ORAU Team **Dose Reconstruction Project for NIOSH** Technical Basis Document for Paducah Gaseous Diffusion Plant – Occupational Internal Dose	Document Number: ORAUT-TKBS-0019-5 Effective Date: 09/30/2004 Revision No.: 00 Controlled Copy No.: _____ Page 1 of 26
Subject Expert: Carol D. Berger Document Owner Approval: Signature on File Date: 09/22/2004 Jay J. Maisler, TBD Team Leader Approval: Signature on File Date: 09/20/2004 Judson L. Kenoyer, Task 3 Manager Concurrence: Signature on File Date: 09/20/2004 Richard E. Toohey, Project Director Approval: Signature on File Date: 09/30/2004 James W. Neton, Associate Director for Science	Supersedes: None

TABLE OF CONTENTS

LIST OF TABLES

RECORD OF ISSUE/REVISIONS

ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	11/24/2003	00-A	New technical basis document for the Paducah Gaseous Diffusion Plant – Occupational Internal Dose. Initiated by Jay J. Maisler.
Draft	07/21/2004	00-B	Incorporates internal and NIOSH comments and Gaseous Diffusion Plant consistency review issues. Initiated by Jay J. Maisler.
Draft	08/11/2004	00-C	Incorporates internal and NIOSH review comments. Initiated by Jay J. Maisler.
Draft	08/30/2004	00-D	Incorporates internal and NIOSH review comments. Initiated by Jay J. Maisler.
Draft	09/03/2004	00-E	Incorporates internal and NIOSH review comments. Initiated by Jay J. Maisler.
Draft	09/03/2004	00-F	Incorporates internal and NIOSH review comments. Initiated by Jay J. Maisler.
09/30/2004	09/30/2004	00	First approved issue. Initiated by Jay J. Maisler.

ACRONYMS AND ABBREVIATIONS

AMAD	activity median aerodynamic diameter
Bq	becquerel
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
g	gram
hr	hour
ICRP	International Commission on Radiological Protection
KPA	kinetic phosphorescence analysis
L	liter
MDA	minimum detectable amount
MDC	minimum detectable concentration
mg	milligram
ml	milliliter
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
ORNL	Oak Ridge National Laboratory
pCi	picocurie
PGDP	Paducah Gaseous Diffusion Plant
ppm	parts per million
TRU	transuranic
U.S.C.	United States Code
µCi	microcurie
µg	microgram
µm	micrometer

5.1 INTRODUCTION

Technical Basis Documents and Site Profile Documents are general working documents that provide guidance concerning the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist the National Institute for Occupational Safety and Health (NIOSH) in the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer facility" or a "Department of Energy facility" as defined in the *Energy Employees Occupational Illness Compensation Program Act of 2000* [EEOICPA; 42 U.S.C. Sections 7384l(5) and (12)].

This document provides a uniform and consistent approach to assessing occupational internal dose at the Paducah Gaseous Diffusion Plant (PGDP) for the dose reconstructions for NIOSH in relation to the EEOICPA. The document provides guidance to dose reconstructors on input parameters that are specific to employees of PGDP, as well as the approach for employees with either missing or no monitoring information.

In 1951, the first operator of the facility, Union Carbide, began hiring and construction began. In 1952, the first four process buildings began operations. From then until 1977, uranium hexafluoride (UF_6) feed material was produced from uranium trioxide (UO_3) at the plant. From 1953 to 1964, and then again from 1968 to 1977, UF_6 was produced from the UO_3 in spent reactor fuel. In May of 1977, the feed plants ceased operation and all feed to the enrichment process was in the form of UF_6 obtained from outside sources.

Over the years, workers at PGDP handled mainly UF_6 and slightly oxidized forms of uranium. The facility processed both virgin feed material and recycled or reprocessed reactor fuel to enrichments of up to 5% (by weight) of ^{235}U in the final product.[1] This processing involved small quantities of transuranic (TRU) elements, primarily neptunium and plutonium, present in various workplaces, as well as the thorium and protactinium progeny of uranium and the fission product ^{99}Tc.

The primary method of monitoring employees for intakes of radionuclides at PGDP was urine bioassay. Bioassay monitoring was instituted at the start of enrichment operations and has continued to the present. However, the focus of the monitoring program in the very early years was the detection of excreted soluble uranium. When monitoring for less soluble isotopes of uranium and TRU elements was necessary, *in vivo* methodologies were implemented, primarily whole-body counting and chest (lung) counting.

Until the mid-1980s, action levels were set based on the amount of uranium excreted. Later, intakes and doses were assessed based on both in vivo and in vitro monitoring results, using DOSEXPRT, a computer program developed by Oak Ridge National Laboratory (ORNL).[2] Data are available from 1952 to the present for both *in vivo* and *in vitro* analysis records and associated interpretations.

[1] The most predominant enrichment level of the final product was 1.5%. Reprocessed fuel was used as feed from 1952 until 1976. At that time, the cascade facilities were upgraded and most of the TRU and fission product materials were removed. Campaigns involving reprocessed fuel elements ended in the 1980s. In 2000, final products were enriched to the maximum value of 5%.

[2] As shown in Eckerman and Ward (1992), DOSEXPRT Version 4.2 was used to analyze intake and dose for PGDP personnel for 1991. Version 4.1 was used for analysis of 1990 bioassay data, and Version 3.0 was used for analysis of 1989

A review of in-house procedures used to assess the concentration of uranium in urine indicates that a variety of quality control steps were an integral part of the process. For example, duplicates were consistently run, and comparison of results to known quantities was a critical step. Therefore, the *in vitro* results from in-house processing, typically reported in units of micrograms of uranium per liter, can be considered generally reliable. However, interpretation of those results can be difficult, primarily because of uncertainty about enrichment, solubility, and the contribution of environmental uranium, but also because samples were collected at work and during the middle of the workweek, meaning that cross-contamination and the inability to separate soluble from insoluble intake fractions contribute to the uncertainty.

Nonetheless, dose reconstructors can prepare reliable estimates of dose from the dates of employment, the employment locations, and the urine bioassay results. (*In vivo* results, because they were acquired primarily in response to an incident, are less reliable for assessing routine intakes.) Assumptions such as absorption types and the presence or absence of TRU elements can be derived from the historical records.

Section 5.2 provides guidance on selection of source terms. Sections 5.3 and 5.4 involve interpretation of *in vivo* and *in vitro* measurement results, respectively, each including instructions for assessing dose for both monitored and unmonitored employees. Section 5.5 identifies significant incidents with internal dose potential.

5.2 SOURCE TERM

The mission of PGDP was to enrich uranium in the form of UF_6 (for use in domestic and foreign commercial power reactors) from roughly 0.7% ^{235}U (natural enrichment) to 2.5% ^{235}U (DOE 2000, p. 6). In addition, other compounds of uranium were present throughout the plant's history, including UO_2F_2, UF_4, and UO_3. The primary radionuclides of concern for the plant are ^{238}U, ^{235}U, and ^{234}U. The progeny of dosimetric interest for these radionuclides includes ^{230}Th and ^{234m}Pa (DOE 2000, p. 24).

Certain TRU isotopes have been a factor at PGDP, including ^{237}Np and ^{239}Pu. These are the result of the processing of reactor tails. Reactor tails were fed to the cascade from 1953 to 1964, and again from 1969 to 1974 (with the exception of 1971 when none of the feed was of reactor origin) (Smith 1984, p. 9). When this processing was underway, approximately 19% of the feed material in use at the plant was reactor tails (DOE 2000, which included both neptunium and plutonium). Table 5-1 is a summary of the percentages of TRU materials as assessed at specific facilities (IT Corporation 1992).

Table 5-1. Elemental fractions of uranium, neptunium, and plutonium at specific facilities.[a,b]

Plant location	Percent uranium	Percent neptunium	Percent plutonium
Feed plant (general)	100	0	0
Cascade (general)	100	0	0
Bldg. C-333	85.9	11.0	3.3
Bldg. C-720	82.9	16.4	0.7
Bldg. C-337	76.3	22.8	0.98
Bldg. C-400	93.1	6.9	0
Bldg. C-409	76.3	22.0	1.7

a. Elemental fractions, in this case, are presumed to refer to the percentage of total alpha activity.
b. See Table 5-4 for activity concentration of impurities.

data. The method for computing intake, committed dose, and annual dose did not change between Versions 3.0 and 4.2, although a faster algorithm was developed for computation of annual dose for Version 4.1.

It is important to note that this table presents only a snapshot of TRU percentages over time. However, dose reconstructors can apply these percentages to data collected *after* 1953.

At the PGDP, monitoring for intakes of uranium, whether *in vivo* or *in vitro*, often resulted in reports of elemental uranium concentration in urine or the mass of elemental uranium in organs or the whole body. However, internal dose assessment requires the use of isotopic concentrations as input to the assessment process. Therefore, reconstructors should use Table 5-2 to derive the isotopic fractions associated with each microgram of uranium reported in an analytical result. In addition, the table provides default values if the specific location where a claimant worked is not available.

Table 5-2. Isotopic fractions for various enrichment percentages.[a]

Enrichment	Isotopic fractions (Bq of isotope/µg total U)			
	U-234	U-235	U-236	U-238
Natural uranium	1.24E-02	5.50E-04	1.67E-05	1.24E-02
93% enriched feed	2.41E+00	7.38E-03	4.90E-03	6.83E-04
4% enriched feed	6.42E-02	3.17E-03	6.67E-05	1.18E-02
Typical commercial (3% enriched) feed	6.55E-02	2.99E-03	5.00E-05	1.19E-02
3.5% enriched feed	6.67E-02	2.80E-03	5.84E-05	1.20E-02
Low-enrichment (2% enriched) feed	2.52E-02	1.58E-03	3.33E-05	1.21E-02
Typical depleted uranium	2.30E-03	1.59E-04	2.40E-06	1.24E-02
Recycled Uranium	Table 5-3	Table 5-3	Table 5-3	Table 5-3
Default	2.52E-02	1.58E-03	3.33E-05	1.21E-02

a. Sources: ANSI (1995); ORAU (2003); IMBA computer software (Birchall et al. 2003); Berger (2004).

Dust samples from a variety of process systems in 1989 were analyzed for their isotopic content. Table 5-3 summarizes the results for selected process areas at the plant that dose reconstructors can use in the absence of employee-specific information to evaluate the total uranium bioassay data collected *after* 1953 (Baker 1987, pp. 6 to 8). However, reconstructors should use these values cautiously as they present only a snapshot of information.

Table 5-3. Transuranic element concentrations in plant aerosols (1987).

Location	Pu-239		Np-237		U-238		U-234		Th-230	
	dpm/g U	nCi/µg U	dpm/g U	nCi/µg U	dpm/g U	nCi/µg U	dpm/g U	nCi/µg U	dpm/g U	nCi/µg U
UO_3 Powder Area	500	0.225	270	0.122	750,000	337.5	750,000	337.5	42,000	18.9
C-420 Green Salt Plant	500	0.225	270	0.122	750,000	337.5	750,000	337.5	42,000	18.9
Fluorination Tower Area	150,000	67.5	12,000	5.4	750,000	337.5	750,000	337.5	42,000	18.9
Cold Trap and Refrigeration Operation	120,000	54	9,000	4.05	750,000	337.5	750,000	337.5	34,000	15.3
C-410 Control Room Operations	150,000	67.5	12,000	5.4	750,000	337.5	750,000	337.5	42,000	18.9

The fission product ^{99}Tc has also been present during plant operations, particularly during the processing of reactor tails. The available documentation indicates that the tails contained from 0.041 to 7.0 ppm technetium (Smith 1984, Appendix 12; DOE 2000).

For dose assessment purposes, a nominal distribution of radionuclides must be assumed because not all analytical methods were capable of detecting many of the radionuclides in the PGDP source term. If only *total* uranium results are available for a particular measurement result, Table 5-4 provides a default isotopic distribution.

Table 5-4. Default isotopic distribution.

Radionuclide	nCi/g U
Pu-239	67.5
Am-241/Pu-241	67.5
U-236	0.93
U-235	43.9
U-234	702.0
U-238	337.5
Np-237	5.4
Th-230	18.9
Tc-99	0.12

At the PGDP, and unless site-specific information is available, the particle size is assumed to be 5 µm aerodynamic median activity diameter

Table 5-5. Facility-specific source radionuclides with solubility type and particle size.[a,b,c]

Facility	Compound	Radionuclide	Absorption type	Particle size (µm AMAD)[c]
C-331 Process Building	UF_6	U-234	F	5
		U-235	F	5
		U-238	F	5
		Np-237	M	5
		Pu-239	S	5
		Tc-99	F	5
C-333 Process Building	UF_6	U-234	F	5
		U-235	F	5
		U-238	F	5
		Np-237	M	5
		Pu-239	S	5
		Tc-99	F	5
C-335 Process Building	UF_6	U-234	F	5
		U-235	F	5
		U-238	F	5
		Np-237	M	5
		Pu-239	S	5
		Tc-99	F	5
C-337 Process Building	UF_6	U-234	F	5
		U-235	F	5
		U-238	F	5
C-310 Process Product Withdrawal	UF_6	U-234	F	5
		U-235	F	5
		U-238	F	5
		Th-230	S	5
		Pa-234m	F	5
		Tc-99	F	5
C-315 Process Tails Withdrawal	UF_6	U-234	F	5
		U-235	F	5
		U-238	F	5
		Th-230	S	5
		Pa-234m	F	5
C-337A Feed Materials Process Area	UO_2F_2	U-234	F, M	5
		U-235	F, M	5
		U-238	F, M	5

Table 5-5 (Continued). Facility-specific source radionuclides with solubility type and particle size.[a,b,c]

Facility	Compound	Radionuclide	Absorption type	Particle size (µm AMAD)[c]
C-333A Feed Materials Process Area	UO_2F_2	U-234	F, M	5
		U-235	F, M	5
		U-238	F, M	5
C-720 Maintenance Building	UO_2F_2	U-234	F, M	5
		U-235	F, M, S	5
		U-238	F, M, S	5
		Np-237	F, M, S	5
		Pu-238	S	5
		Pu-239	S	5
		Pu-240	S	5
		Am-241	M	5
		Th-230	S	5
		Pa-234m	F	5
C-405	Ash handling	U-234	F, M, S	5
		U-235	F, M, S	5
		U-238	F, M, S	5
		Np-237	F, M	5
		Pu-239	S	5
		Tc-99	F	5
C-410	UO_2, UO_3, UF_4, UF and UF_6	U-234	F, M, S	5
		U-235	F, M, S	5
		U-238	F, M, S	5
		Np-237	F, M	5
		Pu-239	S	5
		Tc-99	F	5
C-420	UF_4, UO_3, UO_2, and UF	U-234	F, M, S	5
		U-235	F, M, S	5
		U-238	F, M, S	5
		Np-237	M	5
		Pu-239	S	5
		Tc-99	F	5
C-409 Decontamination Building	UF_6, UO_2F_2	U-234	F, M	5
		U-235	F, M	5
		U-238	F, M	5
		Np-237	F, M	5
		Pu-239	S	5

Table 5-5 (Continued). Facility-specific source radionuclides with solubility type and particle size.[a,b,c]

Facility	Compound	Radionuclide	Absorption type	Particle size (µm AMAD)[c]
C-400 Decontamination Building	UF_6, UO_2F_2	U-234	F, M, S	5
		U-235	F, M, S	5
		U-238	F, M, S	5
		Np-237	F, M, S	5
		Pu-238	S	5
		Pu-239	S	5
		Pu-240	S	5
		Am-241	M	5
		Th-230	S	5
		Pa-234m	F	5
C-340 Metal facility	UF_6, UF_4, UO_2	U-234	M, S	5
		U-235	M, S	5
		U-238	M, S	5
		Th-230	S	5
		Pa-234m	F	5
C-746Q Waste Building	UF_6	U-234	F, M	5
		U-235	F, M	5
		U-238	F, M	5
		Np-237	F, M	5
		Pu-238	S	5
		Pu-239	S	5
		Pu-240	S	5
		Am-241	M	5
		Tc-99	F	5
C-746A Waste Building	UF_6	U-234	F, M	5
		U-235	F, M	5
		U-238	F, M	5
		Np-237	F, M	5
		Pu-238	S	5
		Pu-239	S	5
		Pu-240	S	5
		Am-241	M	5
		Tc-99	F	5

Table 5-5 (Continued). Facility-specific source radionuclides with solubility type and particle size.[a,b,c]

Facility	Compound	Radionuclide	Absorption type	Particle size (µm AMAD)[c]
C-746B Waste Building	UF_6	U-234	F, M, S	5
		U-235	F, M, S	5
		U-238	F, M, S	5
		Np-237	M	5
		Pu-238	S	5
		Pu-239	S	5
		Pu-240	S	5
		Am-241	M	5
		Tc-99	F	5
C-404 Burial Ground	UF_4	U-234	F, M	5
		U-235	F, M	5
		U-238	F, M	5
C-745A-T Cylinder Yards		U-234	F	5
		U-235	F	5
		U-238	F	5
C-710 Laboratory	Neptunium recovery	U-234	F, M	5
		U-235	F, M	5
		U-238	F, M	5
		Np-237	M	5
		Pu-238	S	5
		Pu-239	S	5
		Pu-240	S	5
		Am-241	M	5
		Tc-99	F	5
		Th-230	S	5
		Pa-234m	F	5
C-360 Toll Transfer and Sampling Building		U-234	F, M, S	5
		U-235	F, M, S	5
		U-238	F, M, S	5
		Np-237	F, M	5
		Pu-238	S	5
		Pu-239	S	5
		Pu-240	S	5
		Am-241	M	5
		Tc-99	F	5

Table 5-5 (Continued). Facility-specific source radionuclides with solubility type and particle size.[a,b,c]

Facility	Compound	Radionuclide	Absorption type	Particle size (µm AMAD)[c]
Default	Not applicable	U-234	F, M, S	5
		U-235	F, M, S	5
		U-238	F, M, S	5
		Np-237	F, M, S	5
		Pu-238	S	5
		Pu-239	S	5
		Pu-240	S	5
		Am-241	M	5
		Th-230	S	5
		Pa-234m	F	5
		Tc-99	F, M	5

a. As of the date of this report, there is little information to support a reduction in the types or forms of radionuclides listed in this table on an annual basis, and few effective decommissioning efforts that could have resulted in downgrading these radiologically restricted areas. Therefore, it is assumed that all of the radionuclides shown were present in the buildings beginning in 1953. Before 1953, reactor tails were not used as feed to the cascade, and the TRU materials do not apply.

b. Data and information on source terms and solubility types are from Hill and Strom (1993, Table 16.2), PACE et al. (2000, Table 2.4-1), DOE (2000, Table B-1), and BJC (1999, Table 1.7). (These documents also describe a single particle size study of unknown duration and procedure that shows particle sizes of 3 µm AMAD for ^{237}Np and certain plutonium isotopes in Buildings C-720, C-400, C-476Q, C-746A, C-746B, C-710, and C-360. Because of the limitations of the study, the findings were not incorporated into the table.) Where no facility-specific information was available, the values are from the recommendations in ICRP (1997).

c. Unless site-specific information is available, the particle size is assumed to be 5 µm AMAD, as recommended in ICRP (1994, paragraph 5).

(AMAD), as recommended by the International Commission on Radiological Protection (ICRP 1994, paragraph 5). If information on the source term to which the employee was exposed is available, the dose reconstruction should use that source term. However, if no source term information is available, the values and parameters in Table 5-5 provide conservative input to the process.

5.3 *IN VITRO* MEASUREMENT METHODS

From the start of plant operations in 1952, samples of urine from workers involved in enrichment operations were analyzed for uranium.[3] Over time, other workers were included in the monitoring program. In addition, special sampling occurred in response to incidents or issues (i.e., assessments of the radiological impacts of TRU elements and technetium); however, those analytical methods were typically performed off the PGDP site, at ORNL during the early periods and later at analytical services contractor locations). Table 5-6 lists the sampling frequencies and volumes for workers assigned to specific buildings during various periods. In addition, the table provides default values if the specific location of a claimant is not available.

Table 5-6. *In vitro* measurement frequencies.[a, b]

Period	Facility	Frequency	Volume
1953 to present day	C-310	Every 4 weeks	Spot
1953 to present day	C-315	Every 4 weeks	Spot
1953 to present day	C-340	Every 4 weeks	Spot
1953 to present day	C-400	Every 4 weeks	Spot
1953 to present day	C-410	Every 4 weeks	Spot
1953 to present day	Remainder	Yearly	Spot
1959 to present day	All	Within 0.5 hr of incident	Spot
1960	C-410	Every 2 months	Spot
1960	C-340	Every 3 months	Spot
1960	C-331, C-333, C-335, C-337	Every 4 months	Spot
1960	C-410	Every 6 months	Spot
1991 to present day	All	Every 4 weeks	24-hr collection or simulated 24-hr collection
Default	N.A.	Every 4 weeks	Spot

a. Sources: PACE et al. (2000, Section 7.4); PGDP (2003a).
b. There is evidence that in 1960 the measurement frequency for Building C-410 and C-340 workers could have been either every four weeks or every two months (C-410) or four months (C-340). However, there is no referenceable indication as to whether there was a special class of worker to which the two-month frequency applied. The overlapping values have been left in the table.

At PGDP, routine urine samples were typically single voids collected during the middle of the week. Each could have been measured for specific gravity, pH, sugar, and albumin levels, as well as for uranium content. During the early years, total uranium concentrations were measured using a calibrated fluorimeter with a detection level of 0.005 mg/L. In later years total uranium content was assessed by kinetic phosphorescence analysis (KPA). Neither methodology included isotopic determinations (PACE et al. 2000, Section 4.2.1). To account for fluid mass balance over a 24-hr period in spot samples, the methodology presented by the National Council on Radiation Protection and Measurement (NCRP) Report 87 may be used (NCRP 1987). If the specific gravity was measured and reported for any given spot sample, the following equation may be used to derive the daily urinary excretion rate:

[3] Fecal sampling was occasionally performed for special studies. However, a program for routine or diagnostic monitoring of fecal samples was never implemented.

$$C_{cor} = C_m \times \frac{(1.024 - 1)\ \text{g/mL}}{(SG_m - 1)\ \text{g/mL}} \quad (5\text{-}1)$$

where

C_{cor} = the corrected radionuclide concentration in the urine sample
C_m = the measured concentration
SG_m = the measured specific gravity of that sample

The equivalent 24-hr excretion is then:

$$A_u\left(\frac{pCi}{24\ \text{hr}}\right) = C_{cor}\left(\frac{pCi}{\text{L}}\right) \times \frac{1.4\ \text{L}}{24\ \text{hr}} \quad (5\text{-}2)$$

In cases where the specific gravity is unknown, the measurement result should be normalized to activity per day, as follows, by assuming a reference value V of 1.4 and volume L of 1.0 L:

$$A_u\left(\frac{pCi}{24\ \text{hr}}\right) = C_m\left(\frac{pCi}{\text{L}}\right) \times \frac{VL}{24\ \text{hr}} \quad (5\text{-}3)$$

5.3.1 Measurement Types and Detection Levels

Table 5-7 lists the *in vitro* measurement types and detection levels during various periods. For samples analyzed at the site, the detection level for total uranium in urine was reported as 0.005 mg/L (PACE et al. 2000, Section 4.2.1). If referenceable detection levels were not available, those specified as typical in ICRP (1988) were used.

Table 5-7. *In vitro* measurement types and detection levels for various periods.[a]

Period	Measurement type	Radionuclide	MDC[b] (mg/L)	Recall level (µg/L)
1952 to 1961	In-house fluorimetry	Total uranium	0.01	10
1962 to 1976	In-house fluorimetry	Total uranium	0.01	60 (single sample)
1962 to 1976	In-house fluorimetry	Total uranium	0.01	33 (quarterly average)
1962 to 1969	In-house fluorimetry	Total uranium	0.01	12 (three consecutive samples)
1977 to 1982	In-house fluorimetry	Total uranium	0.01	33 (single sample)
1982 to present	KPA	Total uranium (soluble, and less than 5% U-235)	0.005	50
1982 to present	KPA	Total uranium (insoluble)		20
1969 to 1985	In-house	Gross beta (dpm L^{-1})		13 dpm ml^{-1}
1985 to present	ORNL	Isotopic plutonium	1 pCi/L	
1985 to present	ORNL beta counting	Tc-99	10 dpm/ml	
1989 to present	Contractor	U-234, U-235, and U-238	0.3 pCi/L	
1999 to present	ORNL	Natural uranium	0.06 mg/sample	
1999 to present	ORNL	Tc-99	16 pCi/sample	
1999 to present	ORNL	U-234, U-235, and U-238	0.014 pCi/sample	
1999 to present	ORNL	Th-228, Th-230, and Th-232	0.014 pCi/sample	
1999 to present	ORNL	Np-237	0.009 pCi/sample	
1999 to present	ORNL	Pu-238, Pu-239, and Pu-240	0.009 pCi/sample	
1999 to present	ORNL	Am-241	0.009 pCi/sample	

a. Sources: PACE et al. (2000, Table 4.3); PGDP (2003b); BJC (1999, Table 3.3); PGDP (1999, Section 6.1.8).
b. MDC = minimum detectable concentration.

In some cases, a detection level for a particular radionuclide or analysis method is not available. In that case, dose reconstructors should use the nominal detection levels in Table 5-8. In addition, if it is not clear from the monitoring records how/where a particular claimant's sample was analyzed, it should be assumed that they were analyzed in-house (i.e., at the PGDP) and the MDC from that measurement type used to assess missed dose. Finally, if a record contains a notation of "less than X micrograms/L" or "< x pCi/sample", that value should reflect the MDC for that sample.

Table 5-8. Nominal (default) detection levels for urine bioassay.[a]

Radionuclide	Analytical method	Detection level (pCi/L)
Th-228	Radiochemical separation and alpha spectrometry	0.27
Th-232	Radiochemical separation and alpha spectrometry	0.27
U-234	Alpha spectrometry	0.27
U-235	Alpha spectrometry	0.27
U-238	Alpha spectrometry	0.27
Pu-238	Alpha spectrometry	0.27
Pu-239	Alpha spectrometry	0.27
Pu-240	Alpha spectrometry	0.27
Am-241	Alpha spectrometry	0.27

a. Source: ICRP (1988).

5.3.2 Reporting Formats and Codes

A variety of codes occurs on various urine bioassay records for PGDP. Table 5-9 lists a summary of those known as of the date of this report, along with their interpretations.

5.3.3 Instructions for Addressing Possible Interferences and Uncertainties

The practice of offsite collection of samples that takes place 24 to 48 hr after leaving the plant not only minimizes the possibility of sample cross-contamination, but it ensures that samples are collected after the transfer of the rapid clearance component. Some PGDP employees were asked to collect samples *after* 1 or 2 days off from work; if so, that collection instruction was sometimes noted on the analytical record.

Urine samples were typically collected in the workplace at PGDP, predominantly on Wednesdays.[4] Therefore, contamination of samples from the worker's hands or clothing cannot be ruled out as a contributor to any given result. If a second analysis was performed and if that result was negative, it is reasonable to assume the first result was a false positive due to sample contamination or laboratory error.

Dietary intakes of uranium pose a potential problem in interpreting urine bioassay results for PGDP workers. Because studies of the average daily uranium excretion on Paducah residents do not appear to have been performed, it is not possible to make corrections for the contribution of nonoccupational intakes of uranium to a given urine sample result. However, to put a given result into perspective, a nominal daily (24 hr) urinary excretion rate for uranium of 0.43 µg (environmental decision level at 95% confidence) can be used (BJC 1999). No correction for environmental levels of uranium is required for samples analyzed by fluorimetry or KPA because the MDC is larger than the correction.

[4] At some point, the sample collection day changed to Monday. However, the date of that procedural change is unclear.

Table 5-9. *In vitro* record codes.[a, b, c, d]

Form identifier	Measurement type	Column identifier	Code	Interpretation
WCP-455	Urine bioassay	Reason for Visit	33, 35	Industrial health recheck
WCP-455	Urine bioassay	Reason for Visit	OB, 39, 35-1	Recall sample requested following an elevated sample
WCP-455	Urine bioassay	Reason for Visit	MM recall	Monday morning recall sample; requested after days off work
WCP-455	Urine bioassay	Reason for Visit	32, 33, Term	Termination samples
WCP-455	Urine bioassay	Reason for Visit	30, 22	Rehire
WCP-455	Urine bioassay	Reason for Visit	37, 18, Per.	Periodic physical; confirmatory samples were collected during routine physicals; this typically did not pertain to those on a routine monitoring program.
WCP-455	Urine bioassay	Reason for Visit	26	Pre-employment
WCP-455	Urine bioassay	Reason for Visit	005, 07, 60, Exposure, Special, Release	Samples collected following an exposure or potential exposure in a uranium release or spill
WCP-455	Urine bioassay	Bottle No.		Permanent sample number
PGDP_HISTORICAL_URINE	Urine bioassay	Results		There is no distinction between positive results and detection limits.
PGDP_HISTORICAL_URINE	Urine bioassay	Sample Type	Physical	Routine physicals included the collection of a bioassay sample (random sampling program); this typically did not pertain to those on a routine monitoring program.
WCP-455	Urine bioassay	Reason for Visit	I.H.R. or IHR	Industrial health recheck (associated with routine physicals)
WCP-455	Urine bioassay	Top of Card	“A”	Refers to the shift worked (“A” = day shift)
WCP-455	Urine bioassay	Top of Card	“B”	Refers to the shift worked (“B” = evening shift)
WCP-455	Urine bioassay	Top of Card	“O”	Refers to the shift worked (“O” = midnight shift)
WCP-455	Urine bioassay	F, HG and OTHER	“B” (128)	Indicates the shift and the hours worked
PGDP_ANALIS_URINE	Urine bioassay	Results		Results are given in µg/L.
5EA HPINT - Bioassay Results Report	Urine bioassay	Units	4	dpm/L
5EA HPINT - Bioassay Results Report	Urine bioassay	Units	1	dpm/ml
5EA HPINT - Bioassay Results Report	Urine bioassay	Units	2	dpm/day
5EA HPINT - Bioassay Results Report	Urine bioassay	Units	3	dpm/sample
5EA HPINT - Bioassay Results Report	Urine bioassay	Units	5	µg/ml
5EA HPINT - Bioassay Results Report	Urine bioassay	Units	6	Bq/L
5EA HPINT - Bioassay Results Report	Urine bioassay	Units	7	Bq/day
5EA HPINT - Bioassay Results Report	Urine bioassay	Reason	3	Routine
5EA HPINT - Bioassay Results Report	Fecal analysis	Units	3	dpm/sample
5EA HPINT - Bioassay Results Report	Fecal analysis	Units	7	Bq/day
5EA HPINT - Bioassay Results Report	Breath analysis	Units	2	dpm/day
5EA HPINT - Bioassay Results Report	Breath analysis	Units	3	dpm/sample
5EA HPINT - Bioassay Results Report	Breath analysis	Units	4	dpm/L
5EA HPINT - Bioassay Results Report	Breath analysis	Units	6	Bq/L
5EA HPINT - Bioassay Results Report	Breath analysis	Units	7	Bq/day

Table 5-9 (Continued). *In vitro* record codes (Cont'd.).[a. b, c, d]

Form identifier	Measurement type	Column identifier	Code	Interpretation
WCP-885	Urine bioassay	Schedule		First digit refers to day of week; second digit(s) refer to type of analysis
WCP-885	Urine bioassay	Schedule	A	Day sample is to be taken (1=Monday; 2=Tuesday; 3=Wednesday; 4=Thursday)
WCP-885	Urine bioassay	Schedule	B	Type analysis (1=Uranium; 2=Fluoride; 3=Mercury)
WCP-885	Urine bioassay	Location	Shifts	A, B, O and D
			Frequency	Uranium #1 = 1 a month; Fluoride #2 = 1 a year; Mercury #3 = Blank
Permanent log (sample collection log)	Urine bioassay	Number	Permanent sample number	Numbers are consecutive from top to bottom of page and continue from one page to the next.
IBM Report Cards	Urine bioassay		A	Name
IBM Report Cards	Urine bioassay		B	Badge
IBM Report Cards	Urine bioassay		C	Date (date shown on log)
IBM Report Cards	Urine bioassay		D	Code for this service
IBM Report Cards	Urine bioassay		E	Analysis results in boxes labeled Uranium, Fluorides, and Mercury
UCN-5242, "Sample Analysis (Medical)."	Urine bioassay	Same as for NCP-455	Same as for NCP-455	This form number was in use from 1969 to 1970.

a. Sources: Maisler (2003); Eckerman and Ward (1992); PGDP (2003b),), Tomes (2004)

b. Form WCP-885 is referred to as "NCP-885" in Eckerman and Ward (1992), however they appear to refer to identify the same form.

c. Around June 1956 form WCP-455 was modified with additional columns added. Individual bioassay records reviewed indicates the exact date the newly expanded form was used varies a little from person to person, but was in the middle of 1956 for all records reviewed.

d. The bioassay records we have starting in 1977 (exact dates vary from person to person) are not copies of results recorded on bioassay cards--they are database printouts, e.g., "PGDP_Historical_Urine." The uranium results are in units of ug/L with results records to the nearest integer, i.e., 0, 1, 2, .etc.

5.4 *IN VIVO* MEASUREMENT METHODS

Whole-body counting and other *in vivo* methods were implemented beginning in the early 1950s. Until the present day (2003), this measurement method was used primarily in response to incidents, or for assessing the magnitude of insoluble material intakes. Even when routine whole-body counting was instituted for certain PGDP employees in the late 1960s, the counting frequency was sporadic and seldom greater than once per year.

5.4.1 Measurement Types and Detection Levels

At the PGDP, whole-body counting was performed using a mobile counter provided by the Y-12 plant (sometimes referred to as the MMES Counter) and at other facilities. Table 5-10 lists general information about the detection capabilities of this counting system for various periods.

Table 5-10. *In vivo* measurement types and detection levels for various periods.[a]

Period	Equipment	Measurement type	Radionuclide	MDA[b] (units of record)	Action level for recount	Action level for work restriction
1958	ORNL	Lung	Pu-239	0.04 µCi	Not specified	Not specified
1960-1967	Y-12	Whole body	Np-237	0.5 µCi	Not specified	Not specified
1968-1980	Y-12 mobile counter	Whole body	U-235	83 µg	Not specified	Not specified
1968-1980	Y-12 mobile counter	Whole body	U-238	4 mg	Not specified	Not specified
1968-1980	Y-12 mobile counter	Whole body	Np-237	0.017 µCi	Not specified	Not specified
1965-1991	Y-12 mobile counter	Lung	Total uranium	4 mg	4 mg	27 mg
1965-1991	Y-12 mobile counter	Lung	Enriched uranium (2%)	100 µg	100 µg	240 µg
1965-1991	Y-12 mobile counter	Lung	Depleted uranium	4 mg	4 mg	37 mg
1965-1991	Y-12 mobile counter	Lung	Np-237	0.2 nCi	1.7 nCi	17 nCi
1991-1995	Helgesson counter	Lung	Total uranium	2 – 4 mg	2 – 4 mg	27 mg
1991-1995	Helgesson counter	Lung	Enriched uranium	40 – 70 µg	40 – 70 µg	240 µg
> 1995	No counting performed.					

a. Sources: DOE (2000); Hill and Strom (1993); BJC (1999, Tables 3.2 and 3.5); Scott and West (1967).

b. MDA = minimum detectable activity or amount. The MDAs shown for uranium, while given in units of mass, are presumed to have been based on measurement of Th-234 activities along with an assumed isotopic ratio. It is presumed that the results for enriched uranium are based on measurement of the measurement of U-235 activities and an assumption of enrichment. However, these presumptions cannot be confirmed thus cautious use of these MDA values is recommended.

5.4.2 Reporting Formats and Codes

A variety of codes and reporting formats appear in the *in vivo* bioassay records. Table 5-11 lists the known codes with their interpretations.

5.4.3 Instructions for Addressing Possible Interferences and Uncertainties

For *in vivo* measurements, contamination could have occurred as external to the body or, in the case of chest counting, as external to the lung. If a follow-up *in vivo* count (the same day or within a few days) showed a dramatic decrease in activity or no detectable activity, then external contamination should be assumed.

Radon progeny and medical diagnostic or therapeutic procedures involving radionuclides have caused interferences among *in vivo* measurements, especially when sodium-iodide detectors were used. However, unless the count was invalidated or noted as being influenced by such interferences, the results should be used as recorded.

Table 5-11. *In vivo* record codes.[a]

Form identifier	Measurement type	Column identifier	Code	Interpretation
In Vivo Radiation Monitoring Report	Whole body count	Surface contamination	Check mark, yes	The surface contamination on the subject was checked. If measurable activity was found, it was so noted on the card.
In Vivo Radiation Monitoring Report	Whole body count	Analysis Sequence		These align with the Output - Analysis Sequence Results listed at the bottom of the card; they are not relevant to the dose reconstruction process.
In Vivo Radiation Monitoring Report	Whole body count	Output - Analysis Sequence Results	A. Enriched Uranium	The maximum U-235 enrichment was 2% until 2000, after which it was 5%.
In Vivo Radiation Monitoring Report	Whole body count	Output - Analysis Sequence Results	J. NLO Uranium	Refers to a special spectrum region of interest for National Lead of Ohio, an early operator of the Fernald facility.
In Vivo Radiation Monitoring Report	Whole body count	Hand-written notes	No Np noted	The presence or absence of Np-237 was qualitatively evaluated.
In Vivo Radiation Monitoring Report (with boxes for data entry)	Whole body count	Hand-written notes	BFD	Initials of whole body counter operator (individual who filled out the card)
5EA HPINT - Bioassay Results Report	*In vivo* records	Units	M	µCi
5EA HPINT - Bioassay Results Report	*In vivo* records	Units	N	nCi
5EA HPINT - Bioassay Results Report	*In vivo* records	Units	P	pCi
5EA HPINT - Bioassay Results Report	*In vivo* records	Units	D	dpm
5EA HPINT - Bioassay Results Report	*In vivo* records	Units	B	Bq
5EA HPINT - Bioassay Results Report	*In vivo* records	Units	U	µg

a. Sources: Maisler (2003); Eckerman and Ward (1992); PGDP (2003a).

Uncertainties in the bioassay measurements were not stated in the records. For results near or at the reporting levels, dose reconstructions should apply the prescribed standard deviation of 0.3 times the MDA or reporting level (NIOSH 2002).

On occasion, *in vivo* measurement results included ^{137}Cs. However, those PGDP workers could have had body burdens of ^{137}Cs from nonoccupational sources (e.g., fallout and consumption of specific foodstuffs). There is no evidence of occupational intakes of ^{137}Cs at the PGDP, thus no dose of record should be associated with these measurement result, if any.

5.4.4 Assessment of Intake for Monitored Employees

In general, available urine results should be considered the primary method of dose reconstruction. The *in vivo* measurements, especially in the earlier years of operation, were not used for routine monitoring purposes. However, those results can and should be used to verify assessments of dose based on urine bioassay results, in determining likely absorption types, or to provide upper and lower limits to the range of possible doses.

5.5 SIGNIFICANT INCIDENTS WITH INTERNAL DOSE POTENTIAL

During operations at PGDP, a number of incidents occurred that increased the potential for intakes of radioactive materials. If a claimant (or employee) recalls involvement in one or more of those incidents, the information in Table 5-12 can be used as input to an incident-specific dose assessment.

Table 5-12. Input parameters for significant incidents and events.[a]

Incident date	Incident description	Facility	Other information
1952–1990	Exposure to UF_4, UO, and process dust during guard patrolling	All buildings	
1952–1980	Exposure to uranium metal		
July 1953	First use of reactor tails		
November 1956	Fire	C-310	
1957–1977	Green salt, black oxides on floors and other surfaces	C-340	
December 1962	Fire	C-337	
March 1962	Explosion and fire	C-340	One fatality
April 1968	Worker overexposure	Unknown	Two workers overexposed
January 1978	Fire	C-315	
1958 to 1962	Cascade improvement program	C-331, C-333, C-335 and C-337	
1974 to 1982	Cascade improvement program	C-331, C-333, C-335 and C-337	
	Neptunium production	C-400	
1980 to 1982	Exposure to UF_4 and uranium dust during drum crushing	C-746	

a. Source: PACE et al. (2000).

REFERENCES

Cited References

ANSI (American National Standards Institute), 1995, *Bioassay Programs for Uranium*, HPS N13.22-1995, Washington, DC.

Baker, R. C., 1987, *Exposure Assessment - Uranium Recycle Materials in the Paducah Feed Plant*.

Berger, C. D., 2004, "Information from World Information Service on Energy website," memorandum to file, Oak Ridge Associated Universities, Oak Ridge, Tennessee.

Birchall, A., T. James, J. March, D. Dorrian, K. Davis, T. Smith, A. Phipps, T. Smith, N. Jarvis, T. Riddell, M. Oeoce, F. Smith, M. Puncher, and T. Ward, 2003, *Integrated Modules for Bioassay Analysis (IMBA) Expert*, OCAS Edition, Version 3.0.63, ACJ & Associates, Richland, Washington.

BJC (Bechtel Jacobs Company, LLC), 1999, *Internal Dosimetry Technical Basis Document for Bechtel Jacobs Company LLC*, BJC/OR-258, Oak Ridge National Laboratories, Oak Ridge, Tennessee.

DOE (U. S. Department of Energy), 2000, *Phase II - Independent Investigation of the Paducah Gaseous Diffusion Plant - Environment, Safety and Health Practices, 1952-1990*, Office of Oversight, Environment, Safety and Health, Washington, DC.

Eckerman K. F., and R. C. Ward, 1992, *DOSEXPRT - A Bioassay Dosimetry Code for Martin Marietta Energy Systems, Inc.*, ORNL/TM-11857, Oak Ridge National Laboratories, Oak Ridge, Tennessee.

Hill, R. L., and D. J. Strom, 1993, *Internal Dosimetry Technical Basis Manual for Portsmouth and Paducah Gaseous Diffusion Plants*, PNL-8723, Pacific Northwest Laboratories, Richland, Washington.

ICRP (International Commission on Radiological Protection), 1988, *Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation*, Publication 54, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1994, *Dose Coefficients for Intakes of Radionuclides by Workers*, Publication 68, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1997, *Individual Monitoring for Internal Exposure of Workers*, Publication 78, Pergamon Press, Oxford, England.

IT Corporation, 1992, *Personnel Exposure Potential to Transuranic Materials at the Paducah Gaseous Diffusion Plant*, IT/NS-91-109.

Maisler, J., 2003, "Paducah Gaseous Diffusion Plant discussion with K. Duncan, September 29-30, 2003", memorandum to file, November 20, 2003, Oak Ridge Associated Universities, Oak Ridge, Tennessee.

NCRP (National Council on Radiation Protection and Measurements), 1987, *Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition,* Report 87, Bethesda, Maryland.

NIOSH (National Institute of Occupational Safety and Health), 2002, *Internal Dose Reconstruction Implementation Guide,* OCAS-IG-002, Office of Compensation Analysis and Support, Cincinnati, Ohio.

ORAU (Oak Ridge Associated Universities), 2003, Technical Basis Document for the Hanford Site – Occupational Internal Dose, ORAUT-TKBS-0006-6, Revision 00, Oak Ridge, Tennessee.

PACE (Paper, Allied-Industrial, Chemical and Energy International Union), University of Utah, and U.S. Department of Energy, 2000, *Exposure Assessment Project at the Paducah Gaseous Diffusion Plant*, Utah.

PGDP (Paducah Gaseous Diffusion Plant), 1999, *PGDP Urinalysis Program for Detection of Intakes of Radionuclides*, Procedure CP2-HP-DS1030, Revision 2, Paducah, Kentucky.

PGDP (Paducah Gaseous Diffusion Plant), 2003a, "Urinalysis Program," undated and unnumbered document obtained by J. E. East in September 2003, Paducah, Kentucky.

PGDP (Paducah Gaseous Diffusion Plant), 2003b, "Uranium in Urine Analysis," undated and unnumbered document obtained by J. E. East in September 2003, Paducah, Kentucky.

Scott, L. M., and C. M. West, 1967,*Health Physics Application of In Vivo Gamma Spectrometry in a Uranium Processing Plant*, Y-KB-77, Union Carbide Corporation.

Smith, R. F., 1984, *Historical Impact of Reactor Tails on the Paducah Cascade*, KY/L-1239, Union Carbide Corporation.

Tomes, T., 2004, ORAU Team Dose Reconstruction Project for NIOSH Document Review Comment (ORAUT-TKBS-0019-5 Rev. 00-E), September 10, 2004.

Other References

ATSD (Agency for Toxic Substances and Disease Registry), 2001, *Public Health Assessment, Paducah Gaseous Diffusion (USDOE)*, Paducah, Kentucky.

BJC (Bechtel Jacobs Company, LLC), 2000, *Recycled Uranium Mass Balance Project Paducah Gaseous Diffusion Plant Site Report*, BJC/PGDP-167, Paducah Gasseous Diffusion Plant, Paducah, Kentucky.

DOE (U. S. Department of Energy), 1999, *Phase I - Independent Investigation of the Paducah Gaseous Diffusion Plant - Environment, Safety and Health Issues*, Office of Oversight; Environment, Safety and Health, Washington, DC.

DOE (U. S. Department of Energy), 2001, *A Preliminary Review of the Flow and Characteristics of Recycled Uranium Throughout the DOE Complex 1952 – 1999 Project Overview and Field Site Reports*, F-001-001.

ICRP (International Commission on Radiological Protection), 1959, *Report of Committee II on Permissible Dose for Internal Radiation*, Publication 2, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1974, *Reference Man*, Publication 23, Pergamon Press, Oxford, England.

NRC (U.S. Nuclear Regulatory Commission), 1974, *Applications of Bioassay for Uranium, 1974*, Regulatory Guide 8.11, Washington, DC.

PGDP (Paducah Gaseous Diffusion Plant), 1991, *PGDP Monthly Urinalysis Results, March 1989 to March 1991*, Paducah, Kentucky.

PGDP (Paducah Gaseous Diffusion Plant), 2003c, "Notice to Health Physics & Hygiene Department Personnel," undated and unnumbered, refers to "revised SPP-41", which is a 4-plant committee recommendation on uranium-in-urine procedures, Paducah, Kentucky.

Potter, C. A., 2002, "Intake Retention Fractions Developed from Models Used in the Determination of Dose Coefficients Developed for ICRP Publication 68 - Particulate Inhalation," *Health Physics*, volume 83, number 5, November.

Tockstein, D. J., 1996, *Technical Basis for Routine Bioassay Participation*, Lockheed Martin Energy Systems.

Travers, W. D., 1998, *Report to Congress on the Gaseous Diffusion Plants Located Near Paducah, Kentucky and Portsmouth, Ohio*, SECY-98-275, U. S. Nuclear Regulatory Commission, Washington, DC.

GLOSSARY

activity median aerodynamic diameter (AMAD)
The diameter of a unit density sphere with the same terminal settling velocity in air as that of the aerosol particle whose activity is the median for the entire aerosol.

acute
Pertaining to intakes received "acutely," i.e., within a short period.

bioassay
Measurement of amount or concentration of radioactive material either in the body or in biological material excreted or removed from the body. Another word for *radiobioassay*.

bioassay procedure
A procedure used to determine the kind, quantity, location, and retention of radionuclides in the body by direct (*in vivo*) measurements or by *in vitro* analysis of material excreted or removed from the body.

body burden
The quantity of radioactive material contained in the individual's body at a particular point in time.

chronic
Pertaining to low-level intakes received on a continuous basis.

dose
A general term for absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or total effective dose equivalent.

dose equivalent (H)
The product of absorbed dose (D) in rad (or gray) in tissue, a quality factor (Q), and other modifying factors (N). Dose equivalent is expressed in units of rem (or sievert) (1 rem = 0.01 sievert).

exposure
The general condition of being subjected to ionizing radiation, such as by exposure to ionizing radiation from external sources or to ionizing radiation sources inside the body. In this document, exposure does not refer to the radiological physics concept of charge liberated per unit mass of air.

insoluble material
A term loosely used to describe the relative degree of solubility of a material in body fluids. Recognizing that no material is absolutely insoluble, the terms low solubility or poorly soluble are preferable.

intake
The amount of radionuclide taken into the body by inhalation, absorption through intact skin, injection, ingestion, or through wounds. Depending on the radionuclide involved, intakes may be reported in units of mass, activity, or potential alpha energy.

internal dose or exposure
The dose equivalent received from radioactive material taken into the body (i.e., internal sources).

internal dose assessment
An assessment of the intake and associated internal radiation dose to workers based on measurements taken in the work environment or from individual bioassay measurements.

***In vitro* measurement**
Measurements to determine the presence of or to estimate the amount of radioactive material in the excreta or in other biological materials removed from the body.

***In vivo* measurement**
The measurement of radioactive material in the human body utilizing instrumentation that detects radiation emitted from the radioactive material in the body.

lung solubility type (F, M, or S)
A classification scheme for inhaled material according to its rate of clearance from the pulmonary region of the lung.

minimum detectable amount (MDA)
The smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability□of non-detection (Type II error) while accepting a probability of erroneously deciding that a positive (non-zero) quantity of analyte is present in an appropriate blank sample (Type I error).

minimum detectable concentration (MDC)
The minimum detectable amount, MDA, expressed in units of concentration.

monitoring (personnel)
The measurement of radioactivity in the whole body, in a region of the body, in material eliminated from the body or in the air for reasons related to the estimation of intake of radioactive material. The term *monitoring* includes interpretation of the measurements.

occupational dose
An individual's ionizing radiation dose (external and internal) resulting from that individual's work assignment. Occupational dose does not include doses received as a medical patient or doses resulting from background radiation or participation as a subject in medical research programs.

radiation
Ionizing radiation: alpha particles, beta particles, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, and other particles capable of producing ions. Radiation, as used in this part, does not include nonionizing radiation, such as radio- or microwaves, or visible, infrared, or ultraviolet light.

Reactor Tails
Recycled uranium (typically UO3) from reactor operations (typically Savannah River and Hanford) that contains traces of transuranics that was not removed during prior chemical processing (i.e., REDOX and PUREX processes).

recording level

A value below which data or results were considered to be too low to record and thus may not have been maintained.

rem

A special unit for dose equivalent. One rem is equal to 0.01 sievert.

routine monitoring

Monitoring carried out at regular intervals during normal operations.

sievert

The special name for the International System unit of dose equivalent. One sievert equals 1 joule per kilogram, which equals 100 rem.

special monitoring

Monitoring carried out in actual or suspected abnormal conditions (i.e., measurements performed to estimate the amount of radionuclide deposited in a person when an intake is known or is suspected to have occurred).

spot sample

A single void of urine.