission (AEC) and ERDA over the years is contained in a paper entitled "Safety Criteria for the Design of Facilities Processing Plutonium."⁽¹⁾ The earliest plutonium facilities were built during World War II and were constructed simply because of wartime shortages. In 1969 a major fire occurred at the Rocky Flats plutonium facility, which had been constructed in accordance with industrial building codes. Although no plutonium was released as a result of this fire, the potential for such a release was Not true apparent, as were the potentials for large financial losses and the loss of program continuity. Since the Rocky Flats fire, more stringent requirements for plutonium facilities have been instituted. These are included in the appendix of ERDAM-6301, "General Design Criteria, Plutonium Facilities."⁽²⁾

Radiation protection in plutonium facilities is usually achieved by a mixture of administrative and physical safeguards. A building equipped with a maximum of physical safety features and a minimum of administrative controls should be more economical to operate than one with the reverse characteristics. In designing a new facility, all of the necessary physical features can be included; in an old facility, however, it may be physically or economically impossible to meet all of the requirements.

The criteria presented in this section relate to physical safeguards only; criteria necessary for administrative control are not included. The term "physical safeguards" as used here refers to contamination control rather than to control against theft (as this term would imply with respect to nuclear material management). In addition to the radiation protection requirements, facilities containing more than one-third of a minimum critical mass of fissile material are subject to criticality safety requirements, which include the need for a criticality detection system and criticality dosimeters.

This manual does not attempt to reference all codes and standards that apply to the construction or modification of facilities, except those that relate directly to personnel radiation protection. Designers of plutonium facilities should nevertheless find the material presented here valuable.

3.1 APPLICABLE STANDARDS AND GUIDES

The design criteria contained in the appendix of ERDAM-6301, "General Design Criteria, Plutonium Facilities,"⁽²⁾ are mandatory for all new plutonium facilities containing substantial amounts of plutonium. A facility that will handle more than 2 kg of plutonium must also meet the security requirements of ERDAM-2405, "Physical Protection of Unclassified Special Nuclear Material."⁽³⁾ Other sources of information are NRC's Regulatory Guides for Fuels and Materials Facilities,⁽⁴⁾ and the IAEA booklet, The Safe Handling of Plutonium.⁽⁵⁾ The first of these two guides covers many of the topics handled in the appendix of ERDAM-6301.

3.2 DESIGN OBJECTIVES

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

The specific facility design chosen depends on the quantity and form of the plutonium that will be used. When encapsulated in a double, hermetically sealed container, up to 220 grams of plutonium can be used in any radiological facility, with no special physical safeguards applied except those necessary to assure the integrity of the container. If more than 220 grams is to be permitted in the facility, the criticality safety criteria must be met.

Some simple processes involving very small quantities of unsealed plutonium can be carried out safely in well-designed and adequately filtered open-face hoods in a general radiochemistry facility. The specific quantity that can be so handled depends on the complexity of the process and the specific form of the material; however, as a general rule, if the quantity of plutonium is 100 mg or more, the process must be carried out in a plutonium facility. The converse is not necessarily true. Any use of less than 100 mg of unsealed plutonium must be reviewed by the facility's safety personnel and the feasibility of the proposed use will be established based on the form of the material, the work to be performed, and the physical and administrative controls to be employed.

The application of these guidelines to specific proposals for the modification of existing facilities or the construction of new facilities requires that judgments be made based on detailed information about the facility, its use and its potential impact on surrounding facilities and the public. For some facility modifications, the physical criteria outlined here may be modified or reduced if administrative requirements are increased. A cost/benefit analysis may be necessary to make this decision.

To meet the design objectives just discussed, the primary goal is to keep plutonium confined in its intended place (i.e., its capsule, hood, glove box, etc.), both through normal confinement procedures and under accident conditions. The ultimate decontamination and decommissioning of a facility must also be considered in facility design.

3.2.1 Confinement

The confinement system is a series of physical barriers that, together with a ventilation system, minimize the release of radioactive material to the environment under normal and abnormal conditions. Plutonium must be separated from the ambient environment by at least two barriers and from an operator by at least one barrier.

Primary Confinement

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, 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 some combination of the two. For example, process equipment that is not sealed but contains plutonium material in process must be enclosed in glove boxes or other confinement barriers. Fuel rod cladding, bags and other sealed containers can be considered primary confinement.

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 must (with the exception of fuel rods) be operated at a pressure negative to the secondary confinement in which it is located and be exhausted through a ventilating system. The barrier and its accessory equipment should be designed to prevent accidental liquid flooding. All primary confinement piping joints should be tested

for leak tightness. Penetrations in the primary confinement barrier, such as conduits, ports, ducts, pipes and windows, must be protected against the release of radioactive material.

Secondary Confinement

The secondary confinement barrier encloses the room or compartment in which the primary confinement barrier is located, and provides contamination protection for plant personnel outside the area of secondary confinement. Secondary confinement rooms, compartments or cells should be separated from each other by fire doors. Both the barrier walls and the fire doors should be constructed of materials capable of withstanding a design basis accident (DBA). The secondary confinement should be designed for pressures consistent with criteria for the ventilation system.

The Building

The building is the structure enclosing both primary and secondary confinement barriers, as well as offices, change rooms, and other areas that are not expected to become contaminated. It is the final barrier between the outside environment and potential contamination. The building structure or any portion thereof may serve as the secondary confinement barrier provided the requirements for both structure and confinement are met. That portion of the structure housing activities involving radioactive material in dispersible form must be able to withstand DBAs, site-related characteristics, and missiles without a breach of integrity that would result in releases of radioactive material from the structure in excess of maximum permissible amounts.

3.2.2 Design Basis Accident Events

The plant structure, including access openings, must be designed to maintain control over the release of particulate radioactive material under conditions of a DBA.

- The DBA Fire involves the burning of all flammable materials within an area enclosed by a fire-resistant barrier of at

least a 2-hour rating (ASTM-E119-71⁽⁶⁾). The rates of combustion for materials shall be as specified in the Fire Protection Handbook, 13th Edition, National Fire Protection Association.⁽⁷⁾

- The DBA-Explosion is the rupture of a primary confinement barrier with an accompanying energy release equivalent to an internal pressure of 105 psi (7.38 kg/cm²). (This energy release will not only result in a pressure wave, but may also generate missiles within the process area.)
- The DBA-Criticality is an accidental excursion of a heterogeneous liquid-powder mixture with a neutron spike yield of 10^{18} fissions, releasing about 30,000 BTU in less than 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.)
- The DBA-Power Failure is the loss of TOTAL electric power for 60 seconds, and the loss of NORMAL electric power for 48 hours. Total electric power means all sources of electric energy, delivered as well as auxiliary and standby. Normal electric power means the services usually supplied by a utility company.
- The DBA-Water is the result of an Uncontrolled Water Hazard; this occurs when water that is supplied to the plant from a controlled external source is released, through a mishap within the plant, for 30 minutes in a manner which results in loss of a system, subsystem, structure or component important to the integrity of the confinement system. This accident concept includes both the effect of accidental flooding within the plant and the loss of feedwater to any equipment that, without adequate water supply, would prevent the functioning of the confinement system.
- The DBA-Natural Phenomenon is the effect of site-related conditions and includes earthquakes, tornadoes, and floods as described in Section 3.3.1 of this manual.

3.2.3 Decommissioning

All designs should provide for ease of decontamination and the eventual decommissioning of the facility. It is at least desirable, if not mandatory, that total decommissioning be possible, thereby returning the site and perhaps the main structure itself to unrestricted uses. This must be the goal of both private and public owners of the facilities.

3.3 STRUCTURAL CRITERIA

The structure and its associated critical equipment, ventilation, electrical, fire protection and utility systems must be designed to provide confinement of radioactive materials in the event of any DBA that can be postulated for the facility and for accidents caused by natural phenomena.

The structural design, the loading combinations, and the construction of critical safety and fire protection features should also be in accordance with the latest edition of pertinent 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 should be designed to withstand the "standard" or design basis (DB) tornado, i.e.:

- 300 mph (480 km/hr) tangential velocity.
- 60 mph (96 km/hr) transverse velocity (to be appropriately combined with tangential velocity).
- average drop in ambient pressure of 3 psi (0.2 kg/cm^2) in 3 seconds; resulting loads, taking into consideration any pressure equalization from permissible venting, should be considered additive to velocity-induced pressure loads.
- tornado-generated missiles; characteristics of potential missiles should be conservatively determined from a review of onsite sources and possible missiles that could be borne to the site by the tornado. Small, high-velocity missiles (up to maximum wind velocities), and massive, low-velocity missiles should both be considered in terms of penetration or crushing effects.

- tornado-force loads: critical structures, systems, and associated equipment that might be subjected to tornado forces should be designed for the loads induced by such forces (including missile loads) in combination with applicable functional design loads. For critical facilities, design endurance limits for the model tornado should not exceed 90% of yield for reinforcing steel and 75% of ultimate strength for concrete.

The DB tornado should be assumed capable of occurring at any time; however, the simultaneous occurrence of the tornado and a DBA or any other limiting, site-related event such as an earthquake or flood need not be considered for design purposes.

For areas of the facility involving no critical safety functions and no significant potential for radiological hazard, lesser design values may be proposed based upon a thorough review and documentation of historic wind, hurricane, and tornado data; however, the design of these areas should withstand wind velocities greater than the maximum observed at the site (see American National Standards Institute, ANSI Standard A58.1,⁽⁸⁾ latest edition).

3.3.2 Lightning Protection

Lightning protection should be provided for all facilities.

3.3.3 Earthquake Resistance

Seismic parameters should be developed for the site, to determine a design basis earthquake (DBE) and an operating basis earthquake (OBE). The smaller earthquake, the OBE, is equivalent to approximately one-half the DBE in terms of ground acceleration. Critical areas of the structure and associated critical equipment, ventilation, fire protection, and utility systems should be designed to withstand the DBE. Remaining areas of the structure and its associated equipment should be designed to remain operable under the OBE.

Critical items are those structures, systems, and components whose continued integrity and/or operability is essential to preventing an accident and/or mitigating 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).

Determining the Size of an Earthquake

Procedures similar to those in regulatory criteria, "Seismic and Geologic Siting Criteria for Nuclear Power Plants,"⁽⁹⁾ should be used to determine the quantitative design basis for the site's vibrating ground motion caused by earthquakes.

Earthquake Spectrum and Damping Factors

Critical items should be designed using dynamic analysis techniques and the El Centro spectrum or its equivalent.^(a) The ratio of vertical to horizontal acceleration should be two-thirds unless specific site considerations justify the use of a different number. Damping factors selected by a qualified consultant should be used in the design of critical items.

Load Combinations and Stress Limits

All critical structures, systems and components should be designed to withstand:

- load combinations, including normal design loads and OBE loads, within normal working stress or deflection limits.

(a) This may be a smooth spectrum developed from an envelope of spectra related to the measured vibratory motions caused by more than one earthquake.

- load combinations, including normal operating loads and DBE loads, within yield strength. Allowable stresses under DBE loading should not exceed 90% of the yield stress in reinforcing steel or 75% of the ultimate strength in concrete.
- load combinations, including normal operating loads, DBE loads, and applicable DBA loads, without loss of function of the specific structure, system, or component.

3.3.4 Other Natural Phenomena

Design loads and considerations for other natural phenomena (snow, flooding, etc.) should provide a conservative margin of safety for conditions more extreme than the maximum historical levels recorded for the site. Protection against flooding should be based on the maximum probable flood for the area as computed by the U.S. Army Corps of Engineers.

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 contribute to the development of abnormal situations that may affect personnel, the environs, and/or the cost of operating the facility. Normally, three areas are involved in the overall building layout:

- the process area, where plutonium or other contaminated material is used, handled or stored;
- the controlled area, which is normally free of contamination but could potentially become contaminated; and
- the uncontrolled area, which includes all areas where no radioactive materials are permitted and no radiological controls are normally necessary. Offices and lunch rooms are included in this category.

3.4.1 Objectives

To the extent possible, the following objectives should be achieved in the layout of the facility:

- All movements of personnel, material and equipment between the process area and the uncontrolled area should be through a controlled area or air lock. Doors providing direct access to the process area from the uncontrolled area (including the outside of the building) are permissible provided they are used only as emergency exits or infrequently for nonroutine operations. All such doors must have airtight seals; those doors without air locks must also have alarms that sound when doors are open to signal that a breach in the personnel-control system exists.
- Normal administrative traffic should be restricted to the uncontrolled and controlled areas and should not require passage

through the process area. Process traffic should be restricted to process and controlled areas and should not require passage through uncontrolled areas.

- Process areas should be located in a one-story portion of the facility; where multistories are necessary, the process area should be located on the ground level.

3.4.2 Process Area

The plutonium process area is a contiguous group of rooms containing all operations involving plutonium, including processing, shipping, receiving, storage and waste handling. Sufficient space must be available within the process area to accommodate process or programmatic changes.

The initial line of defense to protect workers in a process area is the primary confinement system, which includes enclosures, glove boxes, conveyer lines, the ventilation system and process piping. Various steps in the process may be compartmentalized by the use of discrete glovebox systems located in separate compartments or rooms, to minimize the severity of process accidents. Storage for plutonium, interim- and long-term storage for solid contaminated wastes, and contaminated maintenance compartments should be available.

The main consideration in the design of primary confinement is the minimization of impact on people, facilities and programs in the event of abnormal operations. To accomplish this goal, air is exhausted directly from the enclosures and conveyer lines; initial filtration is provided at the exit from the enclosures; piping, valve stems, drive shafts, etc., are provided with an airtight entrance through a rigid (metal or concrete) portion of each enclosure; and the back flow of radioactive materials into piping is prevented. The type of confinement enclosure used (hood, glove box, remote operational cell) 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 should be considered. However, the applicability of this guideline will vary based on the individual merits of each case.

Structure

The floors of process areas should be designed to bear the heavy loads of shielding casks or other types of heavy shielding that may be required.

Water collection systems should be provided for water runoff from the process area, such as from firefighting activities. In the design of water collection systems, nuclear criticality, confinement, sampling, volume determination, and the retrievability of liquids and solids should be considered.

Adequate provision for the safe collection, packaging, storage and disposal of solid waste containing plutonium must also be provided. This should include provisions for monitoring all packages containing solid radioactive waste and for identifying and determining the quantities of special nuclear material in the waste. The use of fireproof containers is required.

Future decontamination requirements should be considered in building features, hoods, glove boxes, and process equipment. Rough surfaces, square corners, cracks or crevices, and absorbent materials should be avoided. Disposable linings, covers, coatings, or easily decontaminated surfaces should be utilized throughout the process area. Service piping, conduits, and ductwork should be kept at a minimum in the operating areas and arranged to facilitate maintenance and decontamination.

Storage

Storage facilities in the process areas should be designed to prevent the exposure of operating personnel and to satisfy the requirements for security and safeguards in ERDAM-2405, "Physical Protection of Unclassified Special Nuclear Material"⁽³⁾ and in the ERDA Division of

Safety, Standards and Compliance memo of 4 November 1976, "Storage Criteria for Unirradiated Plutonium and Enriched Uranium."

Shielding

Provisions must be made to accommodate the shielding of all items in the process area. Furthermore, all structures (floors, walls, glove boxes, etc.) may require the addition of extra shielding during the life-time of the facility, because of contamination, increased throughput, or higher radiation levels of the material being processed.

ERDAM-6301⁽²⁾ gives a radiation level of 1 rem/yr as a design guide for the application of shielding, but does not specify how this value is to be used. A value of 500 mrem/yr for routinely occupied areas, with higher values for other areas, seems more appropriate. This approach would allow considerably higher exposure rates in those areas where occupancy factors are low enough to warrant them, and would also provide some margin of safety or operational flexibility that is often needed over time, usually because of increasing throughput. The design of a routinely occupied portion of a process area should never include anticipated dose rates in excess of 100 mrem/hr.

In future, the radiation exposure limits for personnel may require special structural and spatial provisions. Structures should be designed in conjunction with process equipment and associated confinement devices, so that normal operation and maintenance can be performed quickly and efficiently and radiation exposure can be minimized. To this end, the design should utilize adequate amounts of fixed and movable shielding and provisions should be made to accommodate remote-handling devices. Shielding materials should be fire-retardant or fireproof.

Enclosures

Glove boxes, conveyors and other enclosures should be designed to guarantee integrity during normal operations and accident conditions. Noncombustible or fire- and corrosion-resistant materials are important in the construction of the confinement system, including any shielding

employed. Fixed modular construction should be employed wherever possible, utilizing a standardized attachment system that will permit replacement or relocation within the glovebox system with a minimum spread of contamination. The design should incorporate features that encourage efficient operation (good external lighting, accessibility to all parts, etc.) while minimizing contamination control problems (smooth interior and exterior surfaces, round corners, minimal protuberances). The enclosures should be equipped with or capable of accepting any necessary neutron and/or gamma shielding.

Fire Protection

Fire protection is required in the glovebox, enclosure and conveyor systems in order to meet ERDA standards. Automatic fire suppression must be considered when a credible fire could produce a loss (including the cost of decontamination) in excess of \$100,000. When the potential loss might exceed \$1 million, an automatic fire suppression system is mandatory.

Fire breaks should be employed between modules and conveyors and should be designed so that they are normally closed. For systems in which fire breaks 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 in order to protect against loss from a fire originating within the glovebox, enclosure, and/or conveyor system. In lieu of such a fire sensing system, an oxygen deficient atmosphere may be provided as the normal or required operating atmosphere within the glovebox, enclosure, or conveyor system, and fire sensors should be installed along with provisions for manual fire suppression.

Ventilation

A dynamic ventilation system shall be installed on all glovebox, enclosure and conveyor systems to maintain a negative internal pressure with respect to the room. An automatic detection and actuation system should be installed to assure adequate inflow of air through any potential breach in the glovebox, enclosure, and/or conveyor system. Small glove-

box systems with positive-pressure supplied gases require positive-acting pressure-release valves (connected into the exhaust system) to prevent pressurization of the system. Recirculating systems should include detection and cleanup equipment for noxious chemicals and radioactivity. Pipes and tanks used for plutonium storage or transfer should be in protective enclosures to reduce contamination/decontamination problems within the process area.

3.4.3 Controlled Area

All support facilities with a potential for periodic low-level contamination should be located in the controlled area. These facilities include change rooms and decontamination rooms for personnel; health physics labs; facilities for the receipt, temporary storage and shipment of radioactive or potentially contaminated materials; maintenance rooms for regulated equipment; mechanical equipment rooms; and some laboratory facilities.

Change Room

Change rooms are required for both men and women, with locker capabilities to support the anticipated number of radiation workers. These change rooms should include facilities for storing and dispensing clean protective clothing; a well-defined area near the exit from the controlled area, for the temporary storage of potentially contaminated clothing; and adequate shower capabilities. The clean side of the change room should be easily separable from the potentially contaminated side of the room.

Personnel Decontamination Room

A personnel decontamination room or station is required for any plutonium facility. It should be located near or in the change rooms, perhaps between the two change rooms. An adequately equipped station should have a telephone; a work bench with a cabinet for decontamination supplies; an examination chair; and a sink and shower, both connected to either a holding-sampling tank or a monitored sewer system. The room should also contain equipment for nasal irrigation and equipment for analyzing nasal smears.

Health Physics Lab-Office

To function efficiently in a plutonium facility, health physics personnel should be assigned lab-office space at or near the exit from the process area into the controlled area. As a rule of thumb for determining space needs, two monitors are needed for the first 30 radiation workers, with one additional monitor for each additional 30 radiation workers. Space should be included for the readout of radiation protection instrumentation, and for counting equipment, portable instruments, and mask-fit test equipment.

Mechanical Equipment

Where possible, mechanical equipment (motors, pumps, valves, etc., that may be a source of contamination) should be located in the controlled zone. Enclosures that will contain the contamination or will be easy to decontaminate should be placed around the equipment.

3.5 SERVICE AND UTILITY SYSTEMS

The design of utility services must provide reliability consistent with 1) operational requirements for the control and confinement of radioactive materials and 2) potential hazards under all probable conditions. Those services and utilities important to the continuity of essential plant functions (e.g., ventilation systems, criticality alarms, etc.) must be designed to the same integrity level as that of the function they serve. Some service or utility systems are connected to other systems or structures that are essential in preventing the release of radioactive materials; such service or utility systems must be designed so that if they fail connecting systems will not be damaged.

3.5.1 Ventilation and Exhaust

The ventilation and exhaust system includes ductwork; fans; air-cleaning, tempering or humidity control devices; and associated monitoring, instrumentation and controls required to confine radioactive material within the ventilation system and to remove radioactive material from air streams released from the plant. The design of ventilation system components does not include process vessels, primary containment housing, nor the building structure.

Design Objectives

The principal design objectives of the ventilation and exhaust system are to preclude the spread of radioactive contamination within the facility, and to maintain at the lowest practicable levels the release of contamination from the facility under all credible accident conditions. The building's structure and exhaust system should be designed to exhaust all gaseous effluents through the filtered exhaust system, even though this might mean the loss of the desirable negative pressure differential in the building. Effluent concentrations measured at the point of discharge (e.g., exhaust ducts and stacks) should normally be far below the maximum permissible concentration for uncontrolled areas, as given in ERDAM-0524, "Standards for Radiation Protection,"⁽¹⁰⁾ for soluble plutonium.

Air Flow and Balance

A dynamic air-flow pattern should be used to insure that air flows from the environment into the building and, within the building, from non-contaminated areas to potentially contaminated areas to those areas normally containing contamination. These patterns of air flow should be determined by maintaining pressure differentials between the various confinement zones in the building, and between the building and the environment.

A minimum of two negative-pressure zones should exist within the process building: the first, in the process confinement system, should serve the spaces within the dry boxes, conveyors, and 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 contiguous to process areas and potentially free of contamination will constitute a third zone.

A minimum pressure differential between 0.75 and 1.0 in. (1.9 and 2.5 cm) WG, negative with respect to the room, should be maintained in all process confinement systems. A negative pressure differential between 0.05 and 0.1 in. (1.3 and 2.5 cm) WG should be maintained between process and controlled areas, and between controlled and uncontrolled areas. Air locks between zones should be used where necessary to insure the maintenance of proper differentials.

Provision should be made to minimize reversals of the air flow. Penetrations of the building for ventilation ducts should be kept to a minimum and should be designed to protect the critical systems against a DBA. If the barrier around the process area is the building's outside wall, no penetrations should be permitted.

Room air in controlled and process areas may be recirculated if the recirculating air system is provided with two high-efficiency particulate air (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 should be located between the two filters to trip the system to single pass when air contamination reaches a pre-set point.

Exhaust air from the process confinement system may be recirculated, but should not be released to or mixed with room air in the controlled and process areas. An operating principle with respect to air recirculation is that air is never directed to a zone of lower potential contamination than that from which it was withdrawn.

Air Supply

Air supplied to areas that are normally occupied should be controlled and tempered to provide a room temperature and humidity commensurate with levels recommended for comfort.

The ventilation rate in process areas, where uncontained radioactive materials are handled, should be a minimum of 10 air changes per hour. A minimum of 7 air changes per hour should be provided in support facilities within the process area. Adequate air filters should be provided at the intake of the ventilation supply system to minimize dust in the process areas and to reduce dust loading on HEPA exhaust filters.

A downward flow pattern should be provided at operating stations, to direct air from any potential leak point down and away from the operator's face. The distribution of inlet air through a number of small ports or by slot-type distributors decreases the possible occurrence of "dead spots" with little air circulation.

Open-faced hoods may have an adjustable front opening. The linear air velocity through the hood opening should be a minimum of 125 ft/min (38 m/min). The front of the hood should be designed to allow a smooth, eddy-free air flow and to protect against back drafts caused by the movement of workers and objects in the room.

Glove boxes, conveyors, and other systems requiring a controlled atmosphere should be equipped for recirculation of the atmosphere. All parts of the recirculation system should operate at pressures negative

with respect to the room. Process enclosures that use normal air may receive their air supply from the room through dust-stop and HEPA filters mounted on the glove box.

All primary confinement tanks should vent into the primary confinement ventilation system. Vessel vent lines should be designed to prevent the inadvertent transfer of solutions or slurries into ventilation systems.

Consideration should be given to isolating process rooms from each other during accidents. The principle of compartmentalization and separation should be extended to systems handling ventilation in working areas, by the most practicable use of individual systems. Emergency back-up should be provided through combinations of manifolding and damper cutovers between adjacent individual systems; shutdown possibilities should exist in nonemergency areas. Redundancy can be minimized by the provision of back-up units for each pair of individual systems.

Exhaust Systems

Each exhaust filter system for room and process air should consist of a minimum of two HEPA filters for room air and three HEPA filters in series for glovebox or hood exhaust. A baffle or other means of distributing incoming air more or less uniformly over the entire face of the filter bank should be provided. The filter system should be designed to assure convenient and reliable dioctyl-phthalate (DOP) testing of the HEPA filters, as well as convenient filter replacement. Redundancy of systems to provide for filter changes and emergencies should be considered.

The exhaust system for a glove box or hood should be separate from the exhaust system for room air. Exhaust air should be channeled through a HEPA filter at the glove box or hood to maintain primary control at that point. The second HEPA filter for exhaust air should be located several feet down the exhaust duct rather than directly on the box or enclosure, to permit DOP testing of the filter.

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

Exhaust capacity should be available on demand for appropriate enclosures, glove boxes or hoods. This capacity should cause the prompt inflow of air, at a velocity of at least 125 ft/min (38 m/min), through a ruptured glove port.

Integral fire-suppression equipment should be provided as needed within each ventilation system to assure that a DBA-fire could not degrade the integrity of the high-efficiency air-cleaning systems. 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 and perhaps between the last and next to last filters. This cooling spray should be automatically actuated by appropriate temperature- and smoke-sensing devices, as determined by the accident analysis.

All potentially contaminated air should be exhausted through a common stack. Filters, scrubbers, demisters and other air-cleaning devices must be provided to remove toxic or noxious gases from the exhaust prior to its release to the environment. The ventilation exhaust stack should be located as far from an air intake as possible.

System Testing and Control

The ventilation system is essential to the safety of plant employees and the public. It must continue to operate under adverse as well as normal conditions, and must therefore be able to operate effectively for reasonably long periods without requiring shutdown for servicing or filter replacements. The system's effectiveness must be assessable during operation by means of installed testing and measurement devices, including installed spares for critical components.

Air-cleaning systems must be designed for the convenient, repetitive and reliable in-place testing of each stage of the system in accordance with ANSI Standard N101.1⁽¹¹⁾ or a similar test of equal reliability. Provisions for in-place testing should include aerosol injection ports, sampling ports, and connecting and bypass ductwork. Each filter bank must be DOP-tested upon installation and annually thereafter. The filter or filter bank must be 99.95% efficient to a cold DOP test using 0.3-micron particles.

Those portions of the ventilation system essential to preventing releases of highly radiotoxic materials must continue to function (or automatically change to a safe failure mode) in the event of severe natural phenomena or the occurrence of a DBA. The air supply fans should produce a maximum exhaust rate greater than the maximum supply rate. In the event of a system failure, exhaust control dampers must fail to the open position, while supply control dampers must fail to their **preset** closed position. Ventilation and exhaust fans must be provided with emergency power. Supply fans should automatically cut out when exhaust fan capacity in service is not sufficient to maintain the proper pressure differential. Alarms should be provided to signal the loss of fans or an improper air balance. System components or devices that must function under emergency conditions (such as loss of normal electrical power) should be able to be tested periodically, preferably without any interruption of operations.

Appropriate surveillance instrumentation and manual controls over system operations should be provided at a common location.

3.5.2 Electrical Power

Both normal and emergency power supplies must be available to a plutonium facility to assure that critical systems run uninterrupted.

Normal Power

The plutonium facility's electrical power needs should be met by two primary feeders: the preferred primary feeder provides basic service to

the building, while an alternate primary feeder provides back-up to the preferred primary feeder.

The preferred primary feeder should consist of a radial feeder connected directly to the main substation serving the area. In order to minimize power outages, this feeder should be an express feeder and should not have any other loads connected to it.

The alternate primary feeder should be in a standby condition, ready for use by automatic transfer in the event of a forced outage or of planned maintenance of the preferred primary feeder. The alternate feeder, like the preferred feeder, should be radial and should not have any other loads connected to it. It should be connected directly to a substation that is electrically independent of the substation serving the preferred primary feeder. To minimize simultaneous outages of the preferred and alternate primary feeders due to lightning or other physical damage, the two feeders should have maximum physical separation.

Emergency Power

The facility should be provided with a reliable, local source of emergency power for use if both primary sources fail. The emergency power source should be generated onsite by turbines or diesel generators with automatic starting and switch-over equipment. The system, including conductors and conduits, should be independent of and physically separated from the normal power system except at the automatic transfer switch, so that any electrical or mechanical breakdown of the normal power system would 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, and preferably longer, after the restoration of primary power. The emergency sources should have sufficient capacity and sufficient fuel supplies stored onsite to maintain the integrity of all of the critical building systems for 43 hours. It should be able to carry selected loads such as air exhaust systems, air supply

systems, fire protection and detection, all related instrumentation and control functions, certain processing equipment, and any other essential building system.

Noncritical uses of emergency power should be avoided. For example, battery-powered lighting units can easily provide necessary lighting during emergency power periods.

3.5.3 Water Supply

Water storage tanks with multiple or back-up supplies should be provided to meet simultaneously the needs of fire protection, process and potable uses.

Potable water should be distributed to drinking fountains, eye wash fountains, showers, emergency showers, lavatories, toilets, and non-contaminated laboratories. To protect the potable water system against contamination, the 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. Back-flow preventers should be considered to protect public water supplies.

Process water supplied to the process and controlled areas must be isolated from the potable water system. An air gap together with tanks and pumps is the preferred method of isolation; however, if this method is not feasible, approved back-flow preventers may be used.

Water supply systems and process systems that are known to be contaminated must not be connected.

3.5.4 Fire Protection

The plant should be designed to prevent, detect, suppress, and confine fires and products of combustion.

Each zone in the plant building must be equipped with the fire detection devices best suited for that zone, as described in National

Fire Codes, Titles 71 and 72A-72D.⁽¹²⁾ All equipment used must be approved by a recognized testing laboratory. The spacing, sensitivity and location of the detectors should be given careful consideration to assure rapid response.

All areas of the plant must have automatic suppression systems; these systems and the facility's structural design must meet, as a minimum, the requirements of the National Fire Codes, Title 801, "Facilities Handling Radioactive Materials;"⁽¹³⁾ the draft ANSI standard under development by N46-8; ERDAM Chapters 0552,⁽¹⁴⁾ 6301,⁽²⁾ and 0524;⁽¹⁰⁾ and ERDAM-RL-0505.⁽¹⁵⁾ The minimum requirements include the provision of complete automatic sprinkler or equivalent coverage throughout the facility and of fire control measures for special hazards where such hazards exist. An assured water supply, adequate for firefighting and fire-suppression needs over a 4-hour period, must be provided. Capacity and reliability requirements must meet applicable Factory Insurance Associated (FIA), Factory Mutual Research Corporation (FMRC) and NFPA standards.

All fire detectors or automatic extinguishing systems must be connected to the fire alarm annunciators, which should be sufficiently subdivided to give the location of the fire.

Fuels and combustible materials should be stored at a central facility remote from the plutonium processing building. Piped natural gas should not be provided to plutonium process or storage areas; separate bottled gas systems may be provided where required.

The facility's ventilation system should be designed to withstand any credible fire or explosion. It should be constructed of fire-resistant materials and have fire-detection and fire-suppression equipment, including heat and smoke detectors, alarms, fire doors and dampers, and heat removal equipment. The final filter bank of the building's exhaust system should be protected from damage by hot gases, burning debris or fire-suppression agents that may be carried through the exhaust duct during a 4-hour fire.

3.5.5 Waste

The facility and all waste systems must be designed to maintain at an absolute minimum the release of radioactive materials to the environs, during both normal plant operation and the occurrence of a DBA. Waste systems include retention containers, cleanup systems for liquids and solids, and analytical equipment.

Sanitary Waste

Sanitary wastes include the nonradioactive wastes usually found at a facility, e.g., discharges from noncontaminated chemical laboratories, showers, and lavatories. The sanitary waste system and the plutonium handling area should not be connected. Sanitary sewers should discharge into an onsite, approved sanitary-sewage treatment system. Current federal, state, and local codes regarding the discharge of sanitary wastes should be met.

Potentially Contaminated Wastes

Potentially contaminated wastes include process-coolant-water blow-downs from heating and cooling systems, process steam condensates, and discharges from mop sinks. Natural runoff from the process building may be included in this category. Sufficient holdup capacity must be provided so that wastes can be retained until they are analyzed and shown to be within acceptable limits for discharge. Sufficient capacity must also be provided for the collection of water from firefighting.

A capability for monitoring prior to discharge must be provided for potentially contaminated wastes. The monitoring requirements can be satisfied by one of the following:

- 1) continuous monitoring of the waste stream, with automatic diversion to a retention system when a preset level of contamination has been reached; or
- 2) holdup of liquid waste, coupled with batch sampling and analysis prior to release.

The signals from continuous waste-monitoring systems should read out in the health physics office and should alarm at the central annunciator panel.

If liquid waste is discharged to the environs, the contamination level should be as low as practicable. The effluent concentration measured at the point of discharge (e.g., pipe or conduit) should not exceed the maximum permissible concentration for uncontrolled areas as given in ERDAM-0524⁽¹⁰⁾ for soluble plutonium, and as specified by the Federal Water Pollution Control Act, Amendments of 1972, Public Law 92-500.

Contaminated Wastes

Any potentially contaminated waste that does not pass the monitoring test for release to the environment, and any liquids known to be contaminated, must be transferred to a liquid waste treatment plant capable of cleaning up liquid waste to meet "as low as practicable" guidelines and other federal, state or local discharge standards. Guidance for the design of liquid waste treatment systems is presented in ERDA Regulatory Guide 3.10.⁽¹⁶⁾

3.6 SPECIAL SYSTEMS

Special instrumentation must be incorporated in plutonium facilities to insure safe and orderly operation. The following examples must be included as a minimum.

3.6.1 Air Sampling

Airborne alpha activity can be monitored either by air samplers that are part of a central system or by a continuous air-monitoring system (CAMS). The use of both is preferable. The collection heads of the air samplers should be located near positions frequently occupied by operating personnel. Continuous air-monitoring systems should be located throughout the radioactive materials handling area and should take samples in a manner that provides a representative basis for determining personnel exposure. The air-monitoring systems should follow the standards set by ANSI's Standard N13.1, "Guide to Sampling Airborne Systems in Nuclear Facilities,"⁽¹⁷⁾ and draft standard N317. These two standards should also be consulted if existing buildings are modified or upgraded.

All gaseous and/or fine particulate effluents exhausted through ducts must be monitored following filtration. Sampling probes should be designed for isokinetic sampling. Analytical procedures should be of sufficient sensitivity to verify the normal annual release, and information should be provided to enable the calculation of the radiation dose by the most recent dosimetry models of the International Commission on Radiation Protection (ICRP).

Stack effluents should also be monitored using CAMS, which records continuously and annunciates excessive levels of radioactivity by an alarm. Effluent releases should be confirmed by a suitable environmental monitoring program.

3.6.2 Breathing-Air

A plutonium facility should be provided with a system capable of supplying breathing-air to a number of work stations in each occupied operating area where:

- under abnormal or accident conditions, radioactive airborne particulates are expected to exceed 20 times the concentrations set forth in 10 CFR, Part 20, Appendix B,⁽¹⁸⁾ or
- maintenance or decontamination may require that personnel enter cells or other areas containing large amounts of loose radioactive material.

Several factors should be considered in the inclusion of a breathing-air supply system in a facility. The quality requirements of the Occupational Safety and Health Administration (OSHA) are applicable to these systems (see 29 CFR 1910.134(d)).⁽¹⁹⁾ The system shall be able to be monitored for purity during its operation and it should be among the critical systems supplied with emergency power.

Hose couplings and manifolds should be designed to preclude the connection of respiratory protective equipment to any system other than the breathing-air system. Manifold outlet stations should be provided near the normal entrance and outside of zones with a high contamination potential, to facilitate "suiting up." A few outlets should be installed at strategic locations outside the building for use in recovery operations in the event of an emergency.

3.6.3 Personnel Monitoring

The use of personnel survey instruments at suitable locations within the process area is desirable. For a compartmentalized facility, survey instruments should be provided at the exit from each compartment; in contamination control change rooms, monitoring instruments must be used. The use of survey meters located on or close to adjacent glove boxes is encouraged, as are procedures requiring that hands be surveyed each time they are withdrawn from the gloves. If air sampling heads are used, flexible lines will allow placement of the heads near the worker's breathing zone.

3.6.4 Criticality Safety

Criticality alarm systems should be provided in each area where an accidental criticality is possible. The alarm system must be consistent

with the provisions of ANSI Standard N16.2-1969, "Criticality Accident Alarm System."⁽²⁰⁾ The criticality alarm system must remain functional or give a warning of any malfunction caused by an accident; it should be designed and constructed to indicate any internal failure.

3.6.5 Nuclear Accident Dosimeters

The nuclear criticality alarm system should be backed up by nuclear accident dosimeters. These dosimeters should meet the performance parameters and location requirements of ANSI Standard N13.3-1969, "Dosimetry for Criticality Accidents."⁽²¹⁾

3.6.6 Other Systems

Many systems employed within a plutonium facility are not directly related to personnel safety or radiation protection; however, people responsible for personnel protection should be aware of these systems. Some examples follow.

- 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 assure the integrity of all process piping, tanks, and other containment equipment, including that used for liquid effluents.
- Vacuum or gravity systems to transfer toxic and corrosive liquids.

Special controls should be provided for flammable, toxic, and explosive gases, chemicals and materials admitted to the plutonium handling areas. Gas and chemical storage facilities, including distribution piping systems, should conform to good design practice and applicable codes and standards. Consideration should be given to compatible groupings which, under accident conditions or leakage, would minimize any adverse combining of materials. Means for the remote shutoff of piping should be provided. In addition, the following specifics should be followed:

- 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.
- Properly vented, pressurized gas lines should be used in the plutonium handling areas.

An adequate supply of instruments, clothing and other appropriate materials should be maintained in separate weatherproof shelters for use in reentry operations following an emergency evacuation. At least one of these shelters must be operational following a DBA. Facilities for equipment maintenance should also be provided. Decontamination facilities should have all necessary utilities, including those for proper disposal of wastes.

3.6.7 Monitoring and Alarms

In order that abnormal conditions may be correctly interpreted and remedial action taken promptly, all monitoring system readouts and alarm indicators that relate to personnel safety or the integrity of a building should be centralized at a location that is continuously manned and has guaranteed accessibility. The inclusion of the following specific alarms and signals should be considered:

- 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 must be supplied with emergency power in the event that normal power fails, as they must remain functional at all times. Reliability should be provided by designed features such as redundant circuits, instruments that self-check and are tamperproof, etc. All monitoring, surveillance and alarm systems should be tested periodically.

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RADIATION PROTECTION

SECTION 4

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SECTION 4

RADIATION PROTECTION

The successful operation of a plutonium facility requires maximum attention to maintaining contamination control at all times. The potential for loss of control must be minimized by any and all means possible, including the selection of qualified personnel. Instruments must cover the spectrum of radiations, i.e., alpha, beta, gamma and neutrons, and the types of shielding needed to control personnel exposure must be effective for both gamma and neutron radiations. This section brings together the basics for the establishment of a sound radiation protection program.

4.1 PERSONNEL REQUIREMENTS

It is the consensus of individuals responsible for radiation protection programs associated with plutonium facilities that experienced personnel are preferred over all others. Those with experience can then be utilized to train others. The selection of operating personnel is also important; adherence to rules and procedures is mandatory to assure their own personal safety and that of others.

4.1.1 Experience

A cadre of operating and maintenance personnel who have experience in the operation of a plutonium facility should be established, if possible, during the construction of a new facility. The remainder of the staff that will operate the facility should be hired as soon as possible and receive formal and informal training from the trained personnel. This step is extremely important to allow all personnel to grow with the facility. Once operations start, potential problems will already have been identified and administrative or engineering changes made to alleviate them. Staff members hired later should have 3 to 6 months of direct involvement under trained personnel before they are allowed to operate independently.

At least one professional health physicist should be on the staff of each major plutonium facility. This individual should have several years of experience in the operation of plutonium facilities or should be a certified health physicist. At least one certified radiation protection technologist should also be assigned to the facility. Where possible, radiation monitors should have experience dealing with plutonium, since such experience is an important prerequisite to allowing the monitors to work on their own; an alternative, however, is to hire a person with a minimum of 1 year of experience under a qualified radiation monitor and require 3 to 6 months of direct experience in the facility under the technical leadership of a trained radiation monitor with experience in that facility.

4.1.2 Personnel Training

A thorough training program should be established. All staff members (operations, maintenance and radiation monitoring) should receive training

in the rigid procedures and techniques for handling transuranic radionuclides, and practice in dealing with nonradioactive materials. The following should be included in formal training programs:

- All new employees should be trained to handle transuranics. Practice work with nonradioactive materials should be a fundamental part of each training program, and each should include the demonstration and practice of contamination and dose control methods, equipment usage, emergency procedures, and methods of bagging-in and bagging-out materials and changing gloves in glove boxes.
- The biological and dosimetric aspects of work with transuranic materials should be described. The proper use of all protective equipment and clothing should be described, with emphasis on respiratory protection.
- Waste-handling and disposal methods should be included in the training program.

Refresher training should be given to all employees to assure that proper work methods are sustained.

4.1.3 Performance Standards

No restrictions based on physical characteristics or sex should limit the work of staff in a plutonium facility. The primary requirement should be the ability to perform the work required. There is, however, an ERDA requirement that fertile female workers be advised that the NCRP advises 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."^(a) However, personnel with a dose commitment in excess of 50% due to deposition of long-lived radioactive materials should be assigned to activities in which the potential for additional deposition is extremely low.

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

- following rules explicitly
- not taking procedural shortcuts
- being sure that decisions are properly reviewed prior to initiating any new activities.

(a) NCRP Report No. 39, Basic Radiation Protection Criteria. Recommendations of the National Council on Radiation Protection and Measurements, January 15, 1971.

Since the entire operational and safety program depends heavily on the staff's attitude and commitment, morale must be kept high.

4.1.4 Radiation Monitoring Staff

A successful health physics program in a plutonium facility is highly dependent on the availability of adequate staff support in areas such as environmental monitoring, instrument maintenance and calibration, and internal and external dosimetry. By definition, radiation monitors are those who assure that the facility is operating within established radiation-protection constraints.

In addition to at least one professional health physicist, the facility should be staffed with a minimum of 2 radiation monitors for the first 30 operating personnel, with one additional monitor for each additional 30 operating personnel or fraction thereof. 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 radiation monitors needed.

4.1.5 Health Physics and Operations Organization

A plutonium facility must have a clear organizational chain of command. Health physics personnel should have the authority to balance operations with safety, and should therefore not report directly to the administrators for operations. When shift work is involved, the shift supervisor may make minor health physics decisions in support of the shift's radiation monitors; however, decisions involving basic policies and procedures should be directed to a separate health physics organization.

A system of guides, policies and procedures should be established to clearly identify the interrelationships, responsibilities and authorities of those involved with the development, operation and maintenance of the facility. These guides, policies and procedures should be documented, with a review required at least once each year.

4.2 INSTRUMENTATION CONSIDERATIONS

The radiations resulting from the radioactive decay of plutonium are varied and include alpha, beta, gamma and x-rays, and neutrons (see Appendix). An effective monitoring program for plutonium requires instruments that are responsive to all of these radiations except the 7-keV $\bar{E}$ beta resulting from the decay of ^{241}Pu .

4.2.1 Types of Instruments and Measurements

Most contamination control surveys require alpha-sensitive instruments. Exposure rate surveys are normally conducted with photon-sensitive instruments with known energy responses down to and including 0.010 MeV. Neutron surveys become important when processing tens of grams of ^{238}Pu or hundreds of grams of mixed isotopes of plutonium, particularly plutonium compounds (i.e., PuO_2 , PuF_4 , etc). The neutron survey is possibly the only one of importance 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 personnel exposure.

Under certain conditions, the extremities and forearms of personnel working with plutonium (and other radioactive materials) in glove boxes are exposed to the various types of radiation mentioned above. Generally, so-called "contact dose rates" are established to control the exposure of the workers' hands. No such measurement is currently made to determine the neutron "contact" dose-equivalent rate because no suitable instrument is available to make the measurement. However, an estimate of the neutron dose equivalent could be made and could be added to the other measurements for purposes of exposure control. Californium-252 is suggested for use as the calibration source because its neutron energy distribution is similar to that of plutonium metal and PuO_2 .

4.2.2 General Performance Criteria for Instruments

Programs for inplant monitoring of plutonium consist mainly of surveys of airborne and surface contamination and surveys of dose rate. The general performance criteria for the instrumentation needed to conduct these programs are described in ANSI Standard N317.⁽¹⁾ The following criteria are typical of what this standard requires.

Alpha survey instruments should ideally have a minimum detection capability for alpha surface contamination of 220 dis/min per 100 cm² of surface area but must have a sensitivity of at least 500 dis/min per 100 cm². This requirement must be met in the presence of a radiation field of 0.10 rem/hr of neutrons in the energy range of thermal to 10 MeV, and/or in the presence of 0.10 rem/hr of photons in the energy range of 0.010 to 1.25 MeV.

Airborne contamination monitors, normally of the continuous type commonly referred to as continuous air monitors (CAMs), must meet the following criteria. More information on air monitoring is contained in ANSI Standard N13.10:⁽²⁾

The minimum detection level shall be 1 maximum permissible concentration (MPC) of ²³⁹Pu in 8 hours (8 MPC-hr) in the presence of nominal amounts of naturally-occurring alpha emitters such as radon and thoron and their daughters. (This requirement is applicable to MPC values as of the date of this manual.)

Surface contamination monitors include hand and/or shoe counters and instruments (or probes) with sufficient flexibility to survey pieces of equipment, including exterior clothing.

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.

Criticality alarm systems must comply with ANSI Standard N16.2, "Criticality Accident Alarm Systems."⁽³⁾

Other emergency instrumentation should provide ranges for all radiation dose rates and contamination levels potentially encountered at the time of an accident. Normally, dose rate capabilities from a few mrem per hour to a few hundred rem per hour will be required, while capability requirements for the contamination level may range upward from 500 dis/min per 100 cm² (~ 2 pCi/cm²) for alpha contaminants and 2.00 dis/min per 100 cm² (~ 10 pCi/cm²) for beta-gamma emitters.

Calibration of instrumentation should be conducted under conditions and with radiations similar to those that may be encountered, including those resulting from an accident. The response characteristics for different energies and multiple radiation fields should be known. (See ANSI Standard N323, "Radiation Protection Instrumentation Test and Calibration," currently in draft form.)

4.3 CONTAMINATION CONTROL

Radiological contamination should be surveyed and controlled through operational controls, radiation surveys, and the use of protective clothing.

4.3.1 Operational Procedures

Housekeeping

Housekeeping in a plutonium facility is particularly important for at least three reasons related to radiation effects.

- The inventory of contaminated and potentially contaminated scrap and equipment must be kept to a minimum, as all such materials are subject to special monitoring and accountability, whether maintained or destroyed.
- Housekeeping must promote the control of radioactivity and minimize the spread of contaminants and the potential for accidents involving contaminants.
- Conditions leading to possible personnel exposure must be minimized.

Anything not absolutely necessary to an operation should never enter the contaminated or potentially contaminated zone; all packaging and unnecessary protective envelopes should be removed before materials are introduced into the contaminated zone. By the same token, items not necessary to the process should be promptly removed, particularly from glove boxes, and not left to accumulate and become safety hazards, potential fire hazards, sources for radioactive (dust) accumulation or sources of exposure.

Recent trends in housekeeping inside glove boxes have emphasized fire and explosion control. Currently, only metal containers are recommended for the accumulation of scrap and wastes of any kind. No more than a minimum of any type of combustible material should be allowed to accumulate in glove boxes. When explosive, flammable or volatile liquids are allowed, they must be rigidly controlled and all residues must 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 systems are favored. Electropolishing contaminated metal tools and equipment has been shown a good method of decontamination and allows reuse in some cases.

Good housekeeping must be constantly encouraged by management. Mandatory, routine cleanup periods are becoming more common, encouraged by the increasing cost of storing and disposing of contaminated materials. Real time, computerized accountability for nuclear materials will almost certainly soon be enforced, and will probably necessitate housekeeping of a higher quality than any yet known.

Processing Technology

The processing technology unique to plutonium has resulted from the substance's high radioactive toxicity coupled with criticality restrictions. Security measures may introduce additional, far-reaching restrictions in some areas.

The chemical processing technology for plutonium is not unique, but special aspects of accountability and personnel protection superimpose many restrictions and complicate process machinery and maintenance.

Problems in the handling of plutonium solids are unique. Dispersion, potential proliferation and accident potentials appear less well-defined and more widespread in the handling of solids than in the processing of liquid chemicals. The hot-particle theory has largely been discounted but uneasiness about occupational exposure and environmental control persists, requiring an expanding monitoring capability for the industry.

The introduction of high-exposure plutonium is revolutionizing the solids-handling industry with respect to the technology for personnel exposure control and mandatory automation. This development, coupled with the increasing concern over security, may necessitate the dilution of high-level radiation from plutonium, and complete remote handling in all plutonium processing.

Gloves, Storage, and Ports

In recent years so many new types of gloves have been developed that glove usage should be tailored to the particular needs of the job. For processes that require maximum dexterity, the 0.015-inch (0.038-cm) neoprene glove is still superior. Coated hypalon gloves are superior to neoprene for glovebox process operations that involve nitric acid or ozone levels that may cause problems. Viton gloves have proven to have longer life than neoprene under many operating conditions, but suffer somewhat from stiffness. Where high radiation levels are encountered, lead-loaded gloves are necessary; again, stiffness and loss of manual dexterity must be tolerated.

In general, black neoprene gloves are standard and the most economical to use where process conditions do not produce rapid glove deterioration. Radiation fields quickly induce surface cracking in black neoprene. However, in plutonium glove boxes this change contributes less seriously to glove failure than to greater dust contamination of gloves and higher glove dose rates, which in turn result in extremity exposures. Hypalon is more resistant to surface cracking and acid deterioration; this characteristic will, in many cases, make hypalon gloves the most economical, despite their higher unit cost.

Glove storage problems occasionally occur. Experiments and static tests have not provided an adequate explanation of the sporadic problems encountered. Tests in which gloves were stored under different lighting conditions (ultra-violet and fluorescent) and under stressed conditions (creased or bent) have not given consistent results, nor has low-level radiation (consistent with plutonium levels) produced clear-cut deterioration of gloves. Ozone at levels as low as 0.2 ppm causes rapid deterioration of neoprene gloves but has a negligible effect on hypalon or viton gloves.

Face masks must be worn when gloves or plastic bags on plutonium glove boxes are being changed. Special glove-changing devices that have been developed work well but are still too expensive to have attained extensive use.

All gloves in normal use at plutonium processing installations should be inspected daily. In addition to the monitor's inspection, all operating personnel must check themselves after every glove usage. Gloves not in use should

be capped off with a glove cover or plastic bag. Gloves not in regular use are sometimes pigtailed, with tape placed over the glove port; however, this practice is not recommended. Gloves should never be tied or pigtailed outside the glove box; they should be stored extended into the box when not being used.

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 involving 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, powder-containing glove boxes where the levels of radioactive dust quickly increase personnel exposure, often by an order of magnitude. In many cases vacuuming reduces the exposure level more than a wipe-down with a damp cloth would, and can be done more quickly and with less scrap 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.

4.3.2 Radiation Surveys

The rigor of the various elements of a radiation monitoring program for work involving plutonium depends on the kind and quantity of material present and its dispersibility. To characterize the various aspects of a radiation monitoring program, the process area and controlled areas have been further classified as shown below.

- At a Class A work station, work with plutonium is conducted in one or more glove boxes.
- At a Class B work station, work is performed in open-face hoods.

- 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. Variations in the program would be based on the operating experience, attitude and training of the operating people, the age of the facility, the 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 exist in one room, the features of the program should encompass the requirements of the highest classification of work station within that room.

Contamination Surveys

Controlled areas and support facilities should be surveyed for contamination on a regular basis.

Work Stations. All gloves, work surfaces, floors, etc. within the work area should be surveyed on the following frequencies:

Class A - Weekly survey if there is ≥ 1 mCi within the work station
- Monthly survey if there is < 1 mCi

Class B - Monthly survey

Class C - Quarterly survey

Support Facilities. The change room and other support facilities within the controlled area should be surveyed for contamination on a quarterly basis. The support areas outside the controlled area (office, lunch room, lobby, cold-maintenance shops) should also be surveyed quarterly.

Release Surveys. All scrap and waste should be surveyed prior to removal from the controlled area. Material released for uncontrolled use should have no detectable smearable transuranic contamination and less than $0.1 \text{ nCi}/100 \text{ cm}^2$ of fixed transuranic contamination.

External Exposure Surveys

Measurements of external exposure should be made at all points of personnel exposure 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, to identify the buildup of plutonium in HEPA filters and glove boxes. Glovebox exhaust filters should be changed with sufficient frequency to assure that the resultant exposure will contribute less than 10% per year to the total exposure of operating personnel.

Airborne Contamination Surveys

Where 1 or more mCi of dispersible plutonium is involved, air monitors should be on hand at the work station to promptly indicate any loss of control. The sample taken by the air monitor can also serve as the total integrated measurement of air contamination in the room. The samples should be analyzed after a 5- to 7-day decay period to provide a more sensitive measurement of the air contamination levels.

The work area should be analyzed to identify the points at which the release of airborne contamination is most likely. A sample-collection point should be located near each of the critical release points. Up to five sample-collection points can be connected to one air monitor, using 1/2-inch (1.3-cm) diameter stainless steel tubing with smooth bends of at least a 6-inch (21-cm) radius. The sample point should be no farther than 10 feet (3 m) from the monitor. Up to five lines can be brought together at the monitor's location to represent the total integrated sample.

Monitoring of Operations

While self-monitoring is generally sufficient, specific conditions may require the presence of a radiation monitor.

Self-Monitoring. All personnel working in Class A or B work stations should be required to survey their hands and forearms each time they are withdrawn from the enclosure and after each glove replacement or bag-out operation. A contamination survey of shoes, clothing and hands prior to personnel's leaving the work area should also be required.

Continuous Surveillance By Radiation Monitoring. Continuous radiation monitoring must be provided during periods of high or unusual risk associated with the work in the area. These situations include the potential or actual breaching of the integrity of the glove box or associated systems, including breaching for maintenance (replacement of panels, replacement of filters, or repair of vacuum systems). Work involving the use of greenhouses requires continuous coverage by a radiation monitor.

4.3.3 Protective Clothing

Various types of protective clothing may be required for different laboratory operations. Laboratory coats, shoe covers, gloves, coveralls, plastic or rubber suits and air-purifying or atmosphere-supplying masks may all be required for operations with transuranic radionuclides.

Personnel performing operations in the process area should wear coveralls and shoe covers. For inspections or visits, lab coats are permissible. When contaminated wet areas are to be entered, waterproof (plastic or rubber) clothing should be worn. No personal outer clothing should be permitted under coveralls.

Shoe covers should be worn except where wet operations are involved; in these areas, rubbers or boots should be provided.

Hands should be protected by a minimum of two barriers, including at least one pair of surgeon's gloves plus the glovebox glove. 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.

For work in controlled areas, lab coats, shoe covers and surgical gloves should be considered the minimum requirements.

Special Maintenance Requirements

For special maintenance work that involves significant quantities of contamination, the following should be the minimum requirement for protective clothing:

- two pair of coveralls
- canvas boots taped to the inner pair of coveralls, with rubbers over the canvas boots
- one pair of surgeon's gloves taped to the inner coveralls, with a leather, cotton or rubber outer pair of gloves.

Respiratory protection (with the head covered with a cloth hood and a plastic or rubber waterproof covering) should be added where liquids are involved in the operation.

Respiratory Protection

Respiratory protection should be readily available. Masks should be utilized for all bag-outs, bag and glove changes, and any situation involving a potential or actual breach of containment. Protection in the form of air-purifying or atmosphere-supplying masks is required whenever concentrations of radionuclides in the air may exceed the maximum permissible limits. For good performance, the mask must 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.

The top-grade, full-face, high efficiency filter air-purifying mask is considered about 90% efficient for particulate materials and ineffective against radioactive materials in gaseous form. According to this estimate, a full-face, air-purifying mask can be worn where the concentrations of radionuclides in air range from 1 to 10 times the MPC_a . (See Section 4.4.7 for MPC_a values.)

For air concentrations greater than 10 times the MPC_a , an atmosphere-supplying mask is required. Full-face supplied-air masks can be assumed to be 99% efficient for either particulate or gaseous materials.

Half-masks are used in plutonium facilities; however, most experts agree that full-face masks are more desirable.

The elements of a satisfactory respiratory protection program are described in ANSI Z88.2.⁽⁴⁾ Some of the features important to plutonium facilities are described below.

- Respirator-type masks fitting over the nose and mouth only should not be used for protection against transuranic radionuclides.
- All masks and filters should be tested for adequate performance prior to use.
- After each use, masks should be retested for proper operation, surveyed to assure freedom from contamination, and sterilized prior to reissue.
- Employees using masks should receive training in the performance features of the masks and how they should be used. Emphasis should be given to proper donning of the mask and testing for a good face fit after donning.
- Care should be taken to assure that every air-purifying mask is equipped with a filter and that every atmosphere-supplying mask is used with approved air supplies. Serious accidents have resulted when masks were used with filters not in place or atmosphere-supplying masks were connected to improper supply.
- Standard fittings meeting recommendations of national standards groups should always be used.
- All masks suspected of defective performance should be removed from service.

4.4 INTERNAL EXPOSURE CONTROL

An internal dosimetry program aims at assessing the risk from internally deposited radioisotopes and is essential for a quality health physics program in a facility handling plutonium. The program may vary from a minimal operation for a small laboratory to a relatively sophisticated program for a facility handling kilogram quantities of plutonium.

Three methods can be used to monitor exposure from the body's intake of plutonium:

- bioassay-radioanalysis of plutonium in excreta
- in-vivo examination
- air sampling/worker occupancy.

An effective internal dosimetry program should consider all three methods. Since estimating exposure by air sampling is the least reliable method, air sampling should be directed primarily toward rapidly detecting any loss of containment.

The internal dosimetry program should demonstrate management's responsibility for workers through the consideration of several subprograms:

- Routine Surveillance - to assure that engineered confinement and facility monitoring programs are effective and that no undetected intake above predetermined acceptable levels has occurred.
- Accidental Intake Detection - to assure the rapid detection of accidental intake and provide prompt assessment of the probable magnitude.
- Mitigatory Action - to assure that prompt therapeutic measures are taken following an accidental intake.
- Diagnostic Evaluation - to determine the internal deposition and any need for therapeutic treatment and/or work restrictions following a known or suspected exposure.

This section provides guidance on all but the last of these programs. Diagnostic evaluation is complex and should be performed by those

competent in internal dosimetry evaluations. Evaluation methods are beyond the scope of this section but are included in the Appendix. An ANSI Standard, "Internal Dosimetry Standard for Plutonium," when issued, will provide further guidance on the internal dosimetry programs.

4.4.1 Analytical Capability

In the administration of the internal dosimetry program, special analytical capabilities are required. These capabilities are reviewed in the following subsections.

Bioassay

Estimates of the internal deposition of plutonium can be made by an analysis of the plutonium content in various excreta. The primary excreta analyzed are urine and feces; blood, nasal smears, and sputum samples can also provide meaningful data.

The processing of bioassay samples requires well trained laboratory technicians to isolate small quantities of plutonium and to avoid cross-contaminating samples during processing. The usual method of analyzing urine, blood and fecal samples consists of chemically isolating the plutonium and electrochemically depositing it on a planchet.^(5,6) The total alpha content of the plutonium can then be determined by:

- gross alpha counting using gas-flow proportional counting (detection limit: 0.1 dis/min per sample)
- alpha energy analysis using surface barrier detectors to identify the activity per isotope of plutonium (currently, this technique requires >200 dis/min per sample)
- alpha track counting using nuclear track emulsion film (detection of 0.05 dis/min is possible using this method).

In-Vivo Examination (Chest)

The measurement of plutonium in the chest by in-vivo methods is complex. Complications encountered include the low energy of photons emitted by most isotopes of plutonium; the magnitude of the permissible

burdens; differences in chest wall thickness from one subject to the next; distribution within the lung; interfering Compton radiations from naturally-occurring radionuclides and fallout; and the inability to distinguish which organ or organs in the chest are being measured. Since in-vivo chest examinations cannot determine the burden of individual organs in the chest, the total chest burden is usually attributed to the deposition in the lung.

Several detector systems for the measurement of plutonium in the chest are currently in use.⁽⁷⁻¹³⁾ The most widely used systems currently include NaI(Tl) detectors, phoswich detectors, and proportional counters. Intrinsic germanium detectors are also used occasionally. These detector systems are designed to measure plutonium in the chest by either 1) direct measurement of the 17-keV x-rays emitted by plutonium or 2) measurement of the 60-keV x-rays emitted by ^{241}Am , which is associated with most plutonium mixtures.

With the current state-of-the-art in chest counting, detection of the permissible lung burden of ^{239}Pu (16 nCi) by direct measurement of the 17-keV x-rays is possible only under ideal conditions. In most situations, a more realistic appraisal of the minimum quantity detectable in the chest is ≥ 40 nCi of ^{239}Pu . Plutonium-238 is comparatively easier to detect; a burden of approximately 5 to 6 nCi in the chest can be measured if the conditions are again ideal.

The direct measurement of plutonium is limited by these relatively high detection limits, which tend to become even higher if appreciable quantities of ^{241}Am are present. The evaluation of data obtained by direct measurement is complicated, requiring a knowledge of several parameters. The complexity of these analyses causes the reproducibility of results to be suspect, making their use in routine programs difficult.

Since most plutonium mixtures contain ^{241}Am , plutonium chest burdens may often be determined indirectly by measuring the ^{241}Am 60-keV photon and applying an appropriate Pu/Am ratio. This technique utilizes relatively simple equipment and computational methods, and provides more reproducible results. The ^{241}Am associated with most mixtures of plutonium can be measured for burdens as small as 0.3 nCi of ^{241}Am . If the ratio of

plutonium alpha emitters to ^{241}Am in the material is known or can be determined, the Pu alpha burden can be calculated based on the ^{241}Am measurement. For many mixtures, except freshly separated material (<6 months), the Pu alpha/Am alpha ratio probably will not exceed 15: for high-exposure or well-aged plutonium, the ratio may approach 5 or less. Thus a lung burden of up to 4 to 5 nCi of Pu alpha can be detected indirectly by measuring the ^{241}Am .

While direct measurement of ^{241}Am is relatively easy, the indirect determination of plutonium does require knowledge of the isotopic composition of the contaminant before an accurate evaluation of the plutonium deposition can be made. Indirect determinations are also limited by the fact that certain mixtures of plutonium may not contain significant quantities of ^{241}Am . In order for indirect in-vivo determination of plutonium to be a viable technique, the health physicist must have a thorough knowledge of the various mixtures of plutonium used in the facility.

Additional complications in the indirect determination of plutonium chest burdens are: 1) the difficulty of assigning a Pu alpha-to-Am alpha ratio when the burden is incurred through a series of low-level intakes; 2) the different clearance rates of plutonium and americium; and 3) the ingrowth of americium from the decay of inhaled ^{241}Pu .

The technique that best suits a particular need should be determined based on the detection capabilities required, the isotopic composition of the plutonium encountered, the cost, and the desired reproducibility of results.

Calibration⁽¹³⁾ of in-vivo chest counters is a complicated task requiring documentation of several variables including:

- the chest-wall thickness of the subject
- the effect of Compton radiation from high-energy gamma emitters
- the effect of ^{241}Am on the detection limits of plutonium
- the environmental background
- the source distribution.

The actual determination of the counter's efficiency requires a phantom that closely resembles the characteristics of the subjects to be counted. The phantom should have a variable chest-wall thickness and should allow for differing distributions of the source within the phantom.

In-Vivo Examination (Wounds)

The detection of plutonium in wounds⁽¹⁴⁾ is relatively simple and lacks the problems encountered in in-vivo chest counting. An accurate determination of detectable quantities of plutonium ($>1 \times 10^{-4} \mu\text{Ci}$) in wounds less than 5 mm deep is obtainable using thin-crystal NaI detectors (1 to 2 mm). For deeper wounds, corrections for depth become increasingly important. Collimated NaI crystals are useful in determining the location of the contaminant, but give little or no information as to the depth of the wound.

Reasonable estimates of the depth of the injury can be obtained by the use of solid-state germanium or silicon detectors to measure the relative absorption of the low-energy x-rays emitted by plutonium.⁽¹⁵⁾ Depth data is often invaluable in determining if excision is warranted or even possible without causing physical impairment. Portable wound counters that have special probes but are less sensitive are useful in assisting the physician during excision.

Particle Size, Transportability and Isotopic Composition

A knowledge of particle size and transportability⁽¹⁶⁾ is necessary to estimate deposition using the Task Group Lung Model (TLGM).⁽¹⁷⁾ Isotopic composition is important in interpreting in-vivo examination results and in accounting for the ^{241}Pu content when urine analysis techniques are based on alpha emission only. Knowledge of the isotopic composition is also necessary to estimate the plutonium in feces when photon spectroscopy (60-keV photons from ^{241}Am) is used for rush analyses of fecal samples.

Particle-size determinations are often made by electron microscopic techniques; however, these techniques are expensive and time-consuming.

The autoradiographic technique developed by Andersen⁽¹⁸⁾ is direct, relatively simple and useful for TGLM evaluations. Solubility determinations are normally made using saline or special solutions closely resembling body fluids over several periods of time. The isotopic composition must be analyzed by mass spectrometry and in many cases, because of the low mass of ^{238}Pu (particularly in weapons-grade material), alpha spectroscopy may also be required.

An advance knowledge of the solubility and isotopic composition of a material being processed is useful; however, these data may not apply to the actual exposure material. For the more serious intakes, therefore, these analyses should be obtained directly from the exposure material. Air samples, nasal smears or blows, and smears of skin contamination near the nostrils are useful sources in determining particle size, solubility and isotopic composition. Activity levels of 20 to 30 nCi are normally required to determine isotopic compositions, while less activity is needed to determine solubility and particle size.

Nasal Smears

Nasal smears or nose blows can provide meaningful data for early estimates and a good source of samples for special analyses. Care is necessary when obtaining nasal smears, to avoid cross-contamination with facial contamination and to provide a source suitable for analysis. The following procedure has been found useful in taking nasal smears:

- Fold a slightly damp Whatman filter (1-in. or 2.5-cm diameter) several times.
- Insert a Cue Tip into the center of the fold.
- Have the exposed individual carefully insert the filter into the nostril and thoroughly wipe the nasal passages. Use a separate filter for each nostril. Be careful not to cross-contaminate.
- Dry the smear under a heat lamp.
- Survey the smear with a portable meter to check for gross contamination.
- Measure the radioactivity using an appropriate alpha-counting instrument to obtain an accurate measurement.

4.4.2 Routine Internal Dosimetry Surveillance

Area monitoring to detect plutonium releases is normally conducted in work locations; however, these monitoring programs are not adequate for the routine surveillance of workers. A routine internal dosimeter surveillance program should be conducted for all personnel.

The routine internal dosimetry surveillance program should consist of bioassay and/or in-vivo examinations. A properly designed program will provide assurance that engineered safeguards and administrative controls have been effective. The program should also assure that depositions over 5% of the maximum permissible annual dose commitment to any organ will not go undetected. When establishing a routine program for the surveillance of plutonium workers, several factors must be considered:

- the type and quantity of plutonium
- the transportability of the plutonium
- the probability of accidental release
- the sensitivity or detection level of the examination
- the quality of the air sampling and survey program
- the acceptable undetected dose commitment
- the deposition history and/or work-restriction status of the individual
- the frequency of examinations.

The physical and chemical form of the plutonium, its transportability and the quantities handled are usually outside the immediate control of the health physicist. However, the probability of release, the quality of the air sampling program, the detection sensitivity and the acceptable dose commitment are or should be controllable by health physics management, within the bounds of economics and the state-of-the-art. In addition, the health physicist must consider the deposition history of personnel when making up routine surveillance schedules. Persons who have worked previously with plutonium or are likely to work with it in their current assignment should receive a preparatory evaluation. This evaluation is useful in determining whether a person has received any internal deposition

from prior employment and in evaluating any subsequent intake. A urine sample that contains measurable quantities of plutonium or a positive in-vivo examination provides a good baseline against which to compare the results from subsequent examinations.

Urine and Fecal Sampling

According to the excretion rates predicted using the TGLM, routine fecal sampling should provide detection of a lower dose commitment than urine sampling. This is especially true for high-fired oxides or other very insoluble forms.

For more soluble compounds of plutonium, field data indicate that a lower dose commitment can be detected by urine sampling than by fecal sampling. Except for accidents involving fire and facilities routinely using high-fired oxides or carbides, urine sampling is suggested as the preferred mode. It is also more acceptable to the staff. Urine sampling and/or in-vivo counting are the most widely accepted methods of routine surveillance.

Bioassay and In-Vivo Examinations

Bioassay and in-vivo examinations are normally scheduled as part of a routine program based on calendar periods (monthly, quarterly, semi-annual, annual, etc.). The frequency is chosen in accordance with the factors noted previously. The data listed in Tables 4.1 and 4.2 give estimates, for various sampling frequencies, of the undetected dose commitment during the first 12 months post-intake. Dose commitment to the lung is per nCi initially deposited while dose commitment to the bone is based on an analysis of urine data using appropriate models.

Guidelines for Establishing Routine Surveillance

- When the quantity of unencapsulated plutonium handled in any separate laboratory or work area exceeds 100 nCi per source or 10 μ Ci total, a determination should be made by a qualified health physicist as to whether an internal dosimetry program is required.

TABLE 4.1 Undetected Dose Commitment to the Lung (in rem)
During the First 12 Months Post-Intake (a)

In-Vivo Examination Frequency	Half Life in Deep Lung		
	50 Days	500 Days	700 Days
Monthly	0.3	0.9	0.9
Quarterly	0.8	0.9	1.0
Semiannual	2.7	1.1	1.1
Annual	35.0	1.4	1.3
Biannual	--	2.3	1.9
Five-Year	--	11.0	5.5

(a) Undetected dose commitment to the lungs per year per nCi of ^{239}Pu initially deposited in the lung, as a function of half life in the pulmonary region and in-vivo examination frequency.

TABLE 4.2 Undetected Dose Commitment to Bone (in rem)
During the First 12 Months Post-Intake(a)

Bioassay Sampling Frequency	Half Life in Deep Lung or Other Metabolic Pool		
	50 Days	500 Days	700 Days
Monthly	0.1	0.5	0.6
Quarterly	0.2	0.5	0.6
Semiannual	0.4	0.5	0.6
Annual	0.6	0.5	0.6
Biannual	1.0	0.5	0.64
Five-Year	2.0	1.2	1.0

(a) Undetected dose commitment to bone per year as a function of half life in the pulmonary region or other metabolic pool and bioassay sampling frequency, assuming a detection level of 0.1 dis/min of ^{239}Pu per sample.

- When the quantity of unencapsulated plutonium handled in any separate laboratory or work area exceeds 1 mCi, an internal dosimetry program should be required.
- A routine internal dosimetry program should be considered even though work is exclusively with plutonium that is encapsulated, in a special form, in an approved shipping container, or in approved storage. (Approved storage refers to plutonium stored so that under normal conditions it is nondispersible; that is, the conditions of storage provide the requisite encapsulation.)

All individuals whose assignments involve work in an area where routine surveillance measurements are required should participate in the surveillance program. Visitors and other personnel whose assignments involve only inspection, observation or temporary tasks should be considered for limited participation in the program based on the probability and potential magnitude of internal exposure.

At least one 24-hour (true or simulated) urine sample should be required once a year. An annual in-vivo chest examination is suggested as good practice. Under certain conditions, more frequent examinations plus fecal samplings may be warranted.

4.4.3 Intake Indicators

The health physicist plays a key role in initiating prompt action at the site of an accidental intake and often must make decisions based upon limited information. Indicators that an intake has or may have occurred fall into two categories: incident details and diagnostic measurements. The possible significance of these indicators is reviewed in this section.

Incident Details

Information that is normally promptly available following an accidental intake and that may be indicative of the amount and type of intake includes:

- the level of contamination in the work area

- the skin contamination level, affected skin area, and integrity of the skin
- the chemical form of the material involved
- contamination levels at the site of the wound (taken either by direct measurement or by blood smears with field survey instruments)
- the contamination level on the object causing the wound or on protective apparel at the site of the wound (e.g., inside of gloves)
- data from air samples or continuous air monitors and the length of exposure
- nasal smear or nose-blow activity
- sputum and/or mouth contamination
- incident scenario.

If any of these indicators is positive, diagnostic measurements should be considered. Making meaningful estimates of the magnitude of the intake from these early data is difficult and usually requires experience in the facility and knowledge of numerous variables, including particle size, solubility, the type of operation and the facility design. In most instances, gross intakes are obvious from the initial incident details. However, in some instances the detection of significant intakes that should trigger additional action is less positive and a conservative attitude should be taken in interpreting incident details. Good practice suggests that where there is any doubt, diagnostic measurements should be scheduled not only for those directly involved in the incident but for those peripherally involved as well.

Diagnostic Measurements

Diagnostic, internal dosimetry measurements provide more meaningful data for assessments than do the incident details. Analytical results should be available within a few hours at those sites having an appropriate analytical capability for the following:

- single-void urine samples
- in-vivo examinations
- blood samples
- fecal samples.

Positive data from any of these may trigger additional action, including: continuation of the above analyses; arrangement for support analyses such as solubility, particle size and isotopic composition; and contact with medical staff regarding the initiation or continuation of therapeutic measures. Diagnostic evaluation techniques are discussed in the Appendix.

4.4.4 Prompt Assessments

Making prompt assessments of the magnitude of intake and the probability of rapid transfer to the blood from incident details and early diagnostic measurements is, as mentioned above, difficult and dependent upon many variables. Experience within a given facility or operation is perhaps the health physicist's biggest asset. In most instances, estimating the probable magnitude promptly helps medical personnel determine what therapeutic action should be taken.^(19,20) However, the making of valid semiquantitative assessments from early data is always questionable. From a practical viewpoint, early assessments are important in assuring appropriate treatment of those involved. However, in cases involving a significant deposition there is always pressure to provide early deposition estimates, which are then often interpreted as the actual deposition.

For this reason, and to assist medical staff in determining the course of treatment, some sites have developed guides that have been shown useful in making early estimates when limited to the operations for which they were developed and used only for the purpose intended. For other operations the health physicist should develop his own set of guides. Some sample guides and techniques are included below.

Assessment From Incident Details

Attempts to correlate incident details with the actual deposition have been futile. The most useful data appear to be those from nasal

smears or nose-blows;^(21,22) health physicists are beginning to recognize the value of these data not only for the identification of those who have inhaled radioactive particles, but also as a semiquantitative method of estimating the intake. The method is quick and effective if sufficient care is exercised in obtaining the specimen, as detailed on page 4-22.

The guide shown in Table 4.3 is based on 30 oxide inhalation cases. The correlations shown are valid within a factor of ± 5 . The relationships in these guides are derived from the nasal smear data, in-vivo lung measurements, bioassay data and the TGLM concepts. Extreme care was used in obtaining the data, especially that involving nasal smears, to preclude cross-contamination from the lip or nearby skin surfaces. Judgment must be used in applying this guide since details such as particle size, breathing habits and cross-contamination may cause a large variance in the correlation.

TABLE 4.3 Guide for Estimating Internal Deposition of Plutonium from Inhalation of Plutonium Oxide, Based on Nasal Smear Activity

<u>Nasal Smear</u>	<u>Initial Pulmonary Burden</u>	<u>Ultimate Systemic Burden</u>
1 nCi	1 nCi	0.1 nCi

Note: Guides should be used only for initial estimates, primarily as an aid in determining prompt mitigatory action.

Assessment From Special Measurements

Urine, Blood. Single-void urine samples are normally available shortly after an intake and data from a rush analysis may be applied to the Langham model⁽²³⁾ to provide some insight regarding the magnitude of deposition of plutonium, particularly in the more significant cases. These assessments should be used only as indicators since the estimate may be influenced by 1) dilution by the bladder's content prior to the intake,

2) analytical error, 3) extrapolation of the data to estimate the daily excretion rate, and 4) the time required for translocation. Correction for the effect of diethylenetriaminepentaacetic acid (DTPA) will also be necessary for those cases in which DTPA has been administered by the medical staff.⁽²⁴⁻²⁸⁾

Rush analysis of blood samples may also be useful in making prompt assessments of the magnitude of intake, but to date they have not helped in making meaningful evaluations of deposition. Research currently underway may provide more insight into the significance of the activity level in the blood. Both the blood and urine techniques (early samples) apply only to the readily transportable fraction of the intake and provide no information on the initially nontransportable fraction. Additional analyses of urine are extremely valuable in making interim as well as diagnostic evaluations.

Feces. Data from fecal samples are not normally available for several days and are therefore not useful in making prompt assessments. Peak activity levels may not appear in feces until the second and third day post-intake and may vary considerably from this norm with varying individual bowel habits.⁽²²⁾ Data from the first week post-intake is extremely valuable in making interim evaluations based on the TGLM. Supportive data (solubility, particle size and isotopic composition data), which are needed to apply lung model concepts, are normally available by the time fecal data for the first week are available. The analysis of fecal data by gamma spectroscopy provides prompt results.

In-Vivo Examinations. In-vivo examinations usually provide data promptly and are useful for making quick assessments. The data, however, are of little use in estimating how rapidly the contaminant will translocate to the blood unless data on particle size and solubility are also available. Examinations with a wound counter are useful as an indicator of the magnitude of intake and as a guide to the physician during excision. Sufficient data on relative rates of absorption into the blood stream are not available and techniques for estimating the quantity

translocated to the blood prior to and following excision are not known. Estimates of translocation from wounds are currently dependent upon the analysis of blood and urine.

Chest Examinations. Chest examinations made with appropriate in-vivo detection equipment provide good early estimates of intake and rate of translocation to extracellular fluids. Early chest count data may be applied to the TGLM to make these assessments. A knowledge of solubility and particle size are required for this assessment technique, and this information is not normally available promptly. However, solubility can be estimated based on a knowledge of the processes at the incident site. Studies of the particle size associated with plutonium releases in a specific operation or facility may identify a typical particle size that can be assumed for purposes of prompt assessment. For example, data obtained for the 30 oxide inhalation cases used to develop the nasal smear guides in Table 4.3 indicated a mean particle size of 4 μm activity median aerodynamic diameter (AMAD). The data in these cases were applied to chest examination data and led to the development of the guide given in Table 4.4.⁽²¹⁾ This guide is based upon TGLM concepts and has provided reasonable estimates when applied to several plutonium oxide inhalation cases.

TABLE 4.4 Guide for Estimating Internal Deposition of Plutonium from Inhalation of Plutonium Oxide, Based on In-Vivo Data

<u>Initial Measured Lung Burden</u>	<u>Initial Pulmonary Burden</u>	<u>Ultimate Systemic Burden</u>
1 nCi	0.3 nCi	0.03 nCi

Note: Guides should be used only for initial estimates, primarily as an aid in determining prompt mitigatory action. A particle size of 4 μm (AMAD) is assumed.

4.4.5 Data Collection

A variety of data are required for prompt, interim, and diagnostic evaluations of deposition. These include diagnostic analyses and measurements, plus supportive data. The health physicist plays a key role in the collection of these data, and must promptly initiate actions that will assure the collection and analysis of appropriate samples. In response to the trigger provided by incident details, spot urine samples and in-vivo examinations should be scheduled and special samples (nasal smears, skin and mouth smears, air samples, nasal irrigation, blood smears and/or other samples of the exposure material) should be collected and saved for special supportive analyses.

For those cases in which early data indicate an intake, consideration should be given to initiating urine sampling, in-vivo examinations, and sampling of the blood and feces (for inhalation cases) and making other supportive analyses as appropriate. The following schedule should be considered if warranted:

- Urine - For inhalation or injection cases, collect total urine for the first week to establish the trend for the readily transportable fraction. The collection frequency should be varied thereafter as necessary to evaluate the readily transportable and initially nontransportable fractions. Collection frequency for the latter will vary depending upon the transportability of the material involved. Total sampling may be necessary to determine the effectiveness of treatment during the period in which a chelate such as DTPA is being administered.
- Feces - For inhalation cases, collect fecal samples for the first week. For severe cases, consider extension of the collection period.
- Wound Counting - For injection cases, wound counting should be initiated promptly. Specialized probe detectors are useful in assisting the physician during the excision process. More

accurate measurements with a standard wound counter should be made upon the completion of excision and at appropriate intervals thereafter to obtain data on any plutonium left at the wound site. Excised tissue should be counted to help confirm excision effectiveness; the tissue should then be placed in formaldehyde for later special analyses.

- Chest Examinations - For possible inhalation cases, in-vivo chest examinations should also be made as soon as possible following the removal of skin contamination from chest and back. The interpretation of initial counts may be difficult because of small quantities of skin contamination or material already cleared to the stomach. Several examinations should be scheduled during the first 10 days so that early clearance mechanisms can be followed, and at lesser intervals thereafter for the determination of slower clearance times.

4.4.6 Mitigatory Action

Prompt and appropriate action following an accidental exposure to plutonium may be extremely important. Personnel decontamination in the field may prevent additional intake while therapeutic measures by medical staff may minimize the ultimate internal deposition.⁽²⁴⁻²⁸⁾ Promptness in initiating these early actions cannot be over-emphasized since their effectiveness decreases rapidly with time. Studies on animals have shown that up to 90% of monomeric plutonium that has reached the circulatory system may be decorporated if DTPA is administered within the first 1/2 hour. The effectiveness drops off rapidly thereafter. Most of the plutonium removed is from the blood and liver. That which reaches the bone before treatment is started is difficult to remove.

Mitigatory actions include both those administered by the health physicist and those administered by medical staff. The health physics and medical staff must establish plans of action, interact frequently to review those plans, and coordinate their actions promptly following accidents. Prompt contact is extremely important not only in mitigating deposition,

but also in establishing the proper course of action for the individual involved. For example, typical field actions that are normal for most individuals may not be appropriate for persons who have health problems or are severely affected by the accident (e.g., in shock).

Health Physics Actions

Mitigatory actions that are often delegated by the medical to the health physics staff in the field, so that they may be initiated sooner, include:

- skin decontamination
- water flush of wounds and chemical burns
- nasal irrigation.

Beyond these actions, the role of the health physicist is to alert medical personnel of possible intakes that may require action and to keep them informed of all analysis data, special measurements, and estimates of the degree of deposition and the probability of rapid transfer to the blood. The need for prompt attention to minimize intake and decrease the ultimate systemic deposition makes advisable a pre-established action guide that will trigger medical notification for therapeutic considerations. A typical guide for use in this purpose is shown in Table 4.5.

TABLE 4.5 Recommended Action Levels for Prompt Notification of Medical Staff for Therapeutic Considerations

Nasal Smear Total	5 nCi
Facial Contamination	20 nCi
Wound Contamination	
Field Survey Meter	5 nCi
Wound Counter	1 nCi
Chest Count	positive
Spot Urine (100 ml)	5 dis/min

Personnel Decontamination

Several points should be considered in any personnel decontamination effort:

- the seriousness of any injury that may be involved with the incident (burns, bleeding, shock, etc.)
- the chemical constituents of transuranic radionuclide solutions (acid, caustic, organic)
- the location and extent of contaminated areas, particularly those near body openings
- the condition of the skin at and around contaminated areas.

Medical attention for a serious injury must take precedence over contamination removal. However, if very high levels of contamination are present, those levels should be reduced at the first opportunity. If only plutonium radionuclides are involved, the radiation dose from the external contamination will not be of primary concern. However, if mixtures include significant quantities of ^{241}Am , external exposure must be considered.

Personnel Decontamination Kit. A personnel decontamination kit should be stored in a suitable location near the work area (the change room at the air lock entrance to the process area is a good location for a personnel decontamination station). The kit should be labeled, fully supplied, and sealed, and its contents should be checked and replaced as necessary before resealing after each use. In addition to the kit, a source of water, 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 provided.

Instruments to detect and locate skin contamination should be stored in the personnel decontamination area or should be immediately available. Telephone service at the station can also be helpful for obtaining assistance and reporting contamination status.

A well-supplied personnel decontamination kit should contain:

Cotton-tipped applicators	Mild detergent containing lanolin and antiseptic
Cleansing tissue	Potassium permanganate
Cotton balls or rolls	Sodium bisulfite
Flushing tube	Scissors
Gauze sponges	Surgical gloves
Hand brush	Tongue depressors
Abrasive soap	Tourniquet kit ^(a)
Masking tape	Water
Na ₄ EDTA solution	Waxed bags (for waste disposal)
Nasal swabs (1- by 2-in. or 2.5- by 5-cm filter paper strips)	Hand cream
Paper cups (1 and 4 oz or 29.6 and 118.4 ml)	

^(a)Opinions diverge widely on the safety of nonmedical personnel using tourniquets. They may or may not be included, depending on local opinions relating to their use.

Some of the chemical reagents used for skin decontamination should be prepared to the following specifications:

- Na₄EDTA - A stock solution may be prepared in advance and kept available in the kit. Dissolve 16 g of 70% active Na₄EDTA chemical salt in 100 ml of distilled water for a solution of 10% active ingredient (EDTA).
- KMnO₄ - A saturated solution of potassium permanganate is prepared by dissolving crystals in water. The solution is usable as long as the crystals remain dissolved. If crystals are allowed to form there is danger of scratching the skin when the KMnO₄ is applied.
- NaHSO₃ - Sodium bisulfite should be prepared in a 4% solution just prior to use because its potency is lost when the solution is left standing in air.

Decontamination Procedure. Simple scrubbing techniques can usually be used to remove localized skin contamination. The following steps are recommended.

- Isolate the contaminated area by a careful survey and examine the skin for breaks. If an injury or skin break is detected in the contaminated area, contact the medical staff for any special instructions.

- Wipe off any loose contamination with a gauze sponge or cotton applicators dipped in mild antiseptic detergent. Use care not to spread the contamination to noncontaminated areas.
- Rub the skin with the applicators to produce good sudsing.
- Dry the skin area with cleansing tissue. After the skin is thoroughly dry, survey it for any remaining contamination.
- If no contamination is detected, apply a good quality hand cream to prevent chapping.

If the first attempt is unsuccessful, then apply an abrasive soap with a moist gauze sponge or soft hand brush and rub to develop good sudsing, being careful not to break or damage the skin surface. If contamination still persists,

- Prepare a solution of 10% Na_4EDTA plus a detergent.
- Follow the application, removal and survey techniques described above.

If contamination still persists,

- Prepare solutions of KMnO_4 and NaHSO_3 .
- Using cotton-tipped applicators, paint the contaminated area with KMnO_4 . Allow the solution to dry and paint again two or three times more, allowing the solution to dry thoroughly between each application. The skin will appear almost black. Discard the applicators after each use to avoid spreading contamination to the solutions.
- Rub the treated area with NaHSO_3 using cotton applicators until the brown discoloration is removed, then rinse with water to remove the remaining NaHSO_3 . Dry the area thoroughly and survey it for contamination.

If contamination still persists, a physician should be consulted to direct further decontamination efforts.

For eye, nose, or mouth contamination, irrigation with tepid water or saline solution is the usual procedure. Under no circumstances should this treatment be undertaken by personnel who have not been thoroughly trained, and a physician should always be consulted for special instructions. Agents

other than tap water or normal saline solution should be used only upon the direction of a physician. (See the next subsection for details on the nasal irrigation procedure.)

Cuts, wounds, punctures, and abrasions in areas contaminated by transuranic radionuclides require special monitoring and treatment to determine their contamination status.

Nasal Irrigation

The preferred treatment for nasal contamination is with normal saline solution, which should be provided at each permanent personnel decontamination station. Nasal irrigations may be done as emergency procedure by trained but nonmedical personnel. Physicians or nurses may be called upon to carry out the procedure, but it is important to start the irrigation as soon as possible.

In any case where caustic solutions may have gotten into the nose or mouth, the rinsing should be done at once at the nearest available source of sanitary water.

Equipment and Reagents. The following supplies are required for nasal irrigation:

- Irrigator stand
- Irrigator bottle
- Rubber tubing (6 feet or 1.8 meters long)
- Pinch clamp
- Glass stem from nose dropper
- Patient's throw (plastic apron)
- Small-diameter rubber tubing (2 in. or 5.1 cm long)
- One-gallon (3.8-liter) ice cream carton
- Normal saline solution
- Several disposable cups
- Swabs
- Rubber gloves
- Paper bags

Nasal Irrigation Procedure. For nasal irrigation, rubber tubing only should be inserted into the nose. Glass apparatus should be inserted only by members of the medical staff. For irrigation using rubber tubing, these steps should be followed:

- Seat the patient.
- Have the patient take nasal smears, and save them.
- Have the patient put on a plastic apron or other covering to protect the clothing from liquids.
- Prepare nasal irrigation equipment as follows:
 - a. Position the irrigator bottle filled with normal saline solution directly above the patient's head.
 - b. Place a clean rubber tip of small diameter tubing on the end of the glass stem so that it extends 1/2 in. (1.2 cm) beyond the glass.
- Have the patient blow his nose into a tissue, then clear his throat and spit into a disposable cup. Save both samples.
- Place the gallon ice cream carton in the patient's lap to collect the irrigation water.
- Have the patient insert the rubber tip into the lower portion of his nostril (about 1/2 to 3/4 in. or 1.2 to 1.9 cm).
- Make sure the patient's head is held erect and over the ice cream carton so that the water will run into the lower part of the nostril and drain back out the same nostril. If head is tilted forward the water may run into the sinuses.
- Allow the patient to regulate the flow by pinching the tube with his fingers.
- Allow one-half irrigation bottle of saline solution to each nostril.
- Retain the solution for analysis.
- Have the patient blow his nose and clear his throat again, and save the specimens.

- Have the patient take new smears of each nostril; dry and count them. If positive, repeat the nasal irrigation procedure.

Medical Actions

Remedial action and therapeutic measures such as excision, special skin decontamination and the administration of lung irritants, DTPA, and agents to increase peristalsis are the responsibility of the medical staff.

4.4.7 Body Burdens and Derived Air Concentrations

The permissible air concentration of plutonium to which workers may be exposed has been given several names - maximum permissible concentration in air (MPC_a), derived air concentrations (DAC), or concentration guides (CG's).

ERDA Guidelines

The concentration of airborne plutonium in a facility is limited to the values given in "Standards for Radiation Protection," ERDA Appendix 0524 in Annex A, Table I.⁽²⁹⁾ In the ERDA Appendix, these values are referred to as CG's. ERDA Appendix 0524 does not consider body burdens.

ICRP Guidelines

Table 4.6 contains the maximum permissible body burdens and DAC or MPC_a for plutonium isotopes as recommended by the ICRP and contained in IAEA Safety Series No. 39.⁽³⁰⁾ The DAC values for the plutonium isotopes listed are identical to the CG values given in ERDA Appendix 0524. They were calculated by ICRP Committee II assuming that 45% of the plutonium would be deposited in the bone, 45% would go to the liver, and 10% would be excreted. The body organ burdens and DAC values were calculated assuming an occupational exposure limit of 5 rem/yr to the body, head, gonads, red bone marrow, and active blood-forming organs, and 15 rem/yr to other organs and organ systems. The critical organs are listed in Table 4.6 for soluble (e.g., nitrate) and insoluble (e.g., oxide and fluoride) forms of plutonium.

TABLE 4.6. Maximum Permissible Body Burden and Derived Air Concentration (DAC) Limits for Plutonium Isotopes

<u>Isotope</u>		<u>Critical Organ</u>	<u>Maximum Permissible Body Burden</u>		<u>Derived Air Concentration for 40-hr Week</u>
			μCi	μg	$\mu\text{Ci}/\text{cm}^3$
^{238}Pu	sol.	Bone	0.04	0.0023	2×10^{-12}
	insol.	Lung	$0.015^{(a)}$	0.00086	3×10^{-11}
^{239}Pu	sol.	Bone	0.04	0.65	2×10^{-12}
	insol.	Lung	$0.016^{(a)}$	0.27	4×10^{-11}
^{240}Pu	sol.	Bone	0.04	0.17	2×10^{-12}
	insol.	Lung	$0.016^{(a)}$	0.06	4×10^{-11}
^{241}Pu	sol.	Bone	0.9	0.008	9×10^{-11}
	insol.	Lung	$0.016^{(a)}$	0.00014	4×10^{-8}
^{242}Pu	sol.	Bone	0.05	12.5	2×10^{-12}
	insol.	Lung	$0.016^{(a)}$	4.0	4×10^{-11}
^{243}Pu	sol.	GI(ULI) ^(b)			2×10^{-6}
	insol.	GI(ULI) ^(b)			2×10^{-6}
^{244}Pu	sol.	Bone	0.04	0.002	2×10^{-12}
	insol.	Lung	0.017	0.0009	3×10^{-11}

(a) The figures for the insoluble form in the lung have been taken from the IAEA Safety Series No. 9, Basic Safety Standards for Radiation Protection.

(b) GI = gastrointestinal tract, ULI = upper large intestine.

4.5 EXTERNAL EXPOSURE CONTROL

Measuring the external exposure and resultant dose or dose equivalent for personnel involved with plutonium production is among the most difficult dosimetry problems because of the complexity of the radiations 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 sensitivity of most dosimeters make the interpretation of dosimeter results extremely dependent on adequate 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, as is the presence or absence of any shielding material. 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, and the neutron shield thickness for glove boxes is limited to about 6 in. (15 cm).

In this section guidance is provided which, if used properly, will enable the practicing health physicist to assess the situation and will provide assistance in controlling external exposure to a reasonable level.

4.5.1 Dose Rates

Photon Dose Rates

One of the problems in handling plutonium oxide in a glove box is the contact handling of the bare material with some type of glove. Because of the intensity (yields) of low-energy x-rays, the photon dose rates through gloves can be very high. Roesch and Faust have devised a formula for predicting the surface dose rate of plutonium through a 100 mg/cm² shield. The latest version of the equation^(a) is:

$$D_s(\text{rad/h}) = 171f_{238} + 0.51f_{239} + 2.4f_{240} + 8.7f_{241} \\ + 0.15f_{242} + (0.074f_{241})t \quad (1)$$

where D_s = the surface dose rate of plutonium dioxide or plutonium metal (rad/hr),

(a) Faust, L. G. et al., Summary Report on Plutonium Dosimetry. BNWL-1887, revised - to be published.

f_i = the weight fraction of the i^{th} isotope of plutonium, and
 t = the time since chemical separation (days).

This equation is valid for sources that are "infinitely thick" (about 1/4 in. or 0.63 cm for PuO_2 powder, or 0.040 in. or 0.1 cm for Pu metal). Note that a term in this equation involves time. This term accounts for the build-up of ^{241}Am , which comes from the decay of ^{241}Pu . Americium-241 is important because of its high yield of 60-keV photons and x-rays, which readily penetrate some hood gloves.

If the PuO_2 is mixed with uranium oxide, the gamma dose rate is reduced by self-shielding. Measurements made with 12% and 19% ^{240}Pu in mixed oxide samples from 2% PuO_2 to 50% PuO_2 in mixed oxide indicate a linear relationship between gamma surface dose rates and PuO_2 concentration:

$$D_{\text{MO}_2} = D_{\text{PuO}_2} x + 0.1 \text{ rad/hr} \quad (2)$$

where D_{MO_2} = the surface dose rate from mixed oxide (rad/hr),

D_{PuO_2} = the surface dose rate from PuO_2 (rad/hr), and

x = the weight fraction of PuO_2 in mixed oxide.

This equation has been verified for both new PuO_2 and old PuO_2 samples in which a large fraction of the photon dose is due to ^{241}Am .

An equation similar to Equation(1) has been derived for a lead-loaded rubber glove using the attenuation properties calculated by the computer code PUSHLD.⁽³¹⁾ The lead-loaded neoprene glove is nominally 0.076 cm thick and contains 0.1 cm of lead equivalent. The surface dose rate, D_{PbG1} , is given by:

$$D_{\text{PbG1}}(t) = 2.83f_{238}e^{-0.00789t} + 0.104f_{239} + 0.0315f_{240} + 6.35 \times 10^{-5}f_{242} + f_{241}(158.5e^{-0.0016t} - 152.5e^{-0.0457t}) \quad (3)$$

where $D_{\text{PbG1}}(t)$ = the surface dose rate (rad/hr),

f = the weight fraction of the particular plutonium isotope, and

t = the time since chemical separation of the plutonium (years).

This equation includes only the radiations from plutonium, and the ^{237}U and ^{241}Am daughters from the decay of ^{241}Pu . The expression is valid for times between 50 days and several hundred years after chemical separation of the plutonium. The formula gives results that are within +0% and -20% of PUSHLD calculations.

Neutron Dose Equivalent Rates

The neutron dose is of great importance in any glovebox operation involving 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 multiplication from loose PuO_2 powder; induced fission seems to be a problem only in plutonium metal or very large, high-density arrays of plutonium oxide.

Most neutrons from spontaneous fission originate from the isotopes ^{238}Pu , ^{240}Pu and ^{242}Pu , and in PuO_2 neutrons result from $^{17}\text{O}(\alpha, n)$ and $^{18}\text{O}(\alpha, n)$ reactions. Table 4.7 gives the theoretical yields from various isotopes and compounds. The approximate neutron yield from a plutonium compound (Y in neutrons/sec) can be found by summing the products of the weight fraction of the isotope (W_i) times the spontaneous fission (Q_{si}) and (α, n) yields ($Q_{\alpha i}$) for that particular isotope as listed in Table 4.7.

$$Y = \sum_i W_i (Q_{si} + Q_{\alpha i}) \quad (4)$$

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/sec) can be found from the data in Table 4.8 and the formula:

$$Y_{\text{imp}} = A_{\alpha} \sum_j P_j I_j \quad (5)$$

where $A_{\alpha} = \sum_i A_i$ = the total alpha activity from all alpha emitters, including ^{241}Am ingrowth from ^{241}Pu ,

P_j = the neutron yield from j^{th} low Z element (neutrons/ α -ppm), and

I_j = the impurity concentration in plutonium (ppm).

TABLE 4.7. Neutron Yield from Plutonium and Uranium Compounds

Nuclide	Half Life, years	Alpha Activity, n/sec-g of isotope	Neutron Yields, Q		
			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
$^{241}\text{Pu}^{(c)}$	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
$^{241}\text{Am}^{(d)}$	4.33×10^2	1.27×10^{11}	1.6	2.7×10^3	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 $f_{\text{Am}} = 1.035 f_{\text{Pu}} (e^{-0.0016t} - e^{-0.04625t})$, 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. Some data from ANL-6540.

(f) Estimated for ^{232}U alone; increase by a factor of 5 if in equilibrium with its daughters.

(g) From Thomas E. Sampson, "Neutron Yields from Uranium Isotopes in Uranium Hexafluoride," Nuclear Science and Engineering, 54: 470, August 1974.

TABLE 4.8. Approximate Neutron Yields of
Low Atomic Number Impurities
in Plutonium

Impurity Element	(α, n) Neutron Yield n/α , ppm
Li	6.3×10^{-12}
Be	2.0×10^{-10}
B	4.6×10^{-11}
C	2.8×10^{-13}
O	1.6×10^{-13}
F	2.4×10^{-11}
Na	3.0×10^{-12}
Mg	2.7×10^{-12}
Al	1.5×10^{-12}
Si	3.3×10^{-13}

Measurements of the neutron yields of more than 100 samples of plutonium and plutonium compounds using a precision long counter have shown that nuclear-fuel grade plutonium metal usually has a neutron yield at least 20% greater than the calculated value, probably because of light-element impurities in the parts-per-million range and, possibly, some neutron multiplication. Nuclear-fuels grade PuO_2 usually has a measured neutron yield about 10% greater than the calculated value, caused by (α, n) reactions from such impurities.

In mixtures of plutonium compounds with other compounds, the neutron yield is about the same as that from the pure plutonium compound. In PuO_2 - UO_2 mixtures, the size of the PuO_2 particles is usually large enough so that α particles from the plutonium lose most of their energy inside the PuO_2 particle and the $\text{O}(\alpha, n)$ yield is, therefore, essentially the same as that from pure PuO_2 .

An appreciable neutron dose usually comes from a large amount of plutonium. Large masses of plutonium can generally be treated as a point source of neutrons located at the center of the mass. The neutron dose equivalent

rate H (in mrem/hr) at a distance r (in cm) from the center of a large mass of plutonium, plutonium oxide, or mixed oxides can be found from the formula:⁽³²⁾

$$H = 0.0097 \frac{S}{r^2} \quad (6)$$

where S, the total neutron emission rate, is equal to the product of the mass of the plutonium (grams) times Y, the total neutron yield per gram of plutonium (neutrons/sec-gram) from spontaneous fission, (α ,n) reactions and fission-induced neutrons.

4.5.2 Shielding Materials

The types and thicknesses of shield required to obtain a specific dose rate are a function of the quantity of plutonium present, the type of operation, occupancy factors and the chemical and physical properties of the plutonium. The following data, with the equations in the previous section, can be used to determine what exposures can be expected and what type and thickness of shield will be necessary to achieve a desired dose rate. The results presented are estimates only; if more refined results are desired a detailed calculation must be performed.

Photon Shields

The attenuation of photon dose rates of a finite thickness of material is highly dependent on the isotopic composition of the plutonium. Generally, the greater the content of ²³⁹Pu, the less the attenuation per unit thickness of material; however, the total dose rate is normally lower. Tables 4.9 and 4.10 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 ANISN⁽³³⁾ and BMC.⁽³⁴⁾ However, for completeness some data are included here that give dose reductions as a function of several shield materials and thicknesses. These data can be used with the equations of Section 4.5.1 to estimate the neutron dose.

TABLE 4.9 PuO₂ Sources

Shield Material	Thickness, Inches	Attenuation Factors ^(a) (rad/hr) for:	
		Source A	Source B
None		4.5	1.5
Stainless Steel	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	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	0.071
	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.

Figure 4.2 shows 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 PuO_2 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 $\text{O}(\alpha, \text{n})$ reactions with an average energy of about 2.3 MeV. These results indicate that PuO_2 can be simulated by ^{252}Cf sources.

Figure 4.3 shows a set of shield-effectiveness measurements for a PuF_4 neutron source with an average neutron energy of 1.3 MeV, and 1-ft by 2-ft slab shields. The data points in this graph give an idea of the precision that can be achieved with the TEPC. In this case, polyethylene is 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 from plutonium compounds and may, in fact, reduce the effectiveness of thin shields by reducing the hydrogen density.

Figure 4.4 shows the results of shielding measurements using ^{252}Cf to simulate PuO_2 , and 2-ft by 2-ft slab shields of various materials. The curve for polyethylene can be used for ordinary polyethylene, high density linear polyethylene, polypropylene, Ricorad or Ricorad with 1% B_4C . The curve for Optigel can be substituted for water or some forms of water-extended polyester (WEP) resin. The curves for concrete show the tremendous variability of concrete neutron shields; the best concrete neutron shields have higher hydrogen densities, usually in the form of water of hydration. The 10-year concrete is typical Hanford aggregate used for the walls of a building.

The results of measurements shown in Figures 4.5 and 4.6 are for PuF_4 and ^{252}Cf , respectively, with 2-ft by 2-ft slab shields. The Optigel, boroplaster, water-extended polyester (WEP-55) and mineral oil results have been corrected to remove the effect of the 1/4-in. (0.6-cm) wall thickness of lucite needed to contain them.

FIGURE 4.3. PuF_4 Neutron Source with Slab Shields

FIGURE 4.4. PuO_2 Relative Neutron Dose Reductions

TABLE 4.10 Isotopic Composition of PuO₂ Sources Used in Table 4.9

	Isotopic Concentration, weight percent					
	<u>²³⁶Pu</u>	<u>²³⁸Pu</u>	<u>²³⁹Pu</u>	<u>²⁴⁰Pu</u>	<u>²⁴¹Pu</u>	<u>²⁴²Pu</u>
Source A (2.9 yr since separation)	4.5 x 10 ⁻⁶	1.92	63.3	19.2	11.7	3.88
Source B (5.0 yr since separation)	9.1 x 10 ⁻⁷	0.399	70.0	21.6	6.31	1.73

The dose reduction data shown in Figures 4-1 through 4-6 are based on measured values using PuF₄, PuO₂, and ²⁵²Cf sources. Data showing the quality of the "fit" between PuO₂ and ²⁵²Cf are also given so that the latter can be used in place of PuO₂ for neutron studies.

All neutron dose rate (i.e., kerma) 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 ²⁵²Cf spontaneous-fission neutron source (with an average neutron energy of 2.3 MeV) shielded by polyethylene. The circles indicate TEPC measurements made with 1-ft by 2-ft (0.3-m by 0.6-m) slab shields. The triangles indicate TEPC measurements with spherical shields of polyethylene that match the one-dimensional ANISN computer code geometry. The crosses indicate the relative reduction in kerma, calculated from fluence values using the computer code ANISN and the fluence-to-kerma factors of Ritts, et al.⁽³⁵⁾ ANISN is a one-dimensional, discrete ordinate transport code that yields results in good agreement with the values measured by the TEPC.

FIGURE 4.1. ^{252}Cf Fission Source with Polyethylene Shields

FIGURE 4.2. Dose Reduction with Polyethylene Slab Shields

FIGURE 4.5. Relative Neutron Dose Rate Reduction with Slab Shields and a PuF_4 Source

FIGURE 4.6. Relative Neutron Dose Rate Reduction with Slab Shields and a ^{252}Cf Source

Since 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 $N(n,\gamma)$ reactions. This difference is most apparent for thick shields with a large number of thermal neutrons.

4.5.3 Example of Dose Rate Calculations from Plutonium in Glove Box

The formulas and dosimetry data described in the above sections can be applied to estimate the dose rate to a worker handling plutonium in a glove box. As a hypothetical example, assume that the worker is handling 1.131 kg of PuO_2 (1.00 kg of Pu) with the isotopic compositions specified in Table 4.11 (the expected average compositions of light water power reactors (LWRs) producing plutonium in 1975 and 1980). The plutonium is contained in a steel can 4.45 cm in diameter and 9.59 cm high with 0.51-cm (0.020-in.) thick walls. The PuO_2 is in the form of a loose powder with a density of 1.9 g/cm^3 .

TABLE 4.11. Isotopic Composition of LWR-Produced Plutonium

Year	Pu Isotope, wt%					
	^{236}Pu	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
1975	5×10^{-6}	0.9	65.6	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

Three locations are of interest: 1) the surface of the can where the worker's hand contacts it; 2) approximately 15 cm (6 in.) away from the can where the dose to the forearm is important; and 3) about 36 cm (14 in.) away from the can where the dose to the body is important. Fourteen inches is a convenient working distance for an operator leaning against the side of a glove box with his arms in the gloves. The doses at these points can be compared with the design criteria for ERDA plutonium facilities, which are contained in the section on radiation protection in the Appendix of ERDAM-6301.⁽³⁶⁾ For purposes of designing shielding and equipment layout,

ERDA's upper limit for exposure to the whole body is 1 rem/yr, which is equivalent to an average dose-equivalent rate of 0.5 mrem/hr at the work station. A remote shielded operation must be utilized where exposures to hands and forearms are likely to approach 30 rem and 15 rem per year, respectively. The limits on dose equivalent for design purposes are thus:

Whole body	1 rem/yr
Forearm	15 rem/yr
Hands	30 rem/yr

Equations (1), (2) and (3) (see pp. 4-42 and 4-43) were used to calculate the photon dose rates at the three locations of interest 1 yr after chemical separation of the plutonium. The neutron dose-equivalent rates were estimated from Equation (6) (p. 4-47), with neutron yields of 3510 n/sec-kg for 1975 LWR-PuO₂, and 4540 n/sec-kg for 1980 LWR-PuO₂ at 1 yr 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/hr per n/cm²-sec 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) neoprene gloves are used. For PuO₂ 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. 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) 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. thick (0.32-, 0.64-, and 1.3-cm) 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 reasonably effective as a neutron shield, fireproof, and transparent. Table 4.12 shows the results of these calculations.

TABLE 4.12. Dose Equivalent Rates from Handling a Can Containing 1.13 kg of PuO₂ (1.0 kg Pu) in a Glove Box (a)

	1975 LWR PuO ₂				1980 LWR PuO ₂				
	Dose Equivalent Rates, mrem/hr			Time to Reach Annual Limit(b)	Dose Equivalent Rates, mrem/hr			Time to Reach Annual Limit(b)	
	Gamma	Neutron	Total		Gamma	Neutron	Total		
<u>Surface of Glove Touching 0.020 Inch Thick Steel Can with PuO₂</u>									
0.033-inch PVC plastic or neoprene glove	1600	160	1800	17	hr	2000	210	2200	13.5 hr
0.030-inch thick lead-loaded glove	980	160	1150	26	hr	1200	210	1400	21 hr
0.060-inch thick heavy lead-loaded glove	400	160	560	53	hr	490	210	700	42 hr
<u>Forearm at 6 Inches from Can Containing PuO₂</u>									
0.033-inch PVC plastic or neoprene glove	87	13	100	5	wk	110	17	126	4 wk
0.030-inch thick lead-loaded glove	71	13	83	6	wk	89	17	105	4.8 wk
0.060-inch thick heavy lead-loaded glove	30	13	43	12	wk	38	17	54	9.3 wk
<u>Outside of Glove Box with Surface of Can 14 Inches from Wall</u>									
1/4-inch steel	5.3	3.1	8.4	4	wk	6.6	4.1	10.7	3.1 wk
1/8-inch lead	1.3	3.1	4.5	7.4	wk	1.6	4.1	5.7	6 wk
1/4-inch lead	0.63	3.1	3.8	9	wk	0.77	4.1	4.8	7 wk
1/2-inch lead	0.22	3.1	3.4	10	wk	0.29	4.1	4.4	7.6 wk
1/2-inch lead + 4 inches of Optigel in lucite box	0.22	0.81	1.0	33	wk	0.29	1.1	1.4	24 wk
1/2-inch lead + 6 inches 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 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. (14) It is also assumed that the worker is actually at the glove box 6 hr/day or 30 hr/wk (assuming 2 hr/day for changing clothes, lunch, coffee and smoke breaks, etc.).

(c) Below design exposure limit.

Several conclusions can be drawn from these data. Since no worker would stand holding a can of PuO_2 in his hand, the data in this table represent an extreme case. The worker can actually have his hands in contact with the can of PuO_2 from about 14 to 50 hr before he receives 30 rem to the hands. If 1.31 kg of PuO_2 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 hr/day, with 2 hr/day spent for lunch, coffee and smoke breaks, changing protective clothing, etc.

Restricting the amount of plutonium near the operator in the glove box or placing neutron shields in the floor of the glove box would probably be more effective than putting massive 6-in. 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 even small amounts of shielding are used. The dose limit of 15 rem/yr to the forearms does not appear to be as restrictive as the 30 rem/yr limit to the hands.

4.5.4 Other Considerations

The example given in Section 4.5.3 illustrates the effects of shields in maintaining exposure control. However, shielding by itself is not adequate and other means of control must be considered, among them housekeeping, inventory controls, occupancy factors, etc. For instance, exposure rates as high as 30 mR/hr inside neoprene gloves and 10 mR/hr inside 30-ml lead-loaded rubber gloves have been measured inside a glove box that was used to process high-exposure plutonium powder (12 wt% ^{240}Pu , 6% ^{241}Pu). 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 most

dust being on the region by the hand). At 10 mR/hr, a person would approach the annual limit of 15 rem to the forearm in about 50 weeks with only the PuO_2 dust on the leaded rubber glove as the source. This fact illustrates the necessity of cleaning up the interior surface of the glove box, including the gloves, at least weekly and keeping to a minimum operations that generate dust. A filtered vacuum system may be desirable to vacuum dust from the gloves; a vacuum cleaner inside the glove box is not recommended because its exhaust air blows too much dust around, and the vacuum cleaner itself may be a fire hazard.

The plutonium should be processed as quickly as possible after it is chemically separated. 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 build-up of daughters, notably ^{237}U and ^{241}Am from the decay of ^{241}Pu . The dose rate increases rapidly from the build-up of ^{237}U in the first 50 days, then increases more gradually from the build-up of ^{241}Am and other daughters. Plutonium that contains more than about 5% ^{241}Pu and is several years old should perhaps be reprocessed to remove the ^{241}Am before processing the plutonium.

4.6 QUALITY CONTROL

All critical safety and contamination control systems must be routinely inspected and tested for proper performance. The primary confinement system and the ventilation system require particular attention.

4.6.1 Glove Integrity Inspection

Operating personnel should survey hood gloves frequently. In addition, the gloves in glove boxes should be inspected weekly for cracks, abrasions or any deterioration of the surface caused by chemical agents within the box. The box glove inspections should be made each time by the same team of trained individuals and the condition of each glove should be recorded so that glove failures can be anticipated and precautionary measures taken. The surgeon's gloves should, of course, be surveyed after the inspection of each box glove. Box gloves in questionable condition should be changed without delay.

4.6.2 Ventilation System Tests

The initial instrumentation test of the ventilation systems for the primary confinement system and the process area rooms should be instrumented to permit continuous monitoring of flows and pressure differentials. Since the ventilation systems probably will not be instrumented well enough so that localized failures or off-standard conditions can be detected with certainty, a schedule of inspections and tests should be established to assure that each element of the system is operating properly.

All HEPA filter banks should be tested for leakage at least annually using the cold DOP method. In many cases, the first HEPA exhaust filter is located directly on the glove box to prevent the contamination of exhaust ducts. Because DOP testing of these filters is often not practical, pressure drops across the filter should be continuously monitored, and the filter should be changed periodically. The downstream HEPA filter stages should be equipped with fittings to permit the convenient injection of the DOP aerosol and the measurement of upstream

and downstream concentrations. The test should be performed in accordance with ANSI Standard N101.1-1972, "Efficiency Testing of Air Cleaning Processes Containing Devices for Removing of Particles (N46)."(37)

If open-faced hoods are employed, air velocity measurements should be made quarterly until sufficient data is obtained to demonstrate the reliable performance of the exhaust system and stable adequate air velocities through the opening. The tests may then be performed less frequently but must be made at least annually. The linear air velocity should be maintained at 150 ± 25 ft/min (45.7 ± 7.6 m/min). This velocity is required to assure the confinement of radioactive material in the hood in spite of eddy currents caused by the movement of personnel and objects through the opening. If the hood has an adjustable opening and no means are provided to maintain a constant air velocity over the range of the opening, mechanical stops should be provided to limit the opening as necessary so that velocities will be kept within the required range. The results of all air velocity tests should be kept on record.

4.6.3 Emergency Power Tests for Critical Systems

The operation of critical building systems on emergency power should be tested at least annually. The test must be planned carefully to assure that contamination control is not lost during testing. It should be long enough to confirm that all critical systems are operating as planned. The position of all automatic dampers should be determined and data on air flow and pressure differential should be obtained and recorded. No loss of air flow or pressure differentials should occur during the transfer to and from emergency power. The operation of all monitoring and alarm systems on emergency power should be confirmed. In addition, the onsite emergency power source should be tested at least monthly.

4.6.4 Calibration and Testing of Critical Systems

The calibration and testing of critical systems is extremely important to the overall radiation protection program but has often been

neglected. Effluent monitoring and sampling systems and remote area monitoring systems (RAMS) should be given several tests. The instrumentation portion of the systems should initially be given a full range evaluation of all radiological, environmental and mechanical characteristics to assure compatibility with performance criteria and operating parameters. The effluent sampling and monitoring system should be tested once it is in place to determine sample losses from the sample probe to the collector/detector. This test need not be repeated unless a significant change in the sample equipment is made. However, the sample probe should be examined once a year to assure that its design has not been changed by corrosion. The recorder of the sample flow rate should be calibrated when installed and annually thereafter. The operability of the overall system should be completely tested once, with repeat tests only after modification, repair or maintenance. Operability checks should be scheduled monthly.

The operation of emergency signal systems, including criticality detectors and fire systems, should be checked periodically to assure that the alarms are audible at all potentially occupied locations. To prevent any desensitizing of staff response, the staff should be aware that the tests will be performed. Where possible, tests should be scheduled during off-shift hours to minimize personnel exposure to the testing. Building systems should be tested semiannually and the area-wide system should be tested at least annually. Any portion of the detector/alarm system that is disturbed by the test must be reconfirmed for operability after the test is completed. For example, if a detector is disconnected and a signal injected at that point, the detector should be tested after it has been reconnected.

4.6.5 Administrative Procedure Audits and Inspections

The radiation protection program should be audited at least annually to determine its adequacy in meeting both operations and monitoring requirements. Audits and inspections should be conducted at least quarterly to check that the performance of operations and monitoring are

in accord with the established program. One of these quarterly audits or inspections should be formal. The program for follow-up on deficiencies identified during audits and inspections should include an automatic review every 30 days to assure that deficiencies have been corrected.

4.6.6 Emergency Procedure Exercises

At least two emergency procedure exercises per year should be scheduled in plutonium facilities. One of these should be an actual performance exercise during which the facility is emptied following the prescribed route of evacuation after the appropriate warning sounds on the alarm system. The other exercise may be a test of staff knowledge rather than an actual physical test. During this exercise signals and responses should be reviewed and individual actions discussed. Records of the personnel who have participated in these emergency exercises should be kept.

4.6.7 Dosimeter Assignment

An important part of any radiation protection program involves adequate measurements and records of radiation exposure. The types of radiation present in a process area, and the process itself, indicate the type of dosimeter needed to insure adequate measurements, which then become part of the individual's exposure record.

Of the many types of personnel dosimeters using many materials for measurements, the TLD is recommended here. The most important reason for this recommendation is that existing TLDs for both whole-body and extremity measurements perform better than film-based dosimeters in the presence of the radiations from plutonium.

Whole-Body Dosimeters

A dosimeter capable of measuring beta, gamma and neutron exposures should be assigned to every individual who routinely works in the process area. For those personnel whose total "stay time" is less than 1 or 2 hours per week, a dosimeter for measuring gamma exposures only is probably adequate. The exchange period should be monthly to quarterly, based on facility experiences.

Extremity Dosimeters

Extremity dosimeters must be used by all personnel who come into contact with plutonium. These dosimeters are usually gamma sensitive only and are worn on the hands or fingers. Their exchange period is normally monthly but can be more or less depending on the process and any experiences 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. The use of neutron-to-gamma ratios should not be condoned without supporting evidence that such ratios are reasonably constant.

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EMERGENCY PREPAREDNESS

SECTION 5
EMERGENCY PREPAREDNESS

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SECTION 5

EMERGENCY PREPAREDNESS

A plutonium facility must not only be prepared to cope with emergencies such as fire and chemical explosions that can occur in any type of building, but must also be prepared to cope with any emergency that involves plutonium or other radioactive material. The toxic nature of plutonium warrants special considerations for emergency preparedness.

The following general approach is necessary to determine the emergency posture for the facility:

1. postulating types of events that are feasible based on functions or processes, supporting systems, structures, personnel, and inventory at risk;
2. evaluating the interactions of the feasible events either individually or in combinations;
3. identifying the more feasible events together with resultant source terms;
4. developing an emergency preparedness program based on these analyses.

No specific ERDA or NRC guidance has been provided on possible accidents or on source terms to be used in developing plans. Several studies^(1,2,3) have addressed these problems generally. Section 5.1 describes briefly a typical approach used to develop source terms; however, analysis should be based on a specific facility and its site characteristics. In the development of the final emergency action plan concerning offsite releases, that specific plan should be jointly developed with federal and state agencies for the geographical area involved.

5.1 ACCIDENT CHARACTERISTICS AND SOURCE TERMS

Accidents in mixed oxide fuel fabrication plants can be grouped into three types on the basis of the quantity and character of the radioactive material that would be released to the environs, as described in the accident scenarios in the following subsections.

- The accident, which breaches the primary barrier but not the building confinement barrier, and results from a fire, an explosion, or mechanical damage to a glove box. The consequence of this type of accident is the potential release of small quantities of particulate material via the building exhaust system.
- The criticality event, which releases small quantities of particulate material plus some noble gases and volatile halogens.
- The major event, either an in-plant accident or a natural disaster, which breaches the facility confinement and releases substantial quantities of unfiltered plutonium-bearing air to the environs. The major event is considered extremely improbable; however, it should be considered when estimating the range of release levels, which relates to the instrumentation capabilities required for emergency preparedness.

5.1.1 Breach of Primary Containment (Accident in Operational Area)

Substantial quantities of plutonium could be made airborne within the facility if the primary containment were breached by an explosion, fire, or criticality event, by the overpressurization of tanks or piping containing plutonium solutions, or by the overpressurization of glove boxes containing PuO₂ powder. The exhaust system is assumed to remain functional, although one bank of filtration may be lost.

Several types of explosions are conceivable in a mixed oxide fuel fabrication facility. The most commonly considered are explosions in sintering furnaces, autoclaves, and plutonium solution tanks. These

types of events are highly localized and would result in damage only to small areas. No mechanism has been postulated for an explosion that would affect the entire facility and result in major damage to the building.

Hydrogen explosions in pellet sintering furnaces would have a limited amount of energy. The damage which could result from this type of event would likewise be limited. The explosion would probably be directed out the end of the furnace; the hood could be breached and pellets could be spread around the room. The physical nature of the pellets would make it virtually impossible to produce significant quantities of plutonium particles in the respirable range. This type of accident would result only in an in-plant spread of contamination and a negligible source term to the environs since the plant ventilation system would remain intact.

A chemical explosion involving a plutonium solution could result in the production of a substantial quantity of airborne plutonium particles, but the number of particles that would persist in the air is limited; Swain and Haberman⁽⁴⁾ reviewed data from non-nuclear sources and calculated that 33 mg of plutonium per m^3 was the maximum airborne concentration a few minutes after an accident. Castleman, Horn and Lindauer,⁽⁵⁾ using an exploding wire technique to generate very fine particles, found concentrations as high as 71 mg of plutonium per m^3 a few minutes after generation. By calculation, 10^6 plutonium particles per cm^3 , with an aerodynamic equivalent diameter (AED) of 10 μm , would result in an airborne concentration of 2 g of plutonium per m^3 . Particles generated by a single mechanism are generally log-normally distributed. Thus, the anticipated weight concentration of 10- μm AED particles would be much lower than that calculated for monodispersed particles. An upper value of 100 mg of plutonium per m^3 for particles in the respirable range appears reasonable.

If the room in which the explosion is postulated to occur is assumed to have a volume of $10^4 m^3$, a maximum airborne concentration of 100 mg of plutonium per m^3 of air would limit to about 1 kg the amount of airborne plutonium that could be sustained in the room following the explosion.

The occurrence of a fire near the nitrate blending tanks represents a low probability potential for release of plutonium, but should be considered because its consequences could be serious. If the resulting pressure were high enough, the vessel could rupture. Assuming the rupture occurred above the liquid level, some plutonium would become airborne but the quantity would be insignificant. If a large rupture occurred near the bottom of the tank, a stream of solution would pour onto the floor and extinguish part or all of the fire. The fractional airborne release would be low -- less than 0.003%, if the fire were extinguished.⁽⁶⁾ However, if the fire continued and dried the nitrate solution, the airborne release could be as high as 0.5%. The total amount of plutonium made airborne would be 200 kg x 0.5%, or 1 kg.

An accident involving the breaching of the primary barrier was examined for powder treatment areas. Sized PuO_2 powder, prior to the UO_2 blending, represents one of the greatest potential sources of respirable plutonium in a fabrication facility. To estimate an upper limit, mechanical damage in the area of the PuO_2 powder line was assumed. The damage would rupture the glove box and break an adjacent compressed air line. The line was assumed to sag, directing a jet of air into an open container holding an entire batch of 11.3 kg of PuO_2 powder. The glovebox filter was assumed to load quickly, causing all material to flow out of the glove box thereby bypassing the glovebox filter system.

Under such conditions, more than 50% of the powder could be made airborne. Mishima⁽⁷⁾ found that 60 to 70% of a relatively coarse oxalate powder (median mass diameter of 50 μm) could be made airborne by air drawn tangentially across the powder. Royster and Fish⁽⁸⁾ found that approximately 80% of 5- μm particles would be made airborne from a stainless steel surface by air moving at a velocity greater than 60 m/sec and impinging on the surface at various angles. Most of the plutonium for the powder operation was assumed to be in the respirable range (a PuO_2 sphere less than 3.3 μm). Although deagglomerating a powder

composed of such fine particles is not a trivial task, a "conservative" assumption was made that the airborne material would be deagglomerated and instantaneously and uniformly distributed; under these conditions, a facility with a volume greater than 10^5 m^3 (a building 800 ft long x 100 ft wide x 50 ft high) would be needed to get 11.3 kg of plutonium into the air, because of the "maximum achievable" airborne concentration of 100 mg of plutonium per m^3 .

Assuming a more realistic room size of 10^4 m^3 , the maximum amount of plutonium that could be made instantaneously airborne in the room would be 1 kg. Continuously applying the jet of air on the powder over an extended period would, of course, put more plutonium in the air, but the situation would probably be corrected immediately.

On the basis of these calculations, it appears reasonable to assume that about 1 kg is the maximum amount of plutonium that could be made airborne in an inplant accident. With a single HEPA filter in the building exhaust system, about 1 g of plutonium would be released to the environs. With two stages of HEPA filtration, the release would be about 10 mg of plutonium. The probable release would be via the exhaust "stack" (essentially at ground level), with a probable release duration of less than 2 hours.

5.1.2 Criticality Accident

A criticality of 10^{18} total fissions was assumed to occur in a solution of plutonium nitrate. A criticality event in a solution is self-limiting because microbubbles form from dissociation of the liquid by fission fragments. The formation and decay of the microbubbles result in the system pulsing between prompt critical and subcritical conditions until gradual evaporation of the solution terminates the event.

Vaporization of some of the solution during the initial power pulse was assumed to cause rapid volume expansion of sufficient magnitude to rupture the vessel. The force would be insufficient to damage the building or final filter system. For the 1000- ℓ solution containing

200 g/l, it was assumed that a 10% boiloff would terminate the event. A maximum of 15 g of plutonium would be airborne in the room. With a single HEPA filter in the building exhaust system, about 1.5×10^{-2} g of plutonium would be released; for two HEPA filters in series, the release would be about 1.5×10^{-4} g.

The fission products generated as a result of this criticality are shown in Table 5.1. All of the noble gases and one-quarter of the iodine were assumed to be released from the building. Zero release was assumed for the volatile solids and remaining fission products.

TABLE 5.1. ⁽¹⁾ Fission Product Inventory, in Curies, from 10^{18} Fissions (32 MW-sec)

Fission Product	Duration of Event	Initial ^(a)	10 min	60 min	2 hr	5 hr	10 hr	1 day
Noble Gases	0.5 sec	2.5E6	2.4E3	2.1E2	1.1E2	5.9E1	3.2E1	1.4E1
	5 min	3.8E4	1.8E3	2.0E2	1.1E2	5.9E1	3.2E1	1.4E1
	1 hour	4.0E3	5.3E2	1.5E2	9.9E1	5.4E1	3.1E1	1.4E1
Halogens	0.5 sec	6.0E5	4.7E2	2.7E2	2.1E2	7.4E1	3.4E1	1.5E1
	5 min	2.1E4	3.7E2	2.7E2	2.1E2	7.4E1	3.4E1	1.5E1
	1 hour	2.1E3	2.8E2	2.5E2	1.8E2	6.5E1	3.2E1	1.4E1
Volatile Solids	0.5 sec	9.0E5	2.3E3	6.7E2	2.4E2	1.8E1	5.8E0	3.3E0
	5 min	3.4E4	2.0E3	6.4E2	2.3E2	1.8E1	5.8E0	3.3E0
	1 hour	4.1E3	1.1E3	4.2E2	1.5E2	1.4E1	5.6E0	3.3E0
ARFP ^(b)	0.5 sec	1.5E6	1.5E4	2.0E3	7.5E2	3.3E2	1.8E2	6.4E1
	5 min	1.1E5	1.2E4	1.9E3	7.3E2	3.3E2	1.8E2	6.4E1
	1 hour	1.6E4	4.7E3	1.2E3	6.0E2	3.1E2	1.7E2	6.2E1

(a) $2.5E6 = 2.5 \times 10^6$

(b) All remaining fission products

5.1.3 Confinement Failure (Failure of Exterior Structure of the Facility)

It would require a major disaster to breach facility confinement and release unfiltered, plutonium-bearing air to the environs. Only a few accidents could theoretically produce damage of sufficient magnitude to negate the final confinement barriers. A specific plant will, as a

minimum, be structurally designed and built to exceed normal regional building codes relative to earthquakes and tornadoes. However, finite possibilities exist that the facility could be stressed by forces beyond those used for design or that the structure would yield to less than design-basis forces. While the events postulated below are highly improbable, they were examined nevertheless to define the upper range for instrumentation.

The approach used in examining the consequences of this type of accident was to first estimate the amount of plutonium in the reference processes that would be available for dispersion and could result in respirable plutonium aerosols in the environs following the accident. In making these estimates, it was assumed that mechanisms such as fire and winds occurred during the primary accident to generate plutonium aerosols. The bulk of the plutonium in the facility was assumed to be in storage in a hardened area such as a vault. Only significant quantities of materials in four operational areas -- conversion, powder treatment, scrap recovery and laboratory -- could generate radiologically significant aerosols. The total "at risk" inventory of material in the four areas was assumed to be approximately 100 kg (102.6 kg of plutonium). Thus the inventory "at risk" for the entire facility was considered to be 100 kg of plutonium.

If an earthquake is assumed to be the causative agent for the failure of confinement, severe damage to the glove boxes and enclosures within the building and loss of power to the building exhaust system might be ancillary effects. Under such conditions, as much as 1% of the inventory "at risk" ($0.01 \times 100 \text{ kg of plutonium} = 1 \text{ kg of plutonium}$) might be released, unfiltered, to the environs.

Tornadoes could be postulated as another causative agent. Since a characteristic of tornadoes is very high wind speeds, the fractional release was increased to 10% ($0.1 \times 100 \text{ kg} = 10 \text{ kg of plutonium}$).

A plant fire that could cause confinement failure is not considered possible. With the expected concrete construction of a facility of this

type, the low fire loading, the expected air-tight nature and the presence of fire protection sprinklers, alarms, etc., total burning is incredible. For analysis purposes, a break in the final barrier was assumed to have occurred, resulting from a facility fire involving the glove boxes and other combustibles in process. It was estimated that less than 1% of the in-process dispersible inventory of plutonium would become airborne within the facility. The consequence of a fire would be to load filters with soot particles and water, increasing the pressure drop and reducing the airflow. The multiple filter barrier as a whole would probably retain its integrity. Some holes might occur in the first-stage filter media, but the other back-up filters would be expected to remain highly efficient.

The maximum quantity of plutonium that could be released to the environs as a consequence of confinement failure is on the order of 1 kg for an earthquake or major fire, and on the order of 10 kg as a result of building destruction by a tornado. The release would be at ground level, with a duration of 2 to 8 hr.

5.1.4 Source Terms

Table 5.2 summarizes the maximum potential releases and possible release rates for the sequences described above.

TABLE 5.2.⁽¹⁾ Maximum Potential Release and Release Rates

Event	Material		Release, Ci		Release Rate, Ci/sec
			In-Plant ^(a)	Environs ^(b)	Environs
Local Fire; Explosion; Mechanical Damage	Pu	α	4.3E2 ^(c)	4.3E-1	1.2E-4
		β	1.4E4	1.4E1	3.9E-3
Plant Fire; Earthquake	Pu	α	4.3E4	4.3E2	6.0E-2
		β	1.4E6	1.4E4	1.9E0
Tornado	Pu	α	4.3E4	4.3E3	6.0E-1
		β	1.4E6	1.4E5	1.9E1
Criticality	Pu	α	6.5E0	6.5E-3	1.1E-5
		β	2.1E2	2.1E-1	3.5E-4
	Noble Gases	β	2.4E3	2.4E3	4.0E0
	Halogens	$\beta\gamma$	4.7E2	1.2E2	2.0E-1
	Fission Products	$\beta\gamma$	1.7E4	1.7E1	2.8E-2

(a) Assuming 0.43 α Ci/g and 14 β Ci/g.

(b) Assuming one intact stage of HEPA filters.

(c) 4.3E2 = 4.3×10^2

5.2 DETECTING THE EVENT

The sensitivity of detection systems must be low enough to permit detection of radioactive releases before hazardous levels are reached. The systems should not be disabled by high levels of radioactivity or by the incident itself.

Generally, three detection systems should be available in the facility:

- a routine radiation monitoring system
- a fire detecting system
- an emergency instrumentation system.

Where the systems are separate, at least one decade overlap should exist between the ranges of routine monitoring and emergency instrumentation. The range requirements for various emergency instruments are summarized in Table 5.3.

TABLE 5.3.⁽¹⁾ Detection Range Requirements for Fuel Fabrication Facilities

	Detection Levels	
	Minimum	Maximum
Plutonium (α) Stack Monitor	10^{-10} Ci/cm ³ 10^{-11} Ci/cm ³	10^{-4} Ci/cm ³ 10^{-5} Ci/cm ³
Neutrons	10^{14} fissions	10^{19} fissions
Portable Instruments		
γ	0.1 mR/hr	10^4 R/hr
β	0.1 mR/hr	10^4 R/hr
α	500 dis/min/100 cm ²	10^7 dis/min/100 cm ²

5.2.1 Communicating the Occurrence of an Event

The detection systems should be connected to alarming devices to warn and elicit appropriate action from those involved in or near the facility as well as those charged with emergency responsibilities. The alarm signal must be different for each different response expected from personnel. For example, if only the process area must be evacuated, the alarm signal

given should be different from that requiring evacuation of the entire building.

Alarms should be located in the immediate vicinity of potential accidents in the building, in a routinely occupied location (e.g., at the process-operator control station), and at an emergency control center.

5.2.2 Evacuating the Area of Risk

The area of risk can range from a relatively small part of the process area to the entire facility and its immediate environs. Evacuation routes should lead as directly as possible to a specified location (the Emergency Control Center). For efficiency, emergency response personnel should normally report to the Emergency Control Center. Actions to evacuate personnel (to reduce contamination spread or to secure processes) must be kept to a minimum and must provide accurate personnel accountability.

5.3 EMERGENCY ACTIONS

Those involved in the event are usually the best source of information on what rescue efforts are needed. This information, along with data on the facility's structure and the level of contamination, must be provided to those who undertake rescue efforts. It must be borne in mind that, while additional risk to personnel is justified for rescue, it is not justified for the recovery of a body.

5.3.1 Undertaking Rescue

Any Emergency Control Center should have supplies for radiological protection and rescue equipment. As a minimum, the following are needed:

- Portable radiological survey instruments
- Respiratory protective equipment
- Protective clothing
- Decontamination supplies
- First aid supplies
- Litters.

5.3.2 Isolating the Area of Risk

Simultaneously with rescue efforts, the area of risk should be isolated from any unnecessary entrance. Physical barriers may have to be constructed if none are readily available. The area of risk includes down-wind regions subject to airborne and fallout contamination.

All personnel who are leaving or have left the area of risk must be surveyed. The amount of plutonium or other radionuclide intake, if any, must be estimated as early as possible so that chelation therapy can be started if indicated. Only life-saving medical attention takes precedence over plutonium decontamination. In the event of a possible criticality, quick-sort methods for the detection of induced activity should be employed.

5.3.3 Establishing a Controlled Entry Point and Routes of Travel in the Isolated Area

The controlled entry point may or may not be the Emergency Control Center, but must be established as early as possible.

All entry into and exit from the isolated area must be rigidly controlled. All personnel entering the area must have their routes of travel rigidly defined and must be provided the appropriate protective clothing and radiation protection coverage.

5.3.4 Analyzing the Event

Analysis of the event should begin when the alarm is sounded and will probably continue well into the recovery phase.

In some instances, as thorough an analysis as possible should be performed before attempts are made to stabilize the condition that led to the emergency. All process and building systems and their interactions should be considered. Check-off lists prepared as part of written emergency preparedness would be helpful.

5.3.5 Stabilizing the Emergency Condition

This effort should be directed first to determining actions necessary to assure the safety and well-being of personnel. Next, the correct action should be taken to prevent any additional spread of plutonium or other hazardous material: for example, extinguishing a fire in an office area is secondary to closing off a building contaminated with plutonium or other hazardous materials; however, if the office fire is threatening a contaminated structure, extinguishing the fire is paramount.

5.4 PUBLIC RELATIONS

The increased public concern about the nuclear industry, particularly plutonium, makes prompt, factual news releases extremely important. Consideration should be given to the development of generic news releases that can be used effectively after the appropriate blanks have been filled out.

5.5 EMERGENCY PREPAREDNESS PLAN

An effective emergency preparedness plan for a plutonium facility should encompass the following:

- Lines of authority
- Areas of individual responsibility
- Specific guidelines
- The training and retraining of emergency response personnel
- The testing and retesting of systems, hardware, communications and procedures.

Positions for which personnel must be identified and trained include:

- Emergency director (the controlling authority during the emergency phase)
- Rescue team members
- Personnel decontaminators
- Medical staff
- Exposure evaluator (personnel and environmental)
- Communications specialist
- Personnel accountability aides
- Firemen

Activity centers that may be necessary during an emergency include:

- Evacuee assembly area
- Medical treatment center
- Rescue control point
- Emergency control center
- Communications center
- Transportation center
- Information release center
- Recovery control center
- Personnel dosimetry control center
- Records center.

Plans should be established for the provision of the following services:

- Security control
- Medical (onsite and offsite)
- Transportation
- Traffic control
- Meteorology
- Auxiliary electrical power.
- Communications

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SECTION 6

DECONTAMINATION, DECOMMISSIONING AND DISMANTLING

SECTION 6
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SECTION 6

DECONTAMINATION, DECOMMISSIONING AND DISMANTLING

Recent developments in the continuing debate over nuclear fuel cycle alternatives clearly indicate that licenses for future fuel cycle facilities will deal with the technical and economic aspects of decommissioning as well as traditional considerations. Decommissioning escrow accounts or other similar arrangements will presumably have to be drawn up in the near future; at present, however, the cost of decommissioning a mixed oxide fuel fabrication plant cannot be accurately estimated because this cost depends closely on the cost of retrievable storage for transuranics. Estimates of retrievable storage costs at a suitable repository (when such a facility becomes available for use) vary by a factor of 15 and therefore have a profound effect on decommissioning costs.

The intent of this section is therefore limited to outlining a plan for the decommissioning of a generic plutonium facility and identifying considerations for the safe and economically efficient execution of such a task.

6.1 APPLICABLE STANDARDS AND GUIDES

Guidelines to decontamination and decommissioning (D&D) practices for ERDA facilities are contained in ERDAM-0524 and -6301.^(1,2) ERDAM-0524, "Standards for Radiation Protection," includes on Appendix page 9:

- 5) Decommissioning requirements should be considered in the design of a facility. The avoidance of rough surfaces, cracks, and crevices in potential contamination areas should be considered in this context.
- 6) The use of protective coating in radiation areas should comply with the specifications contained in ANSI N5.9-1967 (Protective Coating for the Nuclear Industry).
- 7) Interior surfaces, as well as layout of ducts and pipes, should be designed to minimize buildup of contamination and exposure to personnel, and to facilitate cleanup.

The appendix of ERDAM-6301, "General Design Criteria, Plutonium Facilities," also contains information on decontamination and decommissioning (Appendix page 81):

- j. Decontamination. Design of the critical areas shall incorporate measures to simplify decontamination. Such items as service piping, conduits, and ductwork should be kept at a minimum in the operating areas and be arranged to facilitate decontamination. Walls, ceilings, and floors should be finished with washable or strippable coverings or covered with metal liners if required.
- k. Decommissioning. Decommissioning of the plutonium facility is of utmost importance and design shall be such that at some future date the facility can be decontaminated for future decommissioning. The design principles which shall be considered are:
 - 1) Use of modular, separable enclosures for use of radioactive materials to preclude contamination of fixed portions of the structure.

- 2) Liquid transfer systems to avoid long runs of buried contaminated piping with emphasis on localized batch solidification of liquid waste.
- 3) The proximity of exhaust ventilation filtration to enclosures to minimize long runs of internally contaminated ductwork.
- 4) Effluent decontamination equipment to preclude, to the extent practicable, the accumulation of radioactivity in relatively inaccessible natural soil columns, such as beneath "seepage basins."

Important sources of information for decontamination and decommissioning are: 1) BNWL-1974, Decontamination and Decommissioning of a Mixed Oxide Facility (this document is currently in draft form and is expected to be released soon); and 2) BNWL-1917, Decontamination and Decommissioning of Nuclear Facilities -- A Literature Search.⁽³⁾ BNWL-1917 contains an extensive literature search. Most of the published information is on the decontamination and dismantling of nuclear reactors. However, the general methods used are applicable to the dismantling of plutonium facilities.

6.2 DEFINITIONS AND TERMINOLOGY

The authors of BNWL-1974, Decontamination and Decommissioning of a Mixed Oxide Facility, have attempted to establish a standardized terminology for D&D activities. Their definitions are presented here.

6.2.1 Use Categories

When decommissioning is being planned, a variety of options for the future use of the site or facility can be considered. Three broad use categories are outlined here, together with the general types of restrictions for each that will limit public exposure to acceptable levels.

No attempt has been made to identify all of the specific uses that might fall into each broad category. The possibilities seem almost unlimited due to the many variables and combinations of variables related to the site (e.g., the amount of radiation present before decommissioning; the population of the surrounding area; the topology, weather, and economic potential). Each case should therefore be decided on an individual basis, taking these variables into account and using as a guideline the categories given below.

Use categories for sites and facilities can be broadly classified as follows:

- Restricted Use
- Conditional Use
- Unrestricted Use

Restricted Use

The "Restricted Use" category limits the decommissioned site and/or facility to "nuclear use" only. Voluntary (i.e., occupational) onsite doses are restricted by the 10 CFR occupational exposure limits.⁽⁴⁾ Under these conditions, the maximum involuntary dose to an offsite individual is limited to levels as low as practicable as specified for reactors by the NRC.⁽⁵⁾ Such limitations are achieved by the application of appropriate security and surveillance measures.

Conditional Use

The "Conditional Use" category utilizes various types and levels of restriction to keep the Maximum Exposed Individual's dose potential at or below the Unrestricted category limits.^(a) These restrictions limit the availability of radioactive or toxic materials and may take the form of environmental barriers and/or use restrictions. The nature of the exposure barriers may dictate the need for surveillance and/or maintenance programs to assure compliance with restrictions and observance of the barriers; or the use of statutory law may be sufficient to limit land use and restrict dose pathways. Whatever the combination of restrictions and barriers, they must limit the site's exposure potential to that of the Unrestricted Use category.

Unrestricted Use

In the "Unrestricted Use" category, the site and facility have been returned to a "pre-facility" state and are unconditionally released for public access. No physical limitations or laws restrict use of the land or its resources.

6.2.2 Disposition Modes

The seemingly infinite variety of methods for facility disposition under the various use categories can in reality be reduced to combinations of the following four distinctive disposition modes:

- Layaway
- Protective Storage
- Entombment
- Dismantling

Layaway

For many nuclear facilities, layaway can be considered an intermediate step prior to further work. A minimum of activity is carried out in deactivating the facility. The activities associated with this mode may include: 1) offsite disposal of all solid and liquid radioactive and non-radioactive process materials in approved repositories; 2) controlled

(a) The Maximum Exposed Individual is a hypothetical member of the population whose location and dietary and living habits tend to maximize the radiation dose he receives.

access to the site and facility (authorized personnel only--this may include the erection and maintenance of fences or other physical barriers); 3) periodic security, physical, radiological and environmental surveys or inspections of the facility and site; 4) operation and maintenance of protective equipment such as ventilation, radiation monitoring, fire alarms and electrical lighting and power systems; and 5) decontamination or shielding of easily accessible contaminated structures and equipment to minimize occupational exposure.

The major benefits of the layaway mode are: 1) reduced radiation levels at a future date if further deposition activities are undertaken, 2) the postponement of irrevocable decommissioning activities, and 3) a low initial outlay of funds.

Protective Storage

Protective storage means the facility is controlled by maintenance and monitoring, as required, to assure that no release of contamination occurs and that no unregulated entry is made. This mode may be adopted as a permanent measure, with the intent to continue surveillance and maintenance activities until the facility can be released for other use; or it may be adopted as a temporary measure, to delay decontamination and/or dismantling operations until radiation levels have dropped off significantly through decay.

Protective storage includes the following minimal activities: 1) offsite disposal of all solid and liquid radioactive and nonradioactive process materials in approved repositories; 2) controlled access to the site and facility as in the layaway mode; 3) periodic security, physical, radiological and environmental surveys and inspections of the facility and site; 4) operation and maintenance of protective equipment such as radiation monitoring, fire alarms and electrical and power distribution systems (ventilation shut down but in operable condition); 5) erection of physical barriers to prevent entry into contaminated zones; and 6) removal and/or fixing of accessible contamination in the facility.

The major benefits of the protective storage mode of disposition are similar to those for the layaway mode with these additions: 1) a low initial outlay of funds (though higher than that for layaway), 2) reduced surveillance, maintenance and operating costs for the facility, 3) a reduced hazard to the general public and workers on the site, and 4) the reuse of the accessible uncontaminated portions of the facility.

Entombment

Generally, entombment involves removing the facility's uncontaminated structures and equipment and surrounding (or perhaps filling) the grossly contaminated areas with concrete. To reduce the volume to be entombed, outlying contaminated components may be dismantled and placed inside the structure to be enclosed.

Entombment of nuclear facilities includes the following activities as a minimum: 1) offsite disposal of all solid and liquid radioactive and nonradioactive process materials; 2) controlled public access to the site and facility; 3) periodic physical, radiation and environmental surveys and inspections; 4) maintenance and limited operation of lighting systems and fire protection alarms; 5) removal of salvageable uncontaminated equipment; 6) erection of "permanent" (>100 years) barriers to protect the environment from radioelements (e.g., filling radioactive vessels and cells with concrete); and 7) decontamination of the site.

Compared with layaway and protective storage, the advantages of decommissioning by entombment are 1) the reduced costs of surveillance and maintenance of the facility, 2) reduced short-term hazards to the general public and workers in the vicinity, and 3) the release of the site for minimal controlled use. Disadvantages include the increased costs of disposition, and the requirement that the wastes resulting from decommissioning be changed to a form from which retrieval is difficult and costly (should policy revisions require such a retrieval).

Dismantling

For dismantling, all radioactive materials, equipment and structures are decontaminated (if only surface contamination is present) or dismantled and removed for disposal elsewhere. The dismantling mode requires that all wastes be packaged and transported in accordance with national and state regulations, and necessitates the segmentation of large radioactive components and structures into pieces small enough to fit into approved containers.

Dismantling nuclear facilities includes these activities: 1) offsite disposal of all solid and liquid radioactive and nonradioactive process materials; 2) removal and offsite disposal of all processing equipment and structural materials that contain levels of contamination above unrestricted release limits; 3) salvage of nonradioactive equipment and structures; 4) removal of all retired nuclear structures; 5) removal of contaminated site materials; and 6) reestablishment of the site environs.

The elimination of radiation hazards and security, maintenance, and surveillance requirements at the site are advantages of the dismantling mode of decommissioning. Disadvantages include the generally higher cost of the disposition activities and the possibility of high occupational exposure to radiation during the dismantling process.

Factors in the Choice of Mode

Determining which decommissioning mode should be chosen for a given nuclear facility involves the careful weighing of technological, economical, societal, and political factors. The final choice of a decommissioning mode depends upon the complex interplay of these factors, which deeply involve the governing regulatory agencies of Federal and state governments and the operator of the nuclear facility. In light of past experience, decommissioning decisions are being expanded to consider long-term factors such as the worth of the site and its resources, population growth, genetic effects, cumulative health effects, and a host of sociopolitical factors.

6.3 FACILITY PLANNING

Planning for the decommissioning of a plutonium facility should start when the facility is first designed. Some of the more obvious good practices to follow in designing the facility are listed in the following subsections.

6.3.1 Glove Boxes

- Absolute filters should be placed on or near the glove boxes to avoid long runs of contaminated ductwork.
- The interior surface of stainless steel glove boxes should not be sandblasted. A sandblasted surface looks nice, but plutonium oxide powders and others compounds are almost impossible to remove from the surface. In addition, the sandblasting process results in high radiation doses to workers.
- Modular construction should be used, with components that can be easily lifted and removed.

6.3.2 Pipes and Cables

- Pipe and cable runs in areas that may be contaminated should be avoided. Put long pipe and cable runs in a "clean" gallery.
- Long, buried pipe runs for radioactive liquids should be avoided if possible, as they are bound to leak some day. The conversion of liquid wastes to solid wastes should be considered. Improved treatment methods should be used to remove plutonium from liquid waste streams; seepage basins for plutonium-contaminated wastes should be avoided.
- Horizontal runs for pipes carrying radioactive liquids or slurries should be avoided, especially in occupied areas.

6.3.3 Rooms Containing Glove Boxes or Process Equipment

- Contamination of structural members of the building should be avoided, as decontamination is difficult and expensive.
- All concrete surfaces should be painted; avoid rough surfaces with cracks and crevices.
- All surfaces that may become contaminated, even by accident, should be covered with a washable or strippable coating that meets the provisions of ANSI Standard N5.9-1967, "Protective Coatings for the Nuclear Industry."⁽⁶⁾
- Metal walls or partitions should be used to minimize the spread of contamination in case of an accident. These can be easily removed and prevent the contamination of structural members of the building.

6.3.4 Heavy Equipment

- If the facility contains contaminated, heavy equipment, an overhead crane may aid in removing the equipment. Since the building must be earthquake and tornado resistant, it should be structurally sound enough to support a crane.
- Tall, double doors should be installed so that glove boxes and equipment can be moved without tearing down walls.

6.4 LEVELS OF DECONTAMINATION

No definite guidelines or regulations currently exist on acceptable levels of radiological contamination for various classifications of nuclear facilities or for potential uses of land after radiological decontamination. The disposition of potentially contaminated land and/or facilities for other uses demands decontamination to residual radiation levels that are acceptable as well as verifiable. Public health and safety must be the primary basis of the criteria for such disposition of facilities.

In the following subsections, some potential uses of decontaminated land and facilities are discussed and categorized, and criteria are developed for the disposition of property in each use category.

6.4.1 Existing Guidance

Criteria for decommissioning nuclear facilities must provide adequate protection to the public and must also be useful to the nuclear industry. These criteria must include the following elements:

- a consistent rationale
- specific numbers defining "acceptable" contamination levels for various site and/or facility uses
- analytical verifiability
- easy application.

Very little guidance exists for the decommissioning of nuclear facilities. That which does exist does not provide the basic information the nuclear industry needs to plan decommissioning. A new approach that uses the methodology of conventional dose calculations is therefore described in this section.

A search of the literature for previously defined criteria reveals that contamination limits for mixed fission product and transuranic radionuclides have been derived piecemeal and independently of one another. Documented incidents, such as the cleanup of a plutonium-contaminated truck terminal⁽⁷⁾ or the cleanup of plutonium contamination spread from damage to several nuclear explosive devices involved in an airplane crash

in Greenland,⁽⁸⁾ required immediate action and decisions on acceptable criteria for the release of related sites and facilities. In addition, past decontamination efforts have focused primarily on surfaces involving only simple radionuclide mixtures. The following are a few examples of existing decontamination guidelines for plutonium-contaminated materials:

- The draft of ANSI Standard N13-3A contains criteria for smearable and total plutonium contamination levels. These are:

	<u>(α) dis/min/100 cm²</u>
Smearable	20
Total	100

- A review^(a) of a draft ANSI standard suggested alternatives for total contamination:

	<u>(α) dis/min/100 cm²</u>
Smearable	50
Total	500

- In a study for the establishment of a regulatory standard,⁽⁹⁾ cutoff between definitions of contaminated and noncontaminated was established and related to criteria for surface and volumetric contamination. The definitions of plutonium contamination were:

Surface	>10 nCi/m ²
Volumetric	>10 pCi/g

The few criteria that exist for the decontamination of bulk materials (such as water and soil) are inadequate for general situations.

Other criteria that have been defined for specific situations are represented by the following:

- The U.S. Department of Transportation defines insignificant removable contamination in 10CFR 49.⁽¹⁰⁾

(a) Letter from J. F. Sommers (Idaho Falls Aerojet Nuclear) to E. J. Vallario (Chairman of Health Physics Society Standards Committee), "Comments on Proposed Standard N-328," SOM-93-74, dated December 13, 1974.

- NRC defines "Acceptable Surface Contamination Levels" in Regulatory Guide 1.86.⁽¹¹⁾
- ERDA defines contaminated waste in ERDAM-0511.⁽¹²⁾
- Healy recommends a total maximum allowable concentration of plutonium in A Proposed Interim Standard for Plutonium in Soils.⁽¹³⁾
- Guthrie and Nichols define acceptable contamination levels for a large number of radionuclides.⁽¹⁴⁾
- The State of Colorado evaluates acceptable residual levels of plutonium contamination.⁽¹⁵⁾

These criteria are summarized in Figure 6.1.

The absence of uniformity in the preceding reference documents makes any intercomparison of these criteria extremely difficult. Their application to situations other than those for which the criteria were developed is practically impossible.

6.4.2 Philosophy and Objectives

Existing guidance for the disposition of radiologically contaminated property does not take account of changes in the composition of contamination mixtures or changes in environmental conditions in the area under question. This inflexibility leads to the conclusion that a philosophy for setting disposition criteria should be based on a broadly accepted criterion such as dose commitment. The objective is to provide a rationale and methodology that are consistent and adaptable, yet provide definitive, technically-based numerical guidance for acceptable criteria for facility release.

The basic philosophy guiding this development of criteria for the decommissioning of potentially decontaminated facilities includes the following considerations.

- Decontaminated facilities and sites may be opened to unrestricted use, or may be put under one of several use restrictions.

- The potential annual and 50-year dose commitments received by the "Maximum Exposed Individual" from the use of any site or facility must under no circumstances exceed the estimated acceptable dose commitment, regardless of the potential use of the site.

The intent behind this methodology is that once parameters for radiological protection have been quantified, the resulting protection level must be met in every use category. That is, under no circumstances may the calculated dose to the maximum exposed individual exceed the dose specified for the Unrestricted Use level. Where higher residual contamination levels are allowable, increased exposure potentials must be prevented through use restrictions specified for each use category.

This philosophy for the development of disposition criteria provides a rationale and methodology that are consistent, adaptable, and based on technical information. The resulting numerical guidelines can be used to determine acceptable combinations of decontamination levels and subsequent uses for any facility or site.

6.4.3 "Acceptable Dose" Considerations

Specific quantitative guidance on what an acceptable dose commitment to the maximum individual might be is not currently available; therefore, a search was begun for qualitative guidance that could be interpreted quantitatively. The State of Colorado⁽¹⁵⁾ has evaluated on the basis of health effects a residual plutonium contamination level of $0.01 \mu\text{Ci}/\text{m}^2$ and found it acceptable. ERDA^(16,10) has promulgated radiological standards couched in terms of removing all detectable transuranics; as interpreted by ANSI, this leads to a level of $0.01 \mu\text{Ci}/\text{m}^2$, indicating agreement between these two groups on the acceptability question.

A dose calculation for a 1-year atmospheric release using a high fuel exposure⁽⁹⁾ plutonium mixture at a concentration of $0.01 \mu\text{Ci}/\text{m}^2$

(over an area of 0.5 km²) yields the necessary quantitative guidance on the maximum "acceptable" 50-year dose commitment to a critical organ. The calculated value of 270 mrem, rounded to 300 mrem, is shown in Table 6.1. The 1-year dose to a critical organ from this calculation (4.5 mrem, rounded to 5) is included in Table 6.1 to limit the annual dose from radionuclides that have a shorter effective half life in man than plutonium has.

TABLE 6.1. Proposed Acceptable Dose Criteria for the Use of Retired Nuclear Facilities

	<u>All Organs</u>
50-Year Dose Commitment, mrem	300
Annual Dose, mrem	5

If the doses listed in Table 6.1 are acceptable for plutonium, they should be acceptable no matter what the composition of the radionuclide mixture. Thus, the calculated doses are proposed as the estimated maximum acceptable doses for all mixtures. These dose limits are tentative, and opinions differing from those of the State of Colorado (on which these calculations were based) may be shown to be equally applicable. Note, however, that the methodology for deriving contamination levels, as described in Section 6.4.5, is independent of which doses are proposed as the basis of the criteria.

6.4.4 Assessment Parameters

Four criteria are used to judge the acceptability of the numerical contamination levels derived here. The criteria are:

1. potential dose commitment to the maximum exposed individual
2. potential annual dose to the maximum exposed individual
3. comparison with previously existing guidance
4. detectability in the field by today's portable instrumentation.

These criteria determine the acceptability of the derived contamination levels for Unrestricted Use. (The relationship between use categories is that the upper limit of a less restrictive category equals the lower limit of the next most restrictive category.)

Potential annual dose or dose commitment is the controlling factor in determining the acceptability of a contamination level. The upper bound for residual contamination in the Unrestricted Use category leads directly to the estimated acceptable or most conservative dose. The proposed acceptable dose is then applied, in combination with dose pathway restrictions, to derive upper bounds for residual contamination in the Conditional Use category. The Restricted Use category has its upper limit set by NRC's maximum voluntary occupational dose and offsite population dose limits.

The numbers derived for the disposition criteria can also be compared with previously suggested guidance; this serves as a second check on acceptability.

Since potential dose is the controlling factor for acceptability, it outweighs the other factors for acceptability listed above. Thus, if the residual contamination level calculated for a particular mixture of radionuclides just meets the dose criterion but is so low as to be undetectable by current field measurement techniques, the detectability criterion is preempted by the dose considerations and a new generation of more sensitive field instruments must be developed.

6.4.5 Method of Derivation

The derivation of disposition criteria is shown schematically in Figure 6.2 and involves the following procedural steps, which are detailed in subsequent subsections:

1. Determine acceptable dose criteria.
2. Determine the composition of radionuclides in the residual inventory.
3. Calculate potential doses from the unit release of and direct exposure to the residual inventory.
4. Examine dose calculations to determine critical pathways involved in the annual dose and dose commitment.

FIGURE 6.2. D&D Logic for Determining Acceptable Contamination Levels

5. Calculate the allowable release rate or allowable surface inventory (as required by the critical pathway) that will result in an annual dose/dose commitment not exceeding the "acceptable" annual dose/dose commitment to a critical organ.
6. Calculate the residual contamination level consistent with the critical pathway that will result in doses not exceeding the maximum allowable.
7. Determine the acceptability of criteria numbers on the basis of the assessment parameters discussed in Section 6.4.4.
8. Calculate the activity per unit mass that is equivalent, on the basis of critical pathway, to the surface contamination limit calculated in Step #6.

9. Evaluate potential mechanisms for reducing the annual dose/dose commitment and incorporate the results into a determination of Conditional Use criteria.
10. Establish smearable contamination limits.
11. Establish a criteria equation relating alpha and beta-gamma compliance measurements.

Background data necessary for the derivation procedure include the specific mixture of radionuclides found at the facility and/or site following retirement, and a description of site and regional meteorology, ecology, demography, and land use and condition.

The objective of this criteria derivation process is to find for each use category the maximum residual contamination level that, in combination with imposed safety restrictions, results in a potential annual dose/dose commitment equal to or less than that for Unrestricted Use sites. It follows that the Conditional Use criteria are in reality floating limits; that is, the contamination levels may be allowed to increase as additional restrictions and/or barriers are applied. As a result, the Conditional Use category is a gray area of possible contamination/restriction combinations, all yielding the same Unrestricted category dose.

To aid in visualizing this derivation process, a matrix with use categories (rows) and contaminated media (columns) was constructed as shown in Table 6.2. Criteria for surface contamination are applied where field measurements of surface activity are feasible. Criteria for mass contamination are applied to bulk materials such as soil and water.

TABLE 6.2. Basic Decommissioning Criteria Matrix

<u>Row</u>	<u>Column 1 Maximum Allowable Surface Contamination</u>	<u>Column 2 Maximum Allowable Activity/Mass Contamination</u>
1	Restricted Use	Restricted Use
2	Conditional Use	Conditional Use
3	Unrestricted Use	Unrestricted Use

Dose Calculation

Based on the specified mixture of radionuclides found at the facility following retirement, the potential 50-year dose commitment and first-year dose to a maximum exposed individual are calculated from a 1-year release of the total inventory and from direct exposure to the total inventory uniformly distributed over the site, using generally accepted computerized dose models and exposure equations. Even though the actual releases of contamination from the decommissioned site or facility is expected to take place over many years, a 1-year release is assumed for conservatism and to simplify assumptions concerning the maximum exposed individual and time-varying release fractions. The total inventory is used as a unit release and will be used for scaling purposes in subsequent calculations. The computer programs listed in Table 6.3 address all radionuclides specified for any facility and all emissions from these nuclides.

According to the programs, the maximum exposed individual is situated on the contaminated site at a point that maximizes his annual dose and dose commitment from the inventory release. The direct exposure calculation assumes a location at the center of the site. Doses are calculated for each of several pathways and then summed for evaluation. The pathways include direct radiation, submersion in the contaminated cloud, inhalation, and ingestion of materials in terrestrial pathways.

Calculation of Allowable Release Rate

When the dose calculation indicates that internal doses from the 1-year releases of the total inventory are limiting (i.e., the internal pathways are the critical pathways), the allowable release rate must be calculated by:

1. forming ratios of the facility's dose commitments to the critical organ (both first-year and 50-year), to the proposed maximum allowable doses shown in Table 6-1; and
2. dividing the release source term used in the dose calculation by the largest of the ratios from Step 1.

TABLE 6.3. Programs Used in Dose Calculations

Program	Item Calculated	Organ Doses Calculated	Reference
FOOD	Concentrations in produce, milk, eggs, meat from aerial deposition or irrigation at specified location	---	BNWL-SA-5523 ⁽¹⁷⁾
	Individual doses from ingestion of farm products(a)	Annual dose to total body GI-LLI, (b) thyroid, bone, liver, lung, kidney	
VITTL ^(c)	Concentrations on soil from aerial deposition or irrigation at specified locations	---	---
	Individual doses from exposure to soil(a)	Annual dose and dose commitment to skin, total body(d)	
	Concentrations in produce, milk, eggs, meat from aerial deposition or irrigation at specified location	---	
	Individual doses from ingestion of farm products(a)	Annual dose to total body, GI-LLI, thyroid, bone, liver, lung, kidney	
GRONK	Annual average atmospheric dilution factors (Ci/m ³ per Ci/sec released) versus distance and direction from the source ($\bar{x}/Q'$)	---	BNWL-1754 ⁽¹⁸⁾
	Individual doses from exposure to half-infinite cloud of effluents, versus distance and direction from the source	Annual dose to skin, total body	
	Integrated man-rem doses to total population within 50 miles	Annual dose to total body	
	Doses to child and adult from inhalation, and consumption of milk and leafy vegetables	Annual dose to thyroid	
GAUCHE	Individual doses from air submersion at a specified location	Annual dose and dose commitment to skin and total body	---
	Individual doses from inhalation and tritium transpiration at a specified location(a)	Annual dose and dose commitment to total body, GI-LLI, thyroid, bone, liver, kidney	---
	Doses to child and adult from inhalation, and consumption of milk and leafy vegetables, at any specified $\bar{x}/Q'$	Annual dose to thyroid	

(a) Population doses can be calculated by using population weighted average input data.

(b) GI-LLI means gastrointestinal tract-lower large intestine.

(c) VITTL also calculates the same internal doses as does FOOD, plus dose commitments.

(d) For these pathways, the calculated total-body dose is also listed under organs other than skin on the printout and is added into the total dose to these organs.

If direct exposure yields the limiting dose (i.e., the critical pathway is external), an allowable surface inventory must be calculated rather than an allowable release rate. The allowable surface inventory is obtained by dividing the inventory used in the direct exposure calculation by the maximum ratio of the critical organ doses described in the preceding paragraph.

Calculation of Allowable Residual Contamination

Where internal dose yields the critical pathway, the calculation of allowable release rate is followed by calculation of the residual contamination level that will result in that potential release rate. Using the area of the contaminated site, the site meteorology, and release fraction data, the decommissioning criterion for surface contamination in the Unrestricted Use category is calculated. For conservatism, the post-decontamination site is assumed to be barren, thus maximizing the potential for release of contamination. (Any vegetation that might remain after decontamination reduces the release fraction which in turn reduces doses resulting from environmental pathways.) The site is also assumed to be uniformly contaminated over its entire area. (Sites with smaller areas of contamination will result in correspondingly smaller releases and doses. Releases from or within buildings are smaller yet, making the criteria usable in that case also.)

Where external pathways are critical, the residual contamination level is determined using the allowable surface inventory calculated earlier and the contaminated site area.

Parameters for Criteria Acceptability

The derived numbers for the Unrestricted Use category are then checked to see that they meet the requirements for 1) acceptable potential annual dose/dose commitment, 2) comparison with previously suggested guidance, and 3) detectability in the field with portable instrumentation. Once the Unrestricted Use values are acceptable on at least bases 1 and 2, criteria for bulk contamination and for the other categories may be generated.

Calculation of Allowable Residual Bulk Contamination

For internal critical pathways, the limit for residual bulk contamination must have a release equivalent to that of the surface limit. The bulk limit is calculated in terms of activity per unit mass, using release fraction data and a methodology similar to the surface contamination calculations.

For external critical pathways, the allowable activity/mass is derived so that the resulting Ci/g quantity yields the same external dose as that calculated from the allowable surface contamination. The total inventory allowed for bulk materials is larger than that for surface contamination because of the self-shielding effect of the bulk material.

Calculation of Criteria for Other Use Categories

Examining the details of the dose calculation reveals the critical pathways (airborne, aquatic, and direct radiation), and thus indicates which mechanisms should be employed for manipulating active exposure pathways, to reduce the dose potential for other use categories. Smearable versus fixed contamination and compliance measurement equations for each use category must be considered.

6.4.6 Disposition Criteria

The numerical values derived for contamination levels are based on 1) the potential dose commitment to individuals or populations, 2) comparison with previously suggested guidance as shown in Figure 6.1, and 3) detectability in the field by portable instrumentation. Limiting the potential dose commitment to individuals or populations establishes an upper bound to the Unrestricted Use category. This limitation of dose commitment potential is explicitly considered in deriving upper bounds for other use categories. The resulting disposition criteria, i.e., the use restrictions at various activity levels, are presented in Figure 6.3.

	SURFACE CONTAMINATION	ACTIVITY/MASS CONCENTRATION
UNRESTRICTED	220 (α) dis/min/100 cm ² fixed 220 (α) dis/min/m ² smearable	10 pCi/g
CONDITIONAL	22000 (α) dis/min/100 cm ² fixed 220 (α) dis/min/m ² smearable	1 nCi/g
RESTRICTED	>22000 (α) dis/min/100 cm ² fixed 220 (α) dis/min/m ² smearable	>1 nCi/g

FIGURE 6.3. Disposition Criteria Matrix

6.5 D&D OF A PLUTONIUM FACILITY

Decisions involved in selecting the D&D mode and the specific D&D methods that will be used for a particular facility are complex, and a detailed discussion of them is beyond the scope of this document. However, an outline of the topics that should be considered in organizing the data base for such a decision follows:

Disposition Alternatives

A. Decontamination

1. detailed decontamination plan
2. contamination level estimates
3. equipment and manpower estimates
4. cost estimate
5. radiation exposure assessment
6. waste management alternatives

B. Dismantling

1. detailed dismantling plan
2. contamination level estimates
3. equipment and manpower estimates
4. cost estimate
5. radiation exposure assessment
6. waste management alternatives

C. Combination disposition modes

A data base derived from this outline defines operational, equipment and manpower requirements and develops the methodology for decontaminating and dismantling the facility using the "best" current technology or techniques available, or developing new techniques or equipment that are technically feasible.

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APPENDIX

DIAGNOSTIC EVALUATION OF INTERNAL DEPOSITION

APPENDIX

DIAGNOSTIC EVALUATION OF INTERNAL DEPOSITION

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APPENDIX

DIAGNOSTIC EVALUATION OF INTERNAL DEPOSITION

The purpose of this appendix is to give an overview of the complexities involved in evaluating depositions of plutonium. The more commonly used models are presented along with methodology for utilizing each model. The information included is not intended to be all-inclusive, but rather to present general evaluation techniques.

A.1 MODES OF INTAKE

The intake of Pu to the body can occur by four common modes: inhalation, injection (through wounds), ingestion and absorption.

Inhalation is the most common mode of intake because respirable aerosols may be encountered whenever Pu escapes from confinement into the working environment. The deposition, retention and translocation of the inhaled material depend on the size, density and chemical state of the particles.

Injection via wounds incurred by Pu-contaminated objects is the second most common mode of intake. Most injection intakes are caused by puncture wounds during routine glovebox operations. Translocation of the contaminant from the wound site into the blood stream depends on the chemical form of the plutonium and can occur rapidly for soluble forms. Subsequent deposition from the blood into the bone and liver also occurs rapidly.

Ingestion is a less significant mode of intake but can result from severe personnel contamination or inhalation. In these cases, the mode of intake into the blood stream (by inhalation or by absorption from the gastrointestinal tract) cannot be distinguished. Animal studies indicate that the influence of chemical form and other factors on the absorption of Pu from the gut is uncertain but that overall absorption is very low. The following absorption factors are currently recommended:

Relatively insoluble plutonium (e.g., PuO_2) - $10^{-4}\%$

Relatively soluble plutonium (e.g., nitrates) - $3 \times 10^{-3}\%$

Chelation therapy may cause significant increases in the fractions absorbed through the gastrointestinal tract.

Absorption through the skin is the least common mode of intake and is not normally a serious problem if the skin is intact. Absorption appears to be influenced by the area of skin exposed, the amount of Pu involved, and the pathological effects of the solvents on the skin. Skin permeability caused by alpha exposure may influence absorption in those cases where a high level of skin contamination is sustained for extended periods; however, currently available information is too fragmentary to permit definitive conclusions. Typical data from animal studies indicate absorption rates of 0.05% per hr for Pu in concentrated nitric acid solutions and $5 \times 10^{-5}\%$ per min in weak acid solutions. Absorption of $<2 \times 10^{-4}\%$ per hr over an 8-hr period was measured during an experiment in which 10 μg of Pu in dilute nitric acid was applied to a human's hand.

A.2 METABOLISM

The translocation of Pu compounds within the body depends on the solubility of the deposited Pu. The usual chemical and physical definitions of solubility in aqueous solutions are not adequate to predict the translocation of Pu from one site (e.g., lung, wound) into extracellular fluid and thence to other organs or tissue. In addition, experience obtained from Pu inhalation and wound cases has shown that a clear interpretation of the solubility of deposited Pu is not always possible; most cases involving exposure to so-called "insoluble" Pu oxides reveal a substantial fraction of relatively soluble Pu, which shows in early urine excreta.

The ICRP, therefore, developed in Publication 10⁽¹⁾ the terms "transportable" and "nontransportable" to describe the property which results in the relative readiness of Pu transfer across body membranes. These terms do not indicate a simple division of Pu compounds into two sharply defined classes, as the terms "soluble" and "insoluble" tend to imply. Rather, a continuous spectrum of transportability ranges from compounds that pass rapidly and completely into extracellular fluids, to those that move very slowly from the site of deposition. In this appendix, the terms "readily transportable" and "initially nontransportable" refer to those transportable fractions that move rapidly or very slowly into extracellular fluids.

The deposition, distribution and early clearance of Pu inhaled into the lung depends on the size and density of the particles, while retention and translocation depend on the chemical state of the Pu. These processes are described in the report of the Task Group on Lung Dynamics for Committee II of the ICRP.⁽²⁾

Once Pu has reached extracellular fluids, the chemical state of the compound does not appear to affect distribution within the body (except for very stable complexes formed with biologically inactive ligands such as ethylenediaminetetraacetic acid (EDTA) and diethylenetriaminepentaacetic acid (DTPA). After entry into the body by any of the four modes of intake, all Pu compounds are assumed to translocate via the circulatory system in

the following proportion: 45% to the skeleton, 45% to the liver, and 10% to other tissues and excreta. The rate of translocation may vary with the compound and route of entry, but it is assumed that the distribution from the blood will not vary significantly from these proportions. The retention time for Pu is long and the half-time retention is normally taken to be about 100 yr for the skeleton and 40 yr for the liver.

A.3 EVALUATION MODELS

Internally deposited Pu is evaluated by fitting applicable models to data from the urine, the feces, and in-vivo chest examinations. The model originally used to evaluate systemic depositions of readily transportable Pu was developed by W. H. Langham.⁽³⁾ Extensions of Langham's model have been developed for the evaluation of depositions of initially nontransportable Pu.⁽⁴⁾ The Task Group Lung Model (TGLM), which is used to evaluate depositions of Pu in the lung and to predict the translocation of Pu from the lung to other organs or to excreta, was developed by ICRP's Task Group for Committee II.^(2,5)

A.3.1 The Langham Model

Regardless of the mechanism by which internal deposition occurs, the rate of urinary excretion of the initially transportable Pu that enters the bloodstream following an acute exposure is described by the following equation:

$$X = 0.002 Q_0 t^{-0.74} \quad (1)$$

where X = the predicted urinary excretion rate of the readily transportable Pu after time t , expressed as a fraction of the initial deposition Q_0 ,

Q_0 = the initial deposition of readily transportable Pu, and

t = the number of days elapsed between deposition and sample collection.

Solving Equation (1) for Q_0 yields:

$$Q_0 = \frac{X}{0.002 t^{-0.74}} \quad (2)$$

A.3.2 The Healy Model

The model used to evaluate the amount of initially nontransportable Pu that has been or will eventually be deposited was derived by J. W. Healy.⁽⁴⁾ Similar models have been developed by Beach and Dolphin,⁽⁶⁾ but these will not be discussed in detail. Healy derived the following equation to describe the

rate of urinary excretion of initially nontransportable Pu from the lung or other metabolic pool, "isolated from the normal metabolism of the body but continually injecting plutonium into the bloodstream at a rate dependent upon the character of the deposited material and relevant physiological processes":

$$E_{\mu} = 0.002 \lambda_s Q_0 \int_0^R e^{-\lambda t} (R - t)^{-0.74} dt \quad (3)$$

where E_{μ} = the daily urinary excretion rate of initially nontransportable Pu, expressed as a fraction of the initial deposition of initially nontransportable Pu;

λ_s = the rate of solubilization and transfer to the bloodstream of initially nontransportable Pu;

Q_0 = the quantity of initially nontransportable Pu deposited in the body (and retained in the body following the initial clearance from the lung by ciliary action, in the case of deposition by inhalation);

$\lambda = \lambda_s + \lambda_c$ = the total rate of transfer of initially nontransportable Pu from the lung or other metabolic pool, due to solubilization and transfer to the bloodstream (and ciliary action following initial ciliary clearance in the case of deposition by inhalation);

R = the number of days elapsed between deposition of the Pu in the metabolic pool and the date of urine sample collection; and

t = the number of days required for initially nontransportable Pu to be transferred from the lung or other metabolic pool to the bloodstream.

Although Equation (3) is not directly integrable, the integral can be solved by series expansion of the exponential term. Healy demonstrated that λ_c , the rate of transfer of initially nontransportable Pu from the lung via ciliary action, is on the order of 10% of λ . Thus, λ_s can be assumed equivalent to λ for all practical purposes. Where the metabolic pool is a site

other than the lung, such as a puncture wound, λ_c is inapplicable and λ_s is thus exactly equivalent to λ . Substituting λ for λ_s and solving Equation (3) for Q_0 yields:

$$Q_0 = \frac{E_\mu}{0.002 \lambda \int_0^R e^{-\lambda t} (R - t)^{-0.74} dt} \quad (4)$$

Note that an estimate of the amount of initially nontransportable Pu deposited is not an estimate of the amount of initially nontransportable Pu that has been solubilized and transferred from the metabolic pool. Rather, in accordance with the theory underlying Healy's model, it is an estimate of the total amount of initially nontransportable Pu deposited which eventually will be solubilized and transferred from the metabolic pool, whether that pool is in the lung or at the site of a puncture wound.

A.3.3 The Task Group Lung Model

The TGLM, as developed and amended by ICRP's Task Group for Committee II,⁽²⁾ is based on biological data. Most of the data available were collected from animal studies with dogs, although some data have been collected from accidental exposures of humans. Studies using primates as experimental animals may result in data even closer to those from the human metabolism.

The TGLM defines classes of compounds and assigns parameters of distribution and retention based on the classes and on particle size. The classes are designated D (days), W (weeks), and Y (years), to define general clearance half-times. All compounds of Pu fall into Classes Y and W.

According to the model, the respiratory tract is divided into three regions, the nasopharyngeal (NP), the tracheobronchial (TB), and the pulmonary (P), as shown in Figure A.1. The partitioning and deposition of inhaled Pu is defined as a function of the activity median aerodynamic diameter (AMAD) of the particles, and is described by the three parameters D_3 , D_4 , and D_5 . These parameters represent the fraction of the inhaled material,

FIGURE A.1. TGLM Clearance Model

Q_I , initially deposited in each of the three regions. The chemical nature of the Pu involved is assumed to have an insignificant influence on partitioning, but does affect retention in the lung. Specific clearance classifications for numerous compounds are given in the model reference.⁽²⁾

The schematic portion of Figure A.1 shows the clearance pathways, (a) through (i), from the three respiratory regions to the gastrointestinal (GI) tract, the lymph (L) and the blood. The tabular portion of the figure gives for each class of material (D, W, Y) and for each clearance pathway (a-i), the values for clearance half-time, T_{a-i} , and the fraction of regional deposition subject to a particular clearance pathway, f_{a-i} . The values given for regional deposition (D_3 - D_5) are based on an aerosol with an AMAD of 1 μm . As the particle size varies from 1 μm , the regional deposition factors vary as shown in Figure A.2. The TGLM is intended for use with aerosol distributions that have an AMAD between 0.2 and 10 μm and a standard deviation less than 4.5. The dashed lines of Figure A.2 show provisional estimates of deposition that further extend the size range. For the unusual distribution with an AMAD greater than 20 μm , complete N-P deposition can be assumed. Aerosols with AMADs below 0.1 μm (e.g., radionuclei and fumes) cannot be applied to the TGLM.

FIGURE A.2. Regional Deposition as a Function of Particle Size

A.4 EVALUATION TECHNIQUES

The evaluation of any deposition of Pu is a process of evolving or refining the evaluated deposition as data become available. The initial data base consists of the results of prompt evaluations. As this data base expands, more sophisticated and accurate evaluations can be made. The goal is to collect enough data so that the appropriate model or models can be accurately and logically applied and the most realistic evaluation made.

The specific functions of interim evaluations are to monitor the effectiveness of any therapy that has been administered, determine the adequacy of the sampling regimen, keep management aware of the current status of the case, and determine the need for work restrictions. The evaluation techniques used are based on incoming in-vivo, fecal, and urinary data.

A.4.1 In-Vivo Data

The deep pulmonary deposition (that fraction of material not cleared rapidly to the GI tract) is assumed to clear with half-times of 50 days for Class W compounds and 500 days for Class Y compounds. Two-thirds of this clearance is via the gastrointestinal tract; the other third is split between blood and lymph nodes, with three-fourths of a W compound going to blood, and three-fourths of a Y compound going to lymph nodes. The rapid clearance rates, the deposition in the nasopharyngeal (N-P) and tracheo-bronchial (T-B) regions and 40 percent of the pulmonary (P) deposition (Classes W and Y) are not a function of the compound class. The route of clearance is, however, determined by the compound class: the more soluble or transportable the compound, the larger the fraction cleared to the blood and the smaller the fraction cleared via the gastrointestinal tract.

The TGLM can be a useful tool in evaluating both lung and systemic depositions. The ICRP states in Publication 19⁽⁵⁾ that the long-term distribution and retention are more important than the early clearance mechanisms in estimating radiation dose to the lung from inhaled Pu. In-vivo chest examinations conducted during the first 7 days post-intake include decreasing fractions of the short-term component plus any long-term component. Chest examinations

conducted after 7 days should by definition include only that portion of the pulmonary deposition that will clear with a long half-time. Data collected after the seventh day post-intake may be extrapolated back to day zero to determine the initial magnitude of the component of long-term clearance. Dividing this component by 0.6 yields an estimate of the initial total pulmonary burden, as shown in Figure A.3.

The initial pulmonary deposition obtained as just described may be used to calculate the ultimate systemic burden. The relationship between initial pulmonary and ultimate systemic depositions for varying particle size is shown in Figure A.4.

FIGURE A.3. Theoretical Pulmonary Clearance for Pu Oxide

A.4.2 Fecal Data

Estimates of the initial systemic and pulmonary burden can be made as early as 1 week post-intake. The estimates are based on early clearance from the N-P, T-B, and P regions to the GI tract. They may be made using gamma scans to measure the ²⁴¹Am content, if present, in first week fecal samples, applying the appropriate Pu alpha/Am alpha ratio, and then using the relationships presented in Figure A.5. When data on the Pu content of fecal samples become available later, the early estimates can be refined.

FIGURE A.4. Systemic Burden as a Function of Particle Size, for an Initial Pulmonary Deposition of 1 nCi (Based on TGLM; for Pu Oxide Only)

FIGURE A.5. Initial Pulmonary and Systemic Burdens as a Function of Particle Size, for 1 nCi of Pu Excreted in the Feces During Week 1 Post-Intake (Based on TGLM; for Pu Oxide Inhalation Only)

Delayed fecal data (>10 days post-intake) may be used to calculate the current pulmonary burden based on the slow clearance from the pulmonary area to the GI tract. Caution must be exercised in using these data, however, because small intakes that may be incurred just prior to sampling can greatly bias the data. Delayed fecal samples must be obtained following at least 1 week's absence from any Pu exposure.

A.4.3 Urinary Data

Interim evaluations based on data from urine samples may be made initially by applying Langham's model to the first available data. As more data become available, other models that more adequately define the deposition can be used. The options available include Healy's model, the Beach-Dolphin model, the TGLM, or a combination of two or more of these models.

A.5 APPLICATION OF MODELS

This section contains basic information on Langham's model, Healy's model, and the Task Group Lung Model, and relates to the evaluation of a systemic deposition or that portion of a plutonium deposition that has reached extracellular fluids and/or lymphatic tissues.

A.5.1 Langham's Model

Equation (1) (see page A-5) indicates that the rate at which Pu is excreted in the urine decreases logarithmically as the number of days between deposition and sample collection increases. For a given value of Q_0 (measured in dis/min), Equation (1) can be expressed graphically as a straight line with a slope of -0.74 when paired values of t and X are plotted logarithmically. In practice, the value $t = 1$ day and the corresponding value of X are substituted into Equation (2), since when $t = 1$ day the equation reduces to the simple form $Q_0 = \frac{X}{0.002}$. It is important to note that the value for $t = 1$ day is obtained graphically. The data obtained on the first day post-intake may be biased high and should not be used for final evaluation purposes. Figure A.6 presents an example to illustrate the use of Langham's model for initially transportable Pu.

Although Langham's model was originally designed to evaluate intakes of readily transportable Pu, several facilities also utilize Langham's model for evaluations of initially nontransportable Pu. This procedure involves applying series of Langham's curves to the data, assuming a new intake each time the measured excretion rate exceeds the levels predicted from the previous sample. The total systemic burden as a result of the intake is the sum of these evaluations.

A.5.2 Healy's Model

Since Healy's model is not integrable, it must be solved by a series expansion of the exponential term. One method of utilizing Healy's model to evaluate depositions of initially nontransportable Pu is to prepare λ curves.

FIGURE A.6. Sample Application of Langham's Model for Urinary Excretion of Initially Transportable Pu

For each λ curve, λ is arbitrarily assigned a value. Values of the integral are then determined corresponding to the value assigned to λ and the several values assigned to R , the number of days elapsed between deposition in the metabolic pool and subsequent excretion in the urine. The values of the integral have no units, but express the relative excretion rates at various times R after deposition, predicted by the integral for the assumed value of λ . Thus, plotting logarithmically the values of the integral versus the corresponding values of R yields a λ curve, for the value of λ assumed, which represents the shape of the excretion curve predicted by Equation (4) regardless of the magnitude of Q_0 .

The appropriate value of λ to use in evaluating the amount of initially nontransportable Pu deposited is thus determined by logarithmically plotting the Pu urinary excretion data and experimentally determining which λ curve best fits the plotted data.

Equation (4) (page A-7) can then be solved for Q_0 by substituting for E_{μ} and then the integral a pair of values corresponding to any given number of days post-intake. In practice it is convenient to choose the maximum value of the integral and the corresponding maximum value of E_{μ} , the maximum excretion rate as determined by the λ curve fitted to the plotted analytical data. Figure A.7 illustrates the use of Healy's model for initially nontransportable Pu.

FIGURE A.7. Sample Application of Healy's Model for Urinary Excretion of Initially Nontransportable Pu

A.5.3 Task Group Lung Model

Based on TGLM concepts and Langham's model, Nelson⁽⁷⁾ derived the relationships between urinary data and systemic and lung depositions shown in Figures A.8 and A.9, respectively. For Pu nitrate, excretion in urine varies with particle size to approximately 1 month post-intake. After 1 month, particle size has no significant effect. The same trend is exhibited for Pu oxide, except that particle size affects excretion for about 1 year post-intake.

FIGURE A.8. Urinary Excretion of Pu as a Function of Time for Various AMADs, for an Inhalation Resulting in a Systemic Burden of 1 nCi

FIGURE A.9. Urinary Excretion of Pu as a Function of Time for Various AMADs, for an Initial Lung Burden of 1 nCi

A.5.4 Factors Affecting Model Application

Several additional factors must be considered in the evaluation of plutonium deposition:

- sample collection periods may vary;
- early treatment with DTPA will affect the accuracy of initial estimates of the systemic burden;
- the estimate may need adjustment to account for previous intakes of Pu; and
- the ^{241}Pu content of the deposition must be calculated or measured.

Sample Collection Periods

The models for evaluating readily transportable and initially nontransportable Pu both involve the daily urinary excretion rate of Pu, that is, the amounts of Pu excreted in the urine during 24-hr periods. Hence, the urine samples analyzed must be representative of such Pu excretion. In some cases the person to be sampled is instructed to collect all urine voided during a 24-hr period. In other cases the person may be requested to collect a "simulated 24-hr urine sample" (sometimes referred to as a "48-hr urine sample"). Such a sample consists of all urine voided 1/2 hr before retiring and 1/2 hr after rising for 2 consecutive 24-hr periods and is assumed to be equivalent in Pu content to a 24-hr total urine sample collected on the middle one of the three dates spanned by this sampling period.

DTPA Interference

Early treatment with DTPA is necessary for maximum effectiveness following Pu intake. Thus, urinary excretion data not biased by DTPA are normally not available for interpretation during the treatment period. Case data have shown that treatment with DTPA immediately following an intake through a wound will increase the excretion of Pu in the urine by a factor of approximately 60 to 100 on the day of treatment. By assuming an increased excretion factor of 50, an early estimate of the original systemic burden can be made, but the estimate will be crude at best.

Once treatment is concluded, reasonable estimates of the effectiveness of treatment cannot be made for at least 20 days. Even though estimates are made after 20 days, it may take 100 days or more after treatment has been discontinued for urinary excretion data to stabilize to the rate predicted by appropriate models.

The data presented in Figure A.10 are a good example of a case treated with DTPA. The deposition estimated from data collected 25 to 120 days after the termination of DTPA treatment (130 to 220 days post-intake) was 18 nCi. Data collected 200 to 920 days after DTPA treatment gave an evaluation of 10 nCi or approximately half of the first estimate. This case illustrates the need for caution when evaluating a deposition following treatment with DTPA, to assure that the data are no longer affected by the treatment.

FIGURE A.10. Urinary Excretion Data and Deposition Estimates: Pu Nitrate Wound Treated with DTPA

Adjustments For Previous Intakes

In many cases the person being evaluated for a potential internal deposition from a current incident has previously received an internal deposition. In such cases, urine samples may contain Pu from the previous deposition, causing the urinalysis results to show higher amounts of Pu than they would from the current incident only. To avoid overestimation of the current deposition, the data must be corrected to account for the previous deposition: subtract from current data the amount of Pu predicted (at the time of current sampling) by the excretion curve(s) for the prior deposition(s).

In addition, when the data indicate that a person has received depositions of both readily transportable and initially nontransportable Pu from the same incident, the data must be corrected. Here, adjustment must be made for the influence of the readily transportable Pu before the amount of initially nontransportable Pu deposited is evaluated.

Analysis of ^{241}Pu Content

When the evaluation of a deposition is based on urinalysis data, only the alpha-emitting isotopes of Pu are generally considered. Analysis for the weak beta emission from ^{241}Pu is not performed because the detection level is too high for practical use. The ^{241}Pu fraction of the deposition, therefore, must be calculated from data on isotopic composition or measured using special urinalysis techniques (practical only in higher deposition cases).

A.6 CALCULATION OF DOSE COMMITMENT

There has been an increased interest of late in determining the organ dose commitments from internally deposited radionuclides. Computer programs have been developed to perform this task by evaluating data from in-vivo examinations and bioassay sampling, using the various models discussed earlier.

The term "dose commitment" is defined as the dose that would result from an intake of Pu and is usually calculated for either a 1-yr or a 50-yr period. Dose commitments are generally calculated by substituting the appropriate functions, parameters, and numerical values into the dose equation presented in ICRP Publication 10:⁽¹⁾

$$\text{Dose (in rem)} = 5.12 \frac{\epsilon}{m} Q_0 \int_0^t e^{-\lambda t} dt \quad (1)$$

where ϵ = effective absorbed energy per disintegration

m = mass of critical organ in grams

Q_0 = burden in critical organ

λ = decay constant

t = time in days.

One of the major difficulties in calculating dose commitments is determining to which organ to assign the systemic burden or the chest burden. The organs of primary concern when calculating dose commitments are the bone, lung, and liver. For systemic depositions, the deposition is assumed to be split equally between bone and liver. Using this 50% - 50% division, the error in percent of deposition in bone should be within a factor of 2. The clearance rates from the bone and liver are extremely slow; the currently accepted half-times in bone and liver are 100 yr and 40 yr, respectively.

An additional factor that must be considered is the build-up of the systemic deposition over time, from external pools such as the lungs or wounds contaminated with initially nontransportable Pu. If Healy's model is used, this build-up is automatically accounted for.

The determination of dose commitments to lung tissue is complicated by an inability to distinguish between lung, lymph, and bone burdens when measuring activity in the chest by in-vivo techniques. Clearance half-times from the lung vary with solubility classification. In addition, as half-times in the pulmonary region increase, the fraction to the lymph increases.

The varying clearance rates from the lung, coupled with the ingrowth of ^{241}Am from mixtures with a high content of ^{241}Pu , require careful monitoring of the employee's deposition. Calculations of dose commitments must take into account the slow build-up of the systemic burden as the lung clears, or else the dose commitment must be updated frequently during the first few years post-intake.

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