DOE Review Release 08/31/2012

Document Title:	Document Number:	ORAUT-TKBS-0019-5
Paducah Gaseous Diffusion Plant – Occupational Internal Dose	Revision:	03
	Effective Date:	08/24/2012
	Type of Document:	TBD
	Supersedes:	Revision 02, ORAUT-OTIB-0037 Revision 00

Subject Expert(s): Daniel S. Mantooth, Clark B. Barton, and Tracy A. Ikenberry

Approval:	Signature on File Jodie L. Phillips, Document Owner	Approval Date:	08/21/2012
Concurrence:	Signature on File John M. Byrne, Objective 1 Manager	Concurrence Date:	08/21/2012
Concurrence:	Signature on File Edward F. Maher, Objective 3 Manager	Concurrence Date:	08/21/2012
Concurrence:	Vickie S. Short Signature on File for Kate Kimpan, Project Director	Concurrence Date:	08/21/2012
Approval:	Signature on File James W. Neton, Associate Director for Science	Approval Date:	08/24/2012

☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

FOR DOCUMENTS MARKED AS A TOTAL REWRITE, REVISION, OR PAGE CHANGE, REPLACE THE PRIOR REVISION AND DISCARD / DESTROY ALL COPIES OF THE PRIOR REVISION.

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
09/30/2004	00	New technical basis document for the Paducah Gaseous Diffusion Plant – Occupational Internal Dose. First approved issue. Initiated by Jay J. Maisler.
10/25/2006	01	Approved issue of Revision 01. This revision addresses the comment from the meeting with the United Steelworkers Local 5-550 and SPFPA Local 111 held on 2/10/2005. Constitutes a total rewrite of document. Incorporates recent direction from NIOSH to include DOL review comments on page 5. Revisions address corrective actions responding to OCAS assessment and internal review comments from Task 5. Changes were made at the end of Section 5.2. The original Table 5-3 has been deleted and replaced with an expansion of the original Table 5-4, which becomes Table 5-3. Table 5-2 has been updated to reflect new Table numbers. Incorporates internal and NIOSH formal review comments. This revision results in an increase in assigned dose and a PER is required. Training required: As determined by the Task Manager. Initiated by Daniel S. Mantooth. Approval: Signature on File 10/12/2006 Daniel S. Mantooth, Document Owner Signature on File 10/12/2006 John M. Byrne, Document Owner Signature on File 10/12/2006 Edward F. Maher, Task 5 Manager Signature on File 10/23/2006 Kate Kimpan, Project Director Brant A. Ulsh Signature on File for 10/25/2006 James W. Neton, Associate Director for Science
01/11/2007	01 PC-1	Approved page change revision initiated to correct the title of Table 5-2 on page 10. Modified Table 5-4 on page 13 to consolidate the years of 1952-1998 onto the first line of the table. Added Table 5-2a to the List of Tables. Incorporates formal internal and NIOSH review comments. Includes changes to Table 5-2a (page 11). This revision results in no change to the assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Daniel S. Mantooth. Approval: Signature on File 01/11/2007 Daniel S. Mantooth, Document Owner Signature on File 01/12/2007 John M. Byrne, Document Owner Signature on File 01/11/2007 Edward F. Maher, Task 5 Manager

		Signature on File 01/24/2007 Kate Kimpan, Project Director Brant A. Ulsh Signature on File for 01/31/2007 James W. Neton, Associate Director for Science
04/04/2007	02	Approved Revision 02 initiated to incorporate Attributions and Annotations section and improved reference citations in the text. No further changes occurred as a result of formal internal review. This revision results in no change to the assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Daniel S. Mantooth.
08/24/2012	03	Added SEC language. Added dates to Table 5-1 for C-720 Maintenance. Clarified column titles in Tables 5-2 and 5-2a. Updated references. Deleted previous Table 5-3 (not used in dose reconstruction). Revised Table numbers. Revised *in vitro* measurement in Table 5-3 for 1952 – 1988 period from 5 µg/L to 0.005 mg/L. Moved reporting formats and codes to Attachment A. Added Attachment B, Internal Dosimetry Coworker Data for PGDP (formerly ORAUT-OTIB-0037 Rev 00). Added the option for dose reconstructors to use the 95th-percentile intake rate in Attachment B. Added Attachment C to provide dose reconstructors a general idea of the types of job categories, work activities, and work locations for which internal dose assignment might be necessary. Added Section 5.4.5, Assessment of Intake for Unmonitored Employees. Moved Table 5-8, A History of Significant Events, to Attachment C. Incorporates formal NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Jodie L. Phillips.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

ACRONYMS AND ABBREVIATIONS

AMAD	activity median aerodynamic diameter
Bq	becquerel
CFR	Code of Federal Regulations
CIP	Cascade Improvement Program
cm	centimeter
CUP	Cascade Upgrade Program
d	day
DOE	U.S. Department of Energy
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
g	gram
GSD	geometric standard deviations
hr	hour
IMBA	Integrated Modules for Bioassay Analysis
IREP	Interactive RadioEpidemiological Program
KPA	kinetic phosphorescence analysis
L	liter
m	meter
MDA	minimum detectable amount
MDC	minimum detectable concentration
mg	milligram
mL	milliliter
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
OCAS	Office of Compensation Analysis and Support
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
pCi	picocurie
PGDP	Paducah Gaseous Diffusion Plant
POC	probability of causation
ppb	parts per billion
RU	recycled uranium
SEC	Special Exposure Cohort
TBD	technical basis document
TRU	transuranic

U.S.C.	United States Code
UF_6	uranium hexafluoride
UO_3	uranium trioxide
µCi	microcurie
µg	microgram
µm	micrometer
§	section or sections

5.1 INTRODUCTION

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historic background information and guidance to assist in 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 NIOSH staff 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 [DOE] facility" as defined in the Energy Employees Occupational Illness Compensation Program Act [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA defines a DOE facility as "any building, structure, or premise, including the grounds upon which such building, structure, or premise is located … in which operations are, or have been, conducted by, or on behalf of, the Department of Energy (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program)" [42 U.S.C. § 7384l(12)]. Accordingly, except for the exclusion for the Naval Nuclear Propulsion Program noted above, any facility that performs or performed DOE operations of any nature whatsoever is a DOE facility encompassed by EEOICPA.

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled "Exposure in the Performance of Duty." That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer "shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation[1]] guidelines established under subsection (c) …" [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation, 42 C.F.R. Pt. 82) restrict the "performance of duty" referred to in 42 U S. C. § 7384n(b) to nuclear weapons work (NIOSH 2010).

The statute also includes a definition of a DOE facility that excludes "buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program" [42 U.S.C. § 7384l(12)]. While this definition excludes Naval Nuclear Propulsion Facilities from being covered under the Act, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled "Exposure in the Performance of Duty"] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally-derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external occupational radiation exposures are considered valid for inclusion in a dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived (NIOSH 2010):

- Background radiation, including radiation from naturally occurring radon present in conventional structures
- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

[1] The U.S. Department of Labor (DOL) is ultimately responsible under the EEOICPA for determining the POC.

5.1.1 Purpose

This technical basis document (TBD) provides technical data and other key information that will serve as the technical basis for evaluating internal occupational radiation dose for EEOICPA claimants who were employed at the Paducah Gaseous Diffusion Plant (PGDP).

5.1.2 Scope

This document discusses the radionuclides potentially encountered by PGDP employees during the Plant's operational history. The PGDP mission was to enrich uranium in the form of uranium hexafluoride (UF_6) from roughly 0.7% ^{235}U (natural enrichment) to around 3% ^{235}U for use in domestic and foreign power reactors (DOE 2000, p. 8). Enrichment operations began in 1952 in the first four process buildings, C-331, C-333, C-310, and C-315. From 1953 until 1977, UF_6 feed material was produced from uranium trioxide (UO_3) at the Plant. From 1953 to 1964, and from 1968 to 1977, UF_6 was produced from the recycled uranium (RU) produced from spent reactor fuel. In May 1977, the feed plants ceased operation and all feed to the enrichment process was in the form of UF_6 obtained from outside sources. Other chemical compounds of uranium were present throughout the Plant's history including UO_2F_2, UF_4, and UO_3.

ORAUT (2012a) contains detailed information on the history of PGDP and the feed conversion and enrichment process.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 5.6.

PGDP is one of the original sites that was designated by Congress as part of the Special Exposure Cohort (SEC) under EEOICPA [42 U.S.C. § 7384l(14)]. This designation is as follows:

> *(A) The employee was so employed for a number of work days aggregating at least 250 work days before February 1, 1992, at a gaseous diffusion plant located in Paducah, Kentucky, Portsmouth, Ohio, or Oak Ridge, Tennessee, and, during such employment—*
>
> > *(i) was monitored through the use of dosimetry badges for exposure at the plant of the external parts of employee's body to radiation; or*
> >
> > *(ii) worked in a job that had exposures comparable to a job that is or was monitored through the use of dosimetry badges.*

Dose reconstruction guidance in this document is presented to provide a technical basis for dose reconstructions for nonpresumptive cancers that are not covered in the SEC class through January 31, 1992. Dose reconstructions for individuals employed at PGDP before February 1, 1992, but who do not qualify for inclusion in the SEC, can be performed using this guidance as appropriate.

5.2 SOURCE TERM

The radionuclides potentially encountered by PGDP employees consist primarily of the isotopes of uranium, ^{238}U, ^{234}U, and ^{235}U. The progeny of dosimetric interest for these radionuclides include ^{230}Th and ^{234m}Pa (BJC 1999, p. 8). While a range of ^{235}U enrichment values (from natural to 3%) was encountered in feed and product material, the dose reconstructor should assume a nominal enrichment value of 2% ^{235}U for all feed or product materials [1].

The presence of transuranic (TRU) and fission product isotopes has been known since the early days of operation (ORAUT 2012a). These radionuclides were present in small amounts in the feed material produced from RU, and include ^{237}Np, ^{239}Pu (PACE and University of Utah 2000), and ^{241}Am (Hightower et al. 2000). Technetium-99 is the fission product of concern from a dosimetry standpoint. During the conversion of the feed material from UO_3 to UF_6, the elemental species of the TRU and fission products react differently from a chemical standpoint. For example, ^{99}Tc tends to form a very volatile fluoride; readily introduced into the enrichment cascade, it essentially follows the UF_6 to the surge and/or product station. On the other hand, both ^{241}Am and ^{239}Pu form essentially nonvolatile fluorides. Most of these isotopes will remain in the feed conversion byproducts (i.e., ash) or in the feed cylinder heels, a primary source of TRU exposure from this process (Hightower et al. 2000). Some ^{241}Am will appear in the cascade components as a result of the decay of ^{241}Pu. While ^{237}Np forms a volatile fluoride, it rapidly oxidizes and plates on the metal cylinder walls (PACE and University of Utah 2000, p. 26), so the main hazard from this isotope is related to cylinder cleaning operations and cascade component maintenance. PGDP conducted operations to recover ^{99}Tc (April 1960 through June 1963) and ^{237}Np (November 1958 through March 1962), which presented an increased potential to exposure from these isotopes (PACE and University of Utah 2000).

Table 5-1 lists information on the isotopic species associated with PGDP operations and facilities and their dates of operation (when available).

Table 5-1. Radiological source term for PGDP processes and facilities.

Building	Process	Dates[a]	Radionuclides of concern[b]	Absorption type[c]
C-360	Product shipping and transfer		U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242 Tc-99	F M M, S F, M
C-400	Converter maintenance Converter salvage CIP/CUP Phase 1 CIP/CUP Phase 2	4/53–present 9/54–6/61 3/73–9/81	U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242 Tc-99	F M M, S F, M
C-400/ C-710	Np-237 recovery Tc-99 recovery	11/58–3/62 4/60–6/63	U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242 Tc-99	F M M, S F, M
C-409	Converter refurbish	1973–1981	U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242 Tc-99	F M M, S F, M
C-410	Green salt production UF_6 production Tc-99/Np-237 recovery (MgF_2)	1953–1956 1953–present 9/61–6/63	U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242 Tc-99 Th-230 Am-241	F, M, S M M, S F, M S M
C-420	Green salt production	1956–1977	U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242 Tc-99 Th-230 Am-241	F, M, S M M, S F, M S M
C-331	Cascade ops/maintenance	9/52–present	U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242	F M M, S
C-333	Cascade ops/maintenance	9/52–present		
C-335	Cascade ops/maintenance	4/54–present		
C-337	Cascade ops/maintenance	7/54–present	Tc-99	F, M

Building	Process	Dates[a]	Radionuclides of concern[b]	Absorption type[c]
C-310	Surge and product Tc-99/Np-237 recovery (MgF_2)	9/52–present 1/63–6/63	U-234, 235, 236, 238 Np-237 Pu-238, -239, -240, -242 Tc-99 Th-230	F M M, S F, M S
C-315	Surge and waste	9/52–present	U-234, -235, -236, -238 Th-230	F S
C-340	Metal production	12/57–12/62 1/68–10/77	U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242 Th-230	M, S M M, S S
C-720	Maintenance	1954-1961 1973-1981	U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242 Tc-99 Th-230 Am-241	F, M, S M M, S F, M S M
C-746	Waste metal smelting		U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242 Th-230	M, S M M, S S
C-749	Uranium metal burial		U-234, -235, -236, -238 Np-237 Pu-238, -239, -240, -242 Th-230	M, S M M, S S

a. All dates are from DOE (2000) except those for C-409 and the Cascade Ops/Maintenance buildings, which are from PACE and University of Utah (2000) and ORAUT (2012a), respectively.
b. From Hill and Strom (1993, Table 16.2).
c. From BJC (1999, Table 1.7).

5.2.1 Isotopic Concentrations

At PGDP, monitoring for intakes of uranium, *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, the internal dose assessment process requires the use of isotopic concentrations as input. As stated above, the chemical properties of the TRU and fission product contaminants resulted in changes in their relative concentrations in different parts of the process when compared to the original feed material. This section provides bounding estimates for radionuclide concentrations. The concentrations relative to the mass of total uranium and the activity of total uranium are listed in Tables 5-2 and 5-2a, respectively. A summary discussion on the information in the tables and its derivation is provided below. Details of unit conversion and data sources can be found in Barton (2006).

Pre-1983 Operations

Column 1, Pulverizer Operations, Ash Handling Activities, Green Salt Production, C-410 Production activities; Column 2, Converter Maintenance and Salvage, C-400 Cylinder Operations

The values in Columns 1 and 2 are from PACE and University of Utah (2000, Table 7.9). The maximum air concentration data were used to provide bounding values for TRU concentrations. The data in Table 7.9 is in units of µCi/cm^3, which were converted to the desired units of pCi (TRU)/g (U) or pCi (TRU)/pCi (U). Details of this process can be found in Barton (2006).

Table 5-2. Bounding isotopic concentrations for PGDP operations (pCi/g U).[a]

Radionuclide	1 Pulverizer operations, ash handling activities, green salt production, C-410 production activities pre-1983[b]	2 Converter maintenance and salvage, C-400 cylinder operations pre-1983[b]	3 Converter maintenance and salvage, C-400 cylinder operations post-1983[c]	4 Tc/Np recovery operations, C-400[d]	5 Balance of plant pre-1983[d]	6 Balance of plant post-1983[c]
Np-237	3.55E+04	1.61E+06	3.81E+04	1.76E+07	1.67E+03	3.67E+00
Pu-239/240	9.00E+05	3.24E+04	7.09E+04	4.42E+06	4.11E+01	1.95E+00
Pu-238	1.94E+05	7.01E+03	1.53E+04	9.54E+05	8.89E+00	4.21E-01
Pu-242	4.46E+01	1.61E+00	3.52E+00	2.19E+02	2.04E-03	9.67E-05
Pu-241	3.51E+07	1.27E+06	2.76E+06	1.72E+08	1.61E+03	3.38E-01
Am-241	1.56E+05	5.62E+03	1.85E+03	7.65E+05	7.13E+00	4.21E+00
Th-230	1.18E+06	1.67E+05	3.64E+05	2.27E+07	2.11E+02	1.00E+01
Tc-99	1.20E+05	1.20E+05	1.20E+05	3.81E+07	1.20E+05	1.20E+05
U-234	6.81E+05	6.81E+05	6.81E+05	6.81E+05	6.81E+05	6.81E+05
U-235	4.27E+04	4.27E+04	4.27E+04	4.27E+04	4.27E+04	4.27E+04
U-238	3.27E+05	3.27E+05	3.27E+05	3.27E+05	3.27E+05	3.27E+05
U-236	9.00E+02	9.00E+02	9.00E+02	9.00E+02	9.00E+02	9.00E+02

a. Based on uranium isotopic distribution for 2% enriched U (BJC 1999, Table 1-5). Assuming enriched rather than natural or depleted uranium provides bounding values for intake assessments based on urine samples, as well as TRU intake estimates.
b. PACE and University of Utah (2000).
c. Hightower et al. (2000).
d. BJC (2000).

Table 5-2a. Bounding isotopic activity concentrations for PGDP operations (pCi/pCi U). [a]

Radionuclide	1 Pulverizer operations, ash handling activities, green salt production, C-410 production activities pre-1983[b]	2 Converter maintenance and salvage, C-400 cylinder operations pre-1983[b]	3 Converter maintenance and salvage, C-400 cylinder operations post-1983[c]	4 Tc/Np recovery operations, C-400[d]	5 Balance of plant pre-1983[d]	6 Balance of plant post-1983[c]
Np-237	3.38E-02	1.53E+00	3.63E-02	1.67E+01	1.59E-03	3.49E-06
Pu-239/240	8.56E-01	3.08E-02	6.75E-02	4.21E+00	3.91E-05	1.86E-06
Pu-238	1.85E-01	6.67E-03	1.46E-02	9.08E-01	8.46E-06	4.01E-07
Pu-242	4.24E-05	1.53E-06	3.35E-06	2.08E-04	1.94E-09	9.20E-11
Pu-241	3.34E+01	1.21E+00	2.63E+00	1.64E+02	1.53E-03	3.22E-07
Am-241	1.48E-01	5.35E-03	1.76E-03	7.28E-01	6.79E-06	4.01E-06
Th-230	1.12E+00	1.59E-01	3.46E-01	2.16E+01	2.01E-04	9.52E-06
Tc-99	1.14E-01	1.14E-01	1.14E-01	3.63E+01	1.14E-01	1.14E-01
U-234a	4.89E-01	4.89E-01	4.89E-01	4.89E-01	4.89E-01	4.89E-01
U-235a	2.17E-02	2.17E-02	2.17E-02	2.17E-02	2.17E-02	2.17E-02
U-238a	4.89E-01	4.89E-01	4.89E-01	4.89E-01	4.89E-01	4.89E-01
U-236a	6.56E-04	6.56E-04	6.56E-04	6.56E-04	6.56E-04	6.56E-04

a. Based on uranium isotopic distribution for natural U (BJC1999, Table 1-5). Assuming natural rather than enriched uranium provides bounding values for uranium intake estimates based on in vivo counting methods.
b. PACE and University of Utah (2000).
c. Hightower et al. (2000).
d. BJC (2000).

The maximum plutonium activity in air is reported in PACE and the University of Utah (2000) as being comprised of ^{239}Pu. However, it is more consistent with all references [BJC (2000), Hightower et al. (2000)] to assume that the faction of the alpha activity attributed to ^{239}Pu be considered as total alpha-emitting isotopes plus ^{241}Am. A nominal isotopic distribution for TRU in fuel-grade plutonium in Mansfield (1997) was used to calculate the ratios of the various isotopes from the value provided in the reference.

Column 1 values are applicable to pulverizer operations, ash handling activities, green salt production, and feed material production operations in Buildings C-410 and C-420. Column 2 values are applicable to converter maintenance/salvage and cylinder cleaning operations in Building C-400.

Column 4, Tc/Np Recovery Operations; Column 5, Balance of Plant
The values in Columns 4 and 5 are from BJC (2000, Table 2.4.1). This reference presents a detailed analysis of the mass balance of ^{239}Pu and ^{237}Np using individual activity analysis. Table 2.4.1 presents the maximum concentrations (in parts per billion) of the ^{239}Pu and ^{237}Np for the processes encountered at PGDP. The remaining plutonium isotopes and ^{241}Am concentrations are determined using the isotopic ratios for fuel-grade plutonium in Mansfield (1997) and as described above.

The values in Column 4 are applicable to all technetium/neptunium recovery operations in Building C-400, including those that might have taken place after 1983.

The values for Column 5 consist of the upper 95% bound of the average Table 2.4.1 values (minus those areas/processes already accounted for in Columns 1 and 2). They are applicable to pre-1983 activities, which include Cascade Operations (Buildings C-333, C-337, C-410), Production/Handling UF_4 (Building C-340), Connecting/Disconnecting UF_6 Tails Cylinders (Buildings C-315, C-340), Connecting/Disconnecting Product Cylinders (Building C-310), Changing/Cleaning MgF_2 Traps (Buildings C-410, C-310), and U Metal Production (Building C-340) (BJC 2000, Table 2.4.1).

Post-1983 Operations
Column 3, Converter Maintenance and Salvage; Column 6, Balance of Plant
The values in Columns 3 and 6 are based on Hightower et al. (2000). These values were included in this TBD to account for the fact that an insignificantly small percentage (0.008% of the total) of RU feed was introduced to the cascade after 1977 (BJC 2000, Appendix C). In addition, a two-phase (1954 to 1961 and 1973 to 1981) upgrade program effectively replaced the major components of the cascade [2]. These facts would imply that, over time, TRU and fission product contaminants would be greatly reduced from the process, with the exception of those entrained in the feed and tail cylinder heels. Only cylinder maintenance/washing operations would have a potential for exposure to these isotopes. Hightower et al. (2000) contains analytical data in parts per billion for ^{237}Np, ^{239}Pu, and ^{241}Am in depleted uranium cylinders and feed material. The values in Table 3, the bounding concentrations for UF_6 feed, were used to represent Converter Salvage Operations (Column 3). As for the pre-1983 operations, these include converter maintenance/salvage and cylinder cleaning in Building C-400. The values for the remaining plutonium isotopes were determined as using the ratios in Mansfield (1997), as described above.

For Balance of Plant Operations (Column 6) ^{239}Pu, ^{237}Np, and ^{241}Am values specified in Hightower et al. (2000, Table 2) were used. These data represent the concentrations of dispersed contamination in DUF_6 cylinders and, considering the negligible quantity of RU introduced to the Plant during this period, is believed to bound the actual ^{239}Pu and ^{237}Np levels because these isotopes were primarily concentrated in the pulverizer ash or plated in feed cylinder walls.

Column 6 is applicable to post-1983 activities, which include Cascade Operations (Buildings C-333, C-337, C-410), Production/Handling UF_4 (Building C-340), Connecting/Disconnecting UF_6 Tails Cylinders (Buildings C-315, C-340), Connecting/Disconnecting Product Cylinders (Building C-310),

Changing/Cleaning MgF_2 Traps (Buildings C-410, C-310), and U Metal Production (Building C-340) (BJC 2000, Table 2.4.1).

Thorium 230
Pace and University of Utah (2000) lists ^{230}Th, a decay product of uranium, as a potential source of concern and provides estimates for the activity concentration in Buildings C-400 and C-410 (Columns 1 and 2 of Table 5-2 and 5-2a). BJC (2000) and Hightower et al. (2000) offer no data on ^{230}Th. The ratio of ^{237}Np to ^{239}Pu data from each (BJC 2000, Table 2.4.1; Hightower et al. 2000) was compared to the same ratio determined for the PACE and University of Utah "mixes". The mix for which the two ratios compared most closely was used to determine the ^{230}Th concentration. A detailed discussion of this technique is in Barton (2006).

Technetium 99
The ^{99}Tc concentration for all operations except ^{99}Tc recovery is assumed to remain constant throughout the process and is based on the reported 7-ppb maximum for the feed material (Smith 1984). The ^{99}Tc concentration for recovery operations is from BJC (2000).

Note: If the work location of a claimant is unknown, use Column 2 or 3 in Table 5-2 (or Table 5-2a), as appropriate to the times of employment [3].

When determining potential intakes of TRU/^{99}Tc by applying the isotopic ratios in Table 5-2 to urinalysis data, dose reconstructors should consider that PACE and University of Utah (2000) noted instances in which the bioassay data in the electronic database are not consistent with original documents. Verify urinalysis results obtained from the electronic database against the written records.

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.[2] 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 Oak Ridge National Laboratory (ORNL) during early periods and later at analytical services contractor locations.

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 (PACE and University of Utah 2000, p. 40).

In later years, total uranium content was assessed by kinetic phosphorescence analysis (KPA). Neither methodology included isotopic determinations (BJC 1999).

5.3.1 Measurement Types and Detection Levels

Table 5-3 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 and University of Utah 2000, Section 4.2.1). If detection levels for specific methods were not found in the available references, levels specified as typical in ICRP (1989) were used [4].

[2] Fecal sampling was occasionally performed for special studies. However, a program for routine or diagnostic monitoring of fecal samples was never implemented [4].

Table 5-3. *In vitro* measurement types and detection levels for various periods.[a]

Period	Measurement type	Radionuclide	MDC[a]	Comments
1952–1998	PGDP fluorimetry[b]	Total uranium	0.005 mg/L	
1985–1989	ORNL[d]	U-234, -235, and -238	0.03 pCi/sample	
1989–present	ORNL	U-234, -235, and -238	0.01 pCi/sample	
1989–present	Contractor[e]	U-234, -235, and -238	0.3 pCi/L	
1999–present	KPA[c]	Total uranium	0.06µg/sample	DR should use the cite MDA[a] for all urinalysis results by KPA, regardless of analysis date
1999–present	ORNL	Natural uranium	0.06 mg/sample	
1969–1985	PGDP gross beta[c]	Tc-99	10 dpm/mL	10 dpm/mL
1985–1989	ORNL	Isotopic plutonium	0.02 pCi/sample	
1989–present	ORNL	Isotopic plutonium	0.01 pCi/sample	
1985–1989	ORNL beta counting	Tc-99	18.1 pCi/sample	
1989–present	ORNL	Tc-99	90.9 pCi/sample	
1999–present	ORNL	Th-228, -230, and -232	0.014 pCi/sample	
1985–1989	ORNL	Np-237	0.04 pCi/sample	
1989–present	ORNL	Np-237	0.01 pCi/sample	
1985–1989	ORNL	Pu-238, -239, and -240	0.02 pCi/sample	
1989–present	ORNL	Pu-238, -239, and -240	0.01 pCi/sample	
1985–1989	ORNL	Am-241	0.02 pCi/sample	
1989–present	ORNL	Am-241	0.01 pCi/sample	

a. MDC = minimum detectable concentration; DR = dose reconstructor; MDA = minimum detectable amount.
b. PACE and University of Utah (2000, p. 40) documents use of fluorimetry through 1982. No definitive reference for the specific method used for total uranium could be found in available documentation from 1983 to 1998. It is favorable to the claimant to assume fluorimetry (rather than KPA) during this period.
c. BJC (1999, Table 3.9).
d. ORAUT (2007) [5].
e. ICRP (1989).

In addition, if it is not clear from the monitoring records how or where a particular claimant's sample was analyzed, assume that it was analyzed in-house (i.e., at PGDP) and use the minimum detectable concentration (MDC) from that measurement type 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 [6].

5.3.2 Reporting Formats and Codes

A variety of codes occur on urine bioassay records for PGDP. Attachment A summarizes those known at the date of this TBD, along with their interpretations.

5.4 *IN VIVO* MEASUREMENT METHODS

In vivo methods were implemented beginning in the early 1950s. These measurement methods were used primarily in response to incidents, or for assessing the magnitude of insoluble material intakes. Routine *in vivo* counting was instituted for certain PGDP employees in the late 1960s (PACE and University of Utah 2000).

5.4.1 Measurement Types and Detection Levels

At PGDP, *in vivo* 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 5-4 lists general information about the detection capabilities of this counting system for various periods.

Table 5-4. *In vivo* measurement types and detection levels for various periods.

Period	Equipment	Measurement type	Radionuclide	MDA[a] (units of record)	Action level for recount	Action level for work restriction
1965–1991	Y-12 mobile counter[b]	Chest	Enriched uranium (U-235)	0.1 mg	0.1 mg	0.24 mg
1965–1991	Y-12 mobile counter[b]	Chest	Depleted uranium (reported as "Total uranium")	4 mg	4 mg	37 mg
1965–1991	Y-12 mobile counter[c]	Chest	Np-237	200 pCi	1,700 pCi	17,000 pCi
1991–1995	Helgeson counter[c]	Chest	Total uranium	2 – 4 mg	2 – 4 mg	27 mg
1991–1995	Helgeson counter[c]	Chest	Enriched uranium	0.04 – 0.07 mg	0.04 – 0.07 mg	0.24 mg
>1995	No counting performed					

a. MDA = minimum detectable activity or amount.
b. ORAUT (2004a) [7].
c. Hill and Strom (1993).

5.4.2 Reporting Formats and Codes

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

5.4.3 Instructions for Addressing Possible Interferences and Uncertainties

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 PGDP, so no dose of record should be associated with these measurement results [8].

5.4.4 Assessment of Intake for Monitored Employees

In general, available urine results should be considered the primary method of dose reconstruction. *In vivo* measurements, especially in the earlier years of operation, were not used for routine monitoring purposes. However, those results might be useful for verifying assessments of dose based on urine bioassay results, in determining likely absorption types, or in providing upper and lower limits to the range of possible doses [9]. When converting urine sample results from units of mass to activity, an assumption of enriched uranium is more favorable to the claimant because of the high specific activity of U-234. Urine sample results reported in units of mass can be converted to units of activity using the information contained in Table 5-2, which gives the specific activity for total uranium as 1.05 pCi/µg. Intakes calculated from urine samples should be projected to the dates of any chest counts to determine if the chest counts results are overpredicted. Chest counts may be reported as "U-235" or "enriched uranium;" both are assumed to refer to the U-235 mass only. This is converted to U-235 activity by multiplying the result (or MDA) by the U-235 specific activity of 2.16 pCi/µg, then dividing by 0.041 pCi U-235/pCi U to get total U activity. Attachment C provides the dose reconstructor a general

idea of the types of job categories, work activities, and work locations for which internal dose assignment might be necessary.

When assessing intakes from chest count results, Table 5-2a is used because in this situation the assumption of natural uranium is more favorable to the claimant since only the U-235 is being measured. The chest count result is again converted to U-235 activity by multiplying the result (or MDA) by the U-235 specific activity of 2.16 pCi/µg. Total uranium activity is then calculated by dividing by 0.0217 pCi U-235/pCi U to get total U activity.

5.4.5 Assessment of Intake for Unmonitored Employees

There are instances of energy employees who, for a variety of reasons, were not monitored for internal exposure during the course of their employment at a DOE facility, or whose records of such monitoring are incomplete or unavailable. In such cases, dose reconstructors can use data from coworkers to approximate an individual's possible exposure. Attachment B provides the details of the calculation and assignment of intakes based on coworker data from the PGDP for estimating unmonitored exposures or where records of monitoring are incomplete or unavailable. Attachment C provides the dose reconstructor a general idea of the types of job categories, work activities, and work locations for which internal dose assignment might be necessary.

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, dose reconstructors can use the information in Attachment C to identify the dates, location, and/or source term for an incident-specific dose assessment.

5.6 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in the preceding text, bracketed callouts have been inserted to indicate information, conclusions, and recommendations to assist in the process of worker dose reconstruction. These callouts are listed in this section with information that identifies the source and justification for each item. Conventional references are provided in the next section that link data, quotations, and other information to documents available for review on the Oak Ridge Associated Universities (ORAU) Team servers.

[1] Barton, Clark B. ORAU Team. Sr. Health Physicist. May 2006.
The actual enrichment of uranium found in the process areas of PGDP is unknown because the facility produced product with a range of enrichments over the years. A nominal value of 2% was assumed because it is the approximate midpoint to the range, and provides a conservative (i.e., favorable to the claimant) result for the calculated values listed in Table 5-2 in comparison to natural uranium, which is assumed in the reference material (PACE and the University of Utah 2000; BJC 2000).

[2] Mantooth, Daniel S. ORAU Team. Sr. Health Physicist. May 2006.
The references PACE and the University of Utah (2000) and DOE (2000) include discussions stating that the Cascade Improvement Program (CIP)/Cascade Upgrade Program (CUP) processes that ended in 1981 involved a cell-by-cell removal and replacement of the compressors and converters, process piping, and support system components. The year 1983 is an arbitrary time assumed to allow additional time to finish converter salvage line work, refurbishment of parts, etc.

[3] Mantooth, Daniel S. ORAU Team. Sr. Health Physicist. May 2006.
Column 2 data, based on PACE and the University of Utah (2000) values for the period before 1983, and Column 3 data, based on Hightower et al. (2000) values for the period after 1983, present the higher values for activities at the site that are not specific to one job or area.

[4] Mantooth, Daniel S. ORAU Team. Sr. Health Physicist. May 2006.
The nominal MDA values in ICRP (1989) are greater than those typically achievable by a laboratory using industry-accepted methods and techniques. Missed dose estimates will be greater with the use of the ICRP (1989) MDA value.

[5] Mantooth, Daniel S. ORAU Team. Sr. Health Physicist. May 2006.
Values for *in vitro* MDAs were often not available in references specific to PGDP. Because it is known that urine samples were sent to ORNL for analysis during the periods listed in Table 5-3, it was assumed that the values reported in ORAUT (2007) would apply.

[6] Mantooth, Daniel S. ORAU Team. Sr. Health Physicist. May 2006.
The in-house MDC values are higher than offsite laboratory results, and any level indicated that is higher must be used. This is to ensure the use of the number most favorable to the claimant.

[7] Mantooth, Daniel S. ORAU Team. Sr. Health Physicist. May 2006.
Values for *in vivo* MDAs were often not available in references specific to PGDP. Because it is known that the Y-12 Mobile Counter was used for both the Paducah and Portsmouth sites for analysis during the periods listed in Table 5-3, it was assumed that the values reported in ORAUT (2004a) would apply.

[8] Mantooth, Daniel S. ORAU Team. Sr. Health Physicist. May 2006.
Extensive document review found no indication that ^{137}Cs had ever been considered an isotope of concern at Paducah beyond the occasional elevated individual due to dietary consumption and the increases due to global testing/incidents.

[9] Mantooth, Daniel S. ORAU Team. Sr. Health Physicist. May 2006.
NIOSH (2002, Section 5.1) states: "Bioassay measurements are generally the most reliable data available for assessing internal exposures." In consideration of the limited *in vivo* data, those uses might be limited.

REFERENCES

Author unknown, no date, “Urinalysis Program,” from Paducah Gaseous Diffusion Plant files, Paducah, Kentucky. [SRDB Ref ID: 12651]

Barton, C. B., 2006, "Verifying and Expanding Paducah Occupational Internal Dose Document Tables," memorandum to D. S. Mantooth, Dade Moeller & Associates, Richland, Washington, May 2006. [SRDB Ref ID: 24017]

BJC (Bechtel Jacobs Company) 1999, *Internal Dosimetry Technical Basis Document for Bechtel Jacobs Company LLC, BJC/OR-258*. [SRDB Ref ID: 12678]

BJC (Bechtel Jacobs Company), 2000, *Recycled Uranium Mass Balance Project Paducah Gaseous Diffusion Plant Site Report*, Draft for Review, BJC/PGDP-167, Paducah Gaseous Diffusion Plant, Paducah, Kentucky, March 5. [SRDB Ref ID: 16498]

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, D.C., February. [SRDB Ref ID: 12040]

Eckerman, K. F., and R. C. Ward, 1992, *DOSEXPRT - A Bioassay Dosimetry Code for Martin Marietta Energy Systems, Inc.*, ORNL/TM-11857, Martin Marietta Energy Systems, Oak Ridge National Laboratory, Oak Ridge, Tennessee, April. [SRDB Ref ID: 12548]

Hightower, J. R., L. R. Dole, D. W. Lee, G. E. Michaels, M. I. Morris, D. G. O'Conner, S. J. Pawel, R. A. Schmoyer, L. D. Trowbridge, and V. S. White, 2000, *Strategy for Characterizing Transuranics and Technetium Contamination in Depleted UF_6 Cylinders*, ORNL/TM-2000/242, UT-Battelle, Oak Ridge National Laboratory, Oak Ridge, Tennessee, October. [SRDB Ref ID: 23571]

Hill, R. L., and D. J. Strom, 1993, *Internal Dosimetry Technical Basis Manual for Portsmouth and Paducah Gaseous Diffusion Plants*, PNL-8723, Battelle Memorial Institute, Pacific Northwest Laboratory, Richland, Washington, June. [SRDB Ref ID: 12024]

ICRP (International Commission on Radiological Protection), 1989, *Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation*, Publication 54, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1995, *Dose Coefficients for Intakes by Workers*, Publication 68, Pergamon Press, Oxford, England.

Maisler, J., 2003, “Results of Onsite Interview at Paducah," memorandum to C. Berger, Integrated Environmental Management, Dayton, Ohio, November 20. [SRDB Ref ID: 13764]

Mansfield, W.G., 1997, *Nuclear Emergency and Radiological Decision Handbook,* ARG-RAP27 FRMAC-NEST, Lawrence Livermore National Laboratory, Livermore, California, May. [SRDB Ref ID: 45243]

NIOSH (National Institute for Occupational Safety and Health), 2002, *Internal Dose Reconstruction Implementation Guideline*, OCAS-IG-002, Rev. 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, August.

NIOSH (National Institute for Occupational Safety and Health), 2010, *Radiation Exposures Covered for Dose Reconstructions Under Part B of the Energy Employees Occupational Illness Compensation Program Act*, DCAS-IG-003, Rev. 01, Division of Compensation Analysis and Support, Cincinnati, Ohio, October 5.

ORAUT (Oak Ridge Associated Universities Team), 2004a, *Technical Basis Document for Portsmouth Gaseous Diffusion Plant – Occupational Internal Dose*, ORAUT-TKBS-0015-5, Rev. 00, Oak Ridge, Tennessee, November 24.

ORAUT (Oak Ridge Associated Universities Team), 2005, *Technical Information Bulletin: Analysis of Coworker Bioassay Data for Internal Dose Assignment,* Rev. 01, ORAUT-OTIB-0019, Oak Ridge, Tennessee, October 7.

ORAUT (Oak Ridge Associated Universities Team), 2004b, *Technical Information Bulletin: Assignment of Environmental Internal Doses for Employees Not Exposed to Airborne Radionuclides in the Workplace,* Rev. 00, ORAUT-OTIB-0014, Oak Ridge, Tennessee, June 22.

ORAUT (Oak Ridge Associated Universities Team), 2006, *Y-12 National Security Complex – Internal Dose*, ORAUT-TKBS-0014-5, Rev. 02 PC-1, Oak Ridge, Tennessee, October 10.

ORAUT (Oak Ridge Associated Universities Team), 2007, *Oak Ridge National Laboratory – Occupational Internal Dose*, ORAUT-TKBS-0012-5, Rev. 01, Oak Ridge, Tennessee, October 1.

ORAUT (Oak Ridge Associated Universities Team), 2012a, *Paducah Gaseous Diffusion Plant – Site Description*, ORAUT-TKBS-0019-2, Rev. 03, Oak Ridge, Tennessee, August 22.

ORAUT (Oak Ridge Associated Universities Team), 2012b, *Paducah Gaseous Diffusion Plant – Occupational Internal Dose*, ORAUT-TKBS-0019-5, Rev. 03, Oak Ridge, Tennessee, August 24.

PACE (Paper, Allied Industrial, Chemical and Energy Workers International Union) and University of Utah, 2000, *Exposure Assessment Project at the Paducah Gaseous Diffusion Plant*, December. [SRDB Ref ID: 10870]

Smith, R. F., 1984, *Historical Impact of Reactor Tails on the Paducah Cascade*, KY/L-1239, Union Carbide Corporation, Paducah Gaseous Diffusion Plant, Paducah, Kentucky, March 19. [SRDB Ref ID: 16826]

GLOSSARY

absorption type

Categories for materials according to their speed of absorption in the body, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F for fast absorption (formerly inhalation class D), M for moderate absorption (formerly inhalation class W), and S for slow absorption (formerly inhalation class Y). Also called solubility type.

activity median aerodynamic diameter (AMAD)

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. In relation to health physics, normally assumed to be 5 micrometers.

bioassay

Determination of kinds, quantities, or concentrations, and in some cases locations of radioactive material in a living body, whether by direct measurement (*in vivo* measurement) or by analysis and evaluation of materials excreted or removed from the body (*in vitro* measurement).

body burden

Amount of radioactive material in an individual's body at a particular point in time.

dose

In general, the effects of ionizing radiation in terms of the specific amount of energy absorbed per unit of mass. Effective and equivalent doses are in units of rem or sievert; other types of dose are in units of roentgens, rads, reps, or grays. Various terms narrow the type of dose, and some are additive:

- Absorbed dose is the amount of energy deposited in a substance by ionizing radiation.
- Collective dose is the sum of the doses to a specific population.
- Committed dose is the dose over time (usually 50 years for workers) to a specific organ or tissue from an intake of radioactive material.
- Cumulative dose is the sum of all doses to the same portion of the body or to the whole body over time.
- Deep dose is the dose at a 1-centimeter depth in tissue (1,000 milligrams per square centimeter).
- Effective dose is the sum of the equivalent doses in the principal tissues and organs of the body, each weighted by a tissue weighting factor that accounts for the probabilities of fatal and nonfatal cancers according to severity and the average length of life lost due to an induced cancer. It indicates the biological effect of the radiation exposure in that tissue.
- Equivalent dose or dose equivalent is the absorbed dose in a tissue or organ multiplied by a weighting factor for the particular type of radiation.
- Organ dose is the dose to a specific organ.

- Penetrating dose is that from higher energy photon (gamma and X-ray) radiation and neutron radiation that penetrates the outer layers of the skin. Nonpenetrating dose is that from beta and lower energy photon radiation.

- Personal dose equivalent is the dose equivalent in soft tissue below a specified point on the body at a specified depth.

- Shallow dose is the dose at a 0.07-centimeter depth in tissue (7 milligrams per square centimeter).

- Skin dose is the dose to the skin.

- Whole-body dose is the dose to the entire body excluding the contents of the gastrointestinal tract, urinary bladder, and gall bladder.

dose equivalent (*H*, DE)
Product of absorbed dose in units of rem or sievert in tissue multiplied by a weighting factor and sometimes by other modifying factors to account for the potential for a biological effect from the absorbed dose. See *dose*.

exposure
(1) In general, the act of being exposed to ionizing radiation. (2) Measure of the ionization produced by X- and gamma-ray photons in air in units of roentgens.

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
Radioactive material taken into the body by inhalation, absorption through the skin, injection, ingestion, or through wounds. Depending on the radionuclide involved, intakes are in units of mass, activity, or potential alpha energy.

internal dose or exposure
Dose received from radioactive material in the body.

internal dose assessment
Estimation of an intake of radioactive material and the consequent internal radiation dose based on measurements in the work environment and/or bioassay.

in vitro
Of or relating to a process that takes place under artificial conditions or outside a living organism (e.g., in the laboratory). From Latin meaning *in glass*.

in vivo
Of or relating to a process that takes place in a living organism. From Latin meaning *in life*.

lung solubility type
See *absorption type*.

minimum detectable amount (MDA)

Lowest amount of radioactive activity or substance amount detectable by a specific instrument or process. Smallest amount or activity of a radionuclide in a sample or organ that yields a result above the detection level with a specific probability of a Type II (false negative) error while accepting an specific probability of a Type I (false positive) error.

minimum detectable concentration (MDC)

Lowest concentration of a material in a substance (e.g., urine) detectable by a specific instrument or process. Minimum detectable activity (or amount) in units of concentration.

minimum reporting level

Level below which an analytical dose is not recorded in the worker's dose record, usually based on a site-specific policy decision. The recording level is not necessarily the same as the minimum detectable amount or activity for that measurement. Also called less-than value, minimum reportable dose, minimum recordable or recording dose, recording level, and reporting level.

monitoring (personnel)

Periodic or continuous determination of the presence or amount of ionizing radiation or radioactive contamination in air, surface water, groundwater, soil, sediment, equipment surfaces, or personnel (for example, bioassay or alpha scans). In relation to personnel, monitoring includes internal and external dosimetry including interpretation of the measurements.

occupational dose

Internal and external ionizing radiation dose from exposure during employment. Occupational dose does not include that from background radiation or medical diagnostics, research, or treatment.

radiation

Subatomic particles and electromagnetic rays (photons) that travel from one point to another, some of which can pass through or partly through solid materials including the human body. Radiation, as used in this document, does not include nonionizing radiation, such as radio- or microwaves, or visible, infrared, or ultraviolet light.

reactor tails

Recycled uranium (typically UO_3) from reactor operations (typically Savannah River and Hanford) that contains traces of transuranic isotopes not removed during chemical processing (e.g., reduction-oxidation).

recording level

See *minimum reporting level*.

rem

Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The average American receives 360 millirem a year from background radiation. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

routine monitoring

Monitoring carried out at regular intervals during normal operations.

sievert (Sv)

International System unit for dose equivalent, which indicates the biological damage caused by radiation. The unit is the radiation value in gray (equal to 1 joule per kilogram) multiplied by a weighting factor for the type of radiation and a weighting factor for the tissue; 1 Sv equals 100 rem.

special monitoring

Monitoring in addition to the routine monitoring program carried out for special purposes such as estimating the amount of radionuclide deposited in a person after a known or suspected accidental intake or after a known or suspected environmental release.

spot sample

In relation to bioassay, usually a single void of urine.

ATTACHMENT A
CODES USED IN INTERNAL DOSE RECORDS
Page 1 of 4

LIST OF TABLES

Table A-1. *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 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 routine monitoring program.
WCP-455	Urine bioassay	Reason for Visit	26	Preemployment
WCP-455	Urine bioassay	Reason for Visit	005, 07, 60, Exposure, Special, Release	Samples collected following exposure or potential exposure in 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 collection of bioassay sample (random sampling program); this typically did not pertain to those on 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 shift worked (“A” = day shift)
WCP-455	Urine bioassay	Top of Card	“B”	Refers to shift worked (“B” = evening shift)
WCP-455	Urine bioassay	Top of Card	“O”	Refers to shift worked (“O” = midnight shift)
WCP-455	Urine bioassay	F, HG and OTHER	“B” (128)	Indicates shift and 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/d
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

ATTACHMENT A
CODES USED IN INTERNAL DOSE RECORDS

Form identifier	Measurement type	Column identifier	Code	Interpretation
5EA HPINT - Bioassay Results Report	Urine bioassay	Units	7	Bq/d
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/d
5EA HPINT - Bioassay Results Report	Breath analysis	Units	2	dpm/d
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/d
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 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); Author unknown (no date).
b. Form WCP-885 is referred to as "NCP-885" in Eckerman and Ward (1992); however, they appear to be the same form.
c. Around June 1956, form WCP-455 was modified with additional columns. Individual bioassay records reviewed indicate the exact date the newly expanded form was used. This 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 micrograms per liter with results recorded to the nearest integer, i.e., 0, 1, 2, etc.

Table A-2. *In vivo* record codes.[a]

Form identifier	Measurement type	Column identifier	Code	Interpretation
In Vivo Radiation Monitoring Report	Chest count	Surface contamination	Check mark, yes	Surface contamination on the subject was checked. If measurable activity was found, it was so noted on card.
In Vivo Radiation Monitoring Report	Chest count	Analysis Sequence		These align with Output - Analysis Sequence Results listed at bottom of card; they are not relevant to dose reconstruction process.
In Vivo Radiation Monitoring Report	Chest count	Output–Analysis Sequence Results	A. Enriched Uranium	
In Vivo Radiation Monitoring Report	Chest count	Output–Analysis Sequence Results	J. NLO Uranium	Refers to special spectrum region of interest for National Lead of Ohio, early operator of Fernald facility.
In Vivo Radiation Monitoring Report	Chest count	Hand-written notes	No Np noted	Presence or absence of Np-237 was qualitatively evaluated.
In Vivo Radiation Monitoring Report (with boxes for data entry)	Chest 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); Author unknown (no date).

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA FOR PGDP
Page 1 of 15

TABLE OF CONTENTS

LIST OF TABLES

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA FOR PGDP
Page 2 of 15

LIST OF FIGURES

B.1 PURPOSE

There are energy employees who, for a variety of reasons, were not monitored for internal exposure during the course of their employment at a DOE facility, or whose records of such monitoring are incomplete or unavailable. In such cases, dose reconstructors can use data from coworkers to approximate an individual's possible exposure. The purpose of this attachment is to provide the details of the calculation and assignment of intakes based on coworker data from the PGDP for estimating unmonitored exposures or where records of monitoring are incomplete or unavailable.

B.2 OVERVIEW

ORAUT-OTIB-0019, *Analysis of Coworker Bioassay Data for Internal Dose Assignment* (ORAUT 2005), describes the general process for analyzing bioassay data for the assignment of doses based on coworker results.

Uranium urine bioassay results from 1952 through 1988 were obtained from the PGDP Dosimetry Section. The data were copied from working files that are the basis for the PGDP annual dose reports and dose histories. Some of the bioassay data were taken from handwritten logs and added to the electronic database.

The database results are in units of micrograms or milligrams of total uranium per liter (µg/L or mg/L, respectively). All results were converted to µg/L for data analysis. There were a limited number of "beta" results in one of the files used (about 11%), which probably reflects measurement for ^{99}Tc. The decision was made not to use these data, because most of the results (> 97%) were reported as zero. The uranium records provide a more robust data set for statistical analysis.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA FOR PGDP

A statistical analysis of these data was performed in accordance with ORAUT-OTIB-0019 (ORAU 2005). The resultant values were input to the Integrated Modules for Bioassay Analysis (IMBA) Expert Office of Compensation Analysis and Support (OCAS) Edition computer program. The bioassay data were assumed to represent excretion of ^{234}U, which is biokinetically identical to other uranium isotopes and would result in the highest dose; a data fit was performed to obtain intake rates for assigning dose distributions. Bioassay results were converted from mass to activity before fitting.

B.3 DATA

B.3.1 Selected Bioassay Data

PGDP urinalysis data were extracted from two Microsoft® Access files provided by the PGDP Dosimetry Section. Table "PGDP_Historical_Urine from the PGDP_Dosimetry _Data_With_IDs database" was the source of urinalysis data from 1977 through 1988. Sample dates were taken from the *Sample_Date* field. Total uranium urinalysis results were taken from the *Result* field in the database; the *Units* field identified result units as micrograms per liter.

File "Historical_Urinalysis_Data_(unverified)" was the source of urinalysis data from October 1952 to mid-1977. Sample dates were taken from the *Date_Sample* field. Total uranium urinalysis results were taken from the *Urinalysis_Result* field in the database; there were no units associated with urinalysis results in this file. Comparison to results with identified units in table "PGDP_Historical_Urine" and with information presented in Section 5.3.1 of this TBD determined the results in this file were in units of milligrams per liter. Results in this file were converted to micrograms per liter for analysis.

Because both of the above files included PGDP uranium urinalysis data from 1977, the data were evaluated to identify and eliminate duplicate entries.

B.3.2 Analysis

Because of the number of sample results, data were analyzed by quarter from the final quarter of 1952 through 1988. The effective bioassay date was set equal to the midpoint of the analysis period. A lognormal distribution was assumed, and the 50th and 84th percentiles were calculated for each quarter, using the method described in ORAUT (2005). Table B-1 lists the statistical analysis results.

B.4 INTAKE MODELING

B.4.1 Assumptions

The IMBA Expert OCAS-Edition computer program requires urine results to be in units of activity per day. The total uranium results are in units of micrograms per liter; therefore, the results were multiplied by 1.4 to normalize them to the Reference Man excretion rate of 1,400 mL/d. Bioassay results were converted from mass to activity before fitting assuming 0.0389 Bq/µg, characteristic of low-enrichment (2%) uranium. Low-enrichment uranium feed is the default value if the specific location where a claimant worked is not available.

The uncertainty for each result in the intake calculation was assumed to be normally distributed. All results were equally weighted by applying a uniform absolute error of 1, indicating to IMBA that all results are (assumed to be) equally precise. A chronic exposure pattern was assumed for PGDP

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA FOR PGDP
Page 4 of 15

Table B-1. Summary of quarterly uranium 24-hour urinary excretion rate analyses, 1952 to 1988.

Effective sample date	50th percentile (Bq/d)	84th percentile (Bq/d)	No. of samples	No. of employees
11/14/1952	0.599	3.246	199	106
8/15/1953	0.205	0.980	1267	646
11/15/1953	0.362	1.373	1336	650
2/15/1954	0.578	2.023	1617	675
5/15/1954	0.466	1.643	1387	621
8/15/1954	0.383	0.951	1302	610
11/15/1954	0.330	1.010	1124	515
2/15/1955	0.331	1.327	1234	600
5/15/1955	0.471	1.419	1189	552
8/15/1955	0.411	1.185	1726	851
11/15/1955	0.468	1.465	1896	759
2/15/1956	0.466	1.454	2098	796
5/15/1956	0.407	0.800	1872	834
8/15/1956	0.323	0.739	1886	968
11/15/1956	0.328	0.930	1643	798
2/15/1957	0.377	0.866	1955	870
5/15/1957	0.453	1.063	2225	805
8/15/1957	0.316	0.886	2232	1005
11/15/1957	0.351	0.900	2399	967
2/15/1958	0.415	1.028	2307	889
5/15/1958	0.353	0.859	2217	939
8/15/1958	0.425	1.040	2291	899
11/15/1958	0.365	0.857	2139	846
2/15/1959	0.380	1.046	1895	818
5/15/1959	0.277	0.770	2026	916
8/15/1959	0.243	0.655	2134	961
11/15/1959	0.311	0.930	1909	873
2/15/1960	0.260	0.737	1780	885
5/15/1960	0.197	0.581	1854	1037
8/15/1960	0.195	0.558	1599	857
11/15/1960	0.287	1.119	1961	866
2/15/1961	0.297	1.083	1782	781
5/15/1961	0.190	0.551	1782	944
8/15/1961	0.176	0.529	1388	742
11/15/1961	0.340	0.977	1269	644
2/15/1962	0.447	1.284	1328	599
5/15/1962	0.412	0.950	1359	659
8/15/1962	0.302	0.797	1056	574
11/15/1962	0.369	0.911	1149	634
2/15/1963	0.343	0.929	1237	614
5/15/1963	0.327	0.807	1069	522
8/15/1963	0.228	0.597	931	521
11/15/1963	0.274	0.678	1096	569
2/15/1964	0.316	0.720	977	556
5/15/1964	0.305	0.917	1053	607
8/15/1964	0.286	0.616	624	376

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA FOR PGDP
Page 5 of 15

Effective sample date	50th percentile (Bq/d)	84th percentile (Bq/d)	No. of samples	No. of employees
11/15/1964	0.233	0.540	590	396
2/15/1965	0.213	0.461	578	434
5/15/1965	0.196	0.441	643	461
8/15/1965	0.242	0.467	424	329
11/15/1965	0.279	0.520	471	360
2/15/1966	0.326	0.672	513	393
5/15/1966	0.336	0.683	543	447
8/15/1966	0.293	0.598	536	412
11/15/1966	0.318	0.558	413	342
2/15/1967	0.305	0.612	545	418
5/15/1967	0.289	0.689	623	457
8/15/1967	0.275	0.503	368	321
11/15/1967	0.224	0.561	452	351
2/15/1968	0.241	0.523	527	410
5/15/1968	0.241	0.449	374	298
8/15/1968	0.293	0.647	416	344
11/15/1968	0.421	0.981	684	433
2/15/1969	0.529	1.165	1040	577
5/15/1969	0.394	0.840	868	514
8/15/1969	0.278	0.677	616	398
11/15/1969	0.344	0.847	707	417
2/15/1970	0.218	0.661	946	569
5/15/1970	0.228	0.636	753	482
8/15/1970	0.223	0.764	610	421
11/15/1970	0.254	0.912	228	190
2/15/1971	0.260	0.797	760	535
5/15/1971	0.208	0.634	508	371
8/15/1971	0.253	0.746	637	447
11/15/1971	0.341	0.947	536	325
2/15/1972	0.298	0.875	782	490
5/15/1972	0.163	0.570	602	369
8/15/1972	0.176	0.548	611	431
11/15/1972	0.216	0.736	654	438
2/15/1973	0.175	0.632	682	417
5/15/1973	0.248	0.624	443	300
8/15/1973	0.212	0.630	532	355
11/15/1973	0.275	0.735	493	319
2/15/1974	0.221	0.665	608	419
5/15/1974	0.177	0.546	179	161
8/15/1974	0.232	0.695	369	291
11/15/1974	0.354	0.980	549	351
2/15/1975	0.422	1.185	930	584
5/15/1975	0.347	0.818	924	591
8/15/1975	0.307	0.698	636	404
11/15/1975	0.362	0.890	778	510
2/15/1976	0.401	0.869	1025	653

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA FOR PGDP
Page 6 of 15

Effective sample date	50th percentile (Bq/d)	84th percentile (Bq/d)	No. of samples	No. of employees
5/15/1976	0.321	0.876	1035	713
8/15/1976	0.329	0.906	1025	768
11/15/1976	0.384	0.965	754	582
2/15/1977	0.260	0.879	1097	750
5/15/1977	0.334	0.972	928	634
8/15/1977	0.321	0.964	954	617
11/15/1977	0.432	1.189	1079	673
2/15/1978	0.418	1.037	1562	925
5/15/1978	0.330	0.862	1356	788
8/15/1978	0.245	0.603	1300	852
11/15/1978	0.209	0.621	1431	856
2/15/1979	0.071	0.376	857	700
5/15/1979	0.159	0.577	1301	828
8/15/1979	0.170	0.447	937	721
11/15/1979	0.258	0.658	1227	820
2/15/1980	0.229	0.674	1339	880
5/15/1980	0.117	0.330	1218	770
8/15/1980	0.120	0.318	1002	757
11/15/1980	0.109	0.232	1003	714
2/15/1981	0.100	0.297	983	681
5/15/1981	0.081	0.243	892	651
8/15/1981	0.097	0.236	1024	738
11/15/1981	0.083	0.207	847	638
2/15/1982	0.106	0.279	913	694
5/15/1982	0.061	0.199	644	466
8/15/1982	0.086	0.267	724	514
11/15/1982	0.081	0.250	686	511
2/15/1983	0.107	0.291	780	544
5/15/1983	0.116	0.307	618	466
8/15/1983	0.060	0.214	731	587
11/15/1983	0.098	0.276	695	489
2/15/1984	0.088	0.275	843	586
5/15/1984	0.069	0.245	484	369
8/15/1984	0.051	0.193	634	469
11/15/1984	0.062	0.222	732	484
2/15/1985	0.086	0.384	862	561
5/15/1985	0.055	0.249	902	610
8/15/1985	0.052	0.157	753	513
11/15/1985	0.053	0.156	1005	717
2/15/1986	0.054	0.250	926	585
5/15/1986	0.039	0.106	904	674
8/15/1986	0.072	0.240	812	600
11/15/1986	0.055	0.185	960	671
2/15/1987	0.088	0.259	1050	632
5/15/1987	0.061	0.213	1063	630
8/15/1987	0.107	0.282	1050	612

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA FOR PGDP
Page 7 of 15

Effective sample date	50th percentile (Bq/d)	84th percentile (Bq/d)	No. of samples	No. of employees
11/15/1987	0.091	0.229	1008	658
2/15/1988	0.114	0.425	1592	754
5/15/1988	0.119	0.341	1153	646
8/15/1988	0.089	0.257	990	607
11/15/1988	0.110	0.358	1180	729

workers; this pattern approximates a series of acute intakes with unknown intake dates. Intakes were assumed to be by inhalation using a default breathing rate of 1.2 m^3/hr and a 5-µm activity median aerodynamic diameter (AMAD) particle size distribution.

The database reported all results as "uranium." The bioassay data were assumed to represent excretion of ^{234}U. All uranium isotopes considered have long half-lives in relation to the assumed intake period so radioactive decay is not a consideration. In addition, all uranium isotopes are biokinetically identical so there is no effect on the fitting of the data for intake determination. Uranium-234 was the isotope selected because it would result in the highest internal dose; the International Commission on Radiological Protection (ICRP) Publication 68 dose coefficients (also referred to as dose conversion factors) for ^{234}U are 7% to 31% larger than those for ^{235}U, ^{236}U, and ^{238}U (ICRP 1995). Because of the isotopic compositions of the source terms, the assumption of ^{234}U will yield doses that are favorable to claimants.

PGDP received uranium and began enrichment operations in July 1952 and first withdrew enriched uranium in November 1952. The November 1952 period is consistent with uranium urinalysis data; however, the first intake period was conservatively assumed to begin on June 1, 1952.

B.4.2 Bioassay Fitting

The IMBA Expert OCAS-Edition computer program was used to fit the bioassay results to a series of inhalation intakes. Quarterly data from 1952 through 1988 were fit as a series of chronic intakes.

The intake assumptions were based on patterns observed in the bioassay data. The analysis for this TBD used periods with constant chronic intake rates by selecting time spans for which the bioassay results are of similar magnitudes. The analysis started a new chronic intake period if the data indicated a significant sustained change in the bioassay results. By this method, the period from 1952 through 1988 was divided into two chronic intake periods.

B.4.3 Uranium Material Types

Section 5.2 in the body of this document indicates uranium could be present in material with all three lung clearance rates (F, M, and S). Therefore, all three material types were evaluated. The bioassay results were entered into IMBA and the 50th-percentile intakes that resulted are shown in the figures in this attachment. In these figures, quarterly bioassay data used in the fits are shown as blue dots (●), and data that are not used in the fits are shown as red dots (●).

The type S uranium compounds present at PGDP have very long radiological half-lives, and the body retains the compounds for long periods. Therefore, the excretion results for different chronic intake periods are not independent for type S materials. For example, an intake in the 1950s could contribute to urinary excretion in the 1980s and later. To avoid potential underestimation of intakes for people who worked at PGDP for relatively short periods, each chronic intake of type S material was fit independently, using only the bioassay results from the single intake period. This will result in an overestimate of intakes for assumed type S exposures extending through both assumed intake periods.

B.4.3.1 Type F

Uranium urine results were fit using a type F material. Figure B-1 shows the fits to the 50th-percentile values from all intakes. The same intake periods were applied to the 84th-percentile values because the values followed a similar pattern, as shown in Figure B-2. These depict the expected excretion rates from an individual exposed for all the periods at the 50th- and 84th-percentile intake rates, respectively. Table B-2 summarizes the intake periods and corresponding intake rates for the 50th and 84th percentiles. The geometric standard deviations (GSDs) were determined by dividing the 84th-percentile intake rates by the 50th-percentile intake rates.

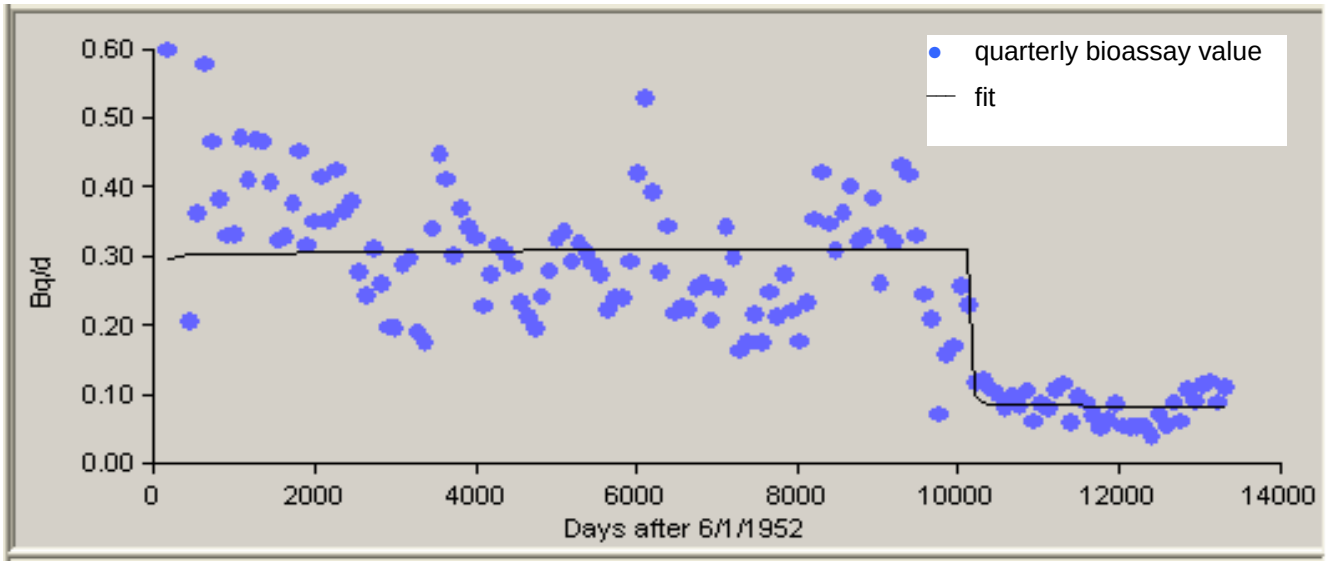

Figure B-1. 50th-percentile uranium urinalysis data used to estimate intakes of type F uranium occurring June 1, 1952, to December 31, 1988.

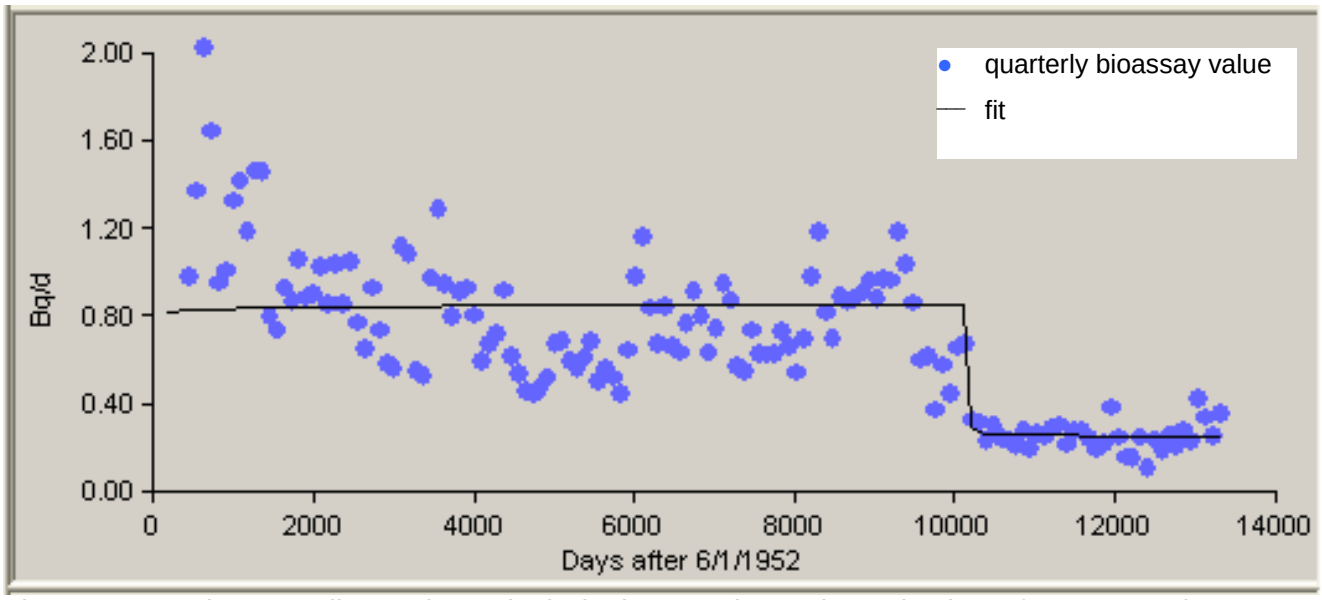

Figure B-2. 84th-percentile uranium urinalysis data used to estimate intakes of type F uranium occurring June 1, 1952, to December 31, 1988.

Table B-2. Type F uranium intake periods and rates.

Start date	Stop date	Uranium intake rate (Bq/d)		GSD
		50th percentile	84th percentile	
6/1/1952	3/31/1980	1.11	3.05	2.75
4/1/1980	12/31/1988	0.279	0.862	3.09

B.4.3.2 Type M

The intake periods used in the type F fits were applied to the type M material fits. Figures B-3 and B-4 show the fits to the 50th- and 84th-percentile values, respectively, from all intakes. Table B-3 summarizes the intake periods and corresponding intake rates for the 50th and 84th percentiles. The GSDs were determined as described for the type F intake rates.

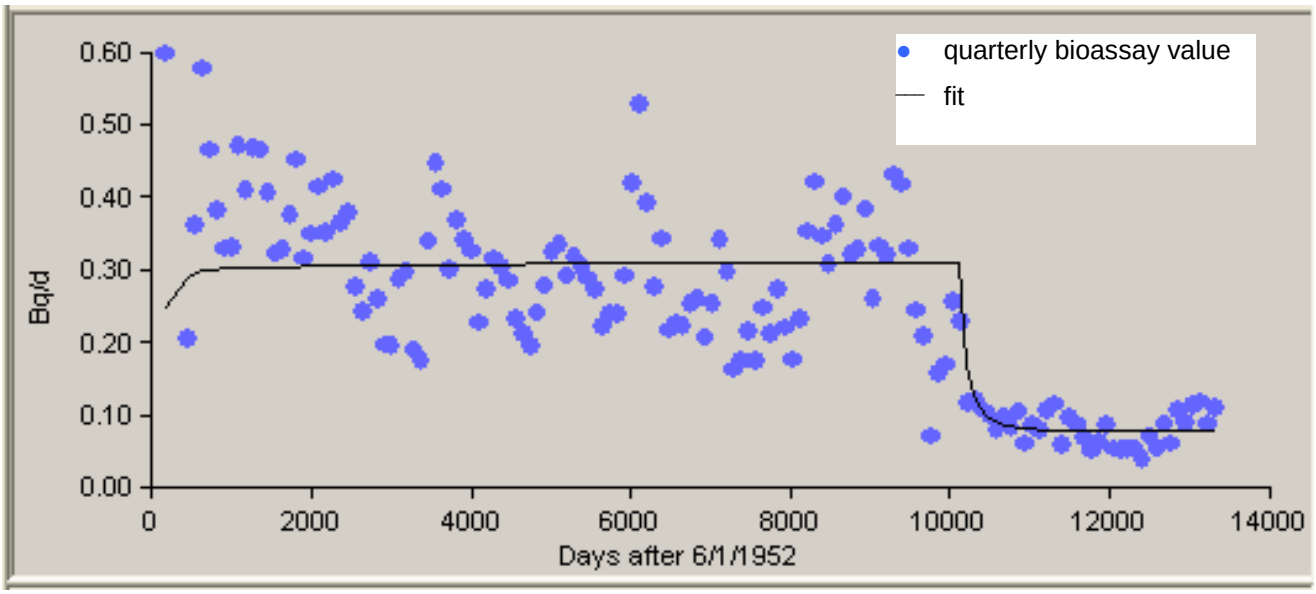

Figure B-3. 50th-percentile uranium urinalysis data used to estimate intakes of type M uranium occurring June 1, 1952, to December 31, 1988.

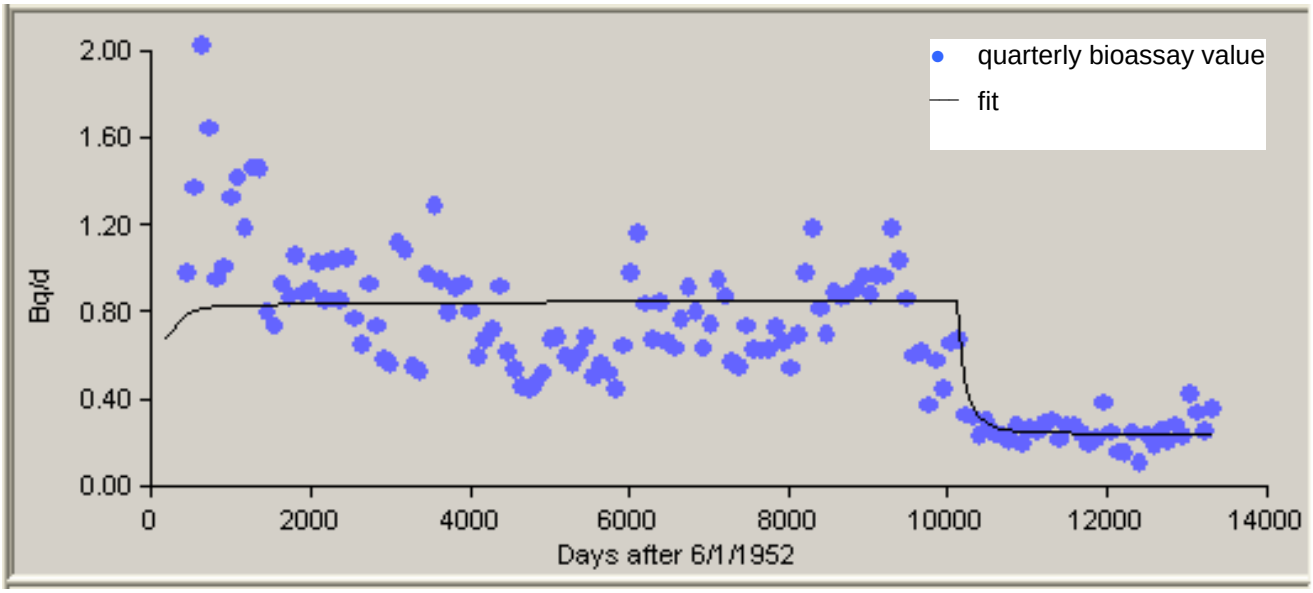

Figure B-4. 84th-percentile uranium urinalysis data used to estimate intakes of type M uranium occurring June 1, 1952, to December 31, 1988.

Table B-3. Type M uranium intake periods and rates.

Start date	Stop date	Uranium intake rate (Bq/d)		GSD
		50th percentile	84th percentile	
6/1/1952	3/31/1980	4.53	12.42	2.74
4/1/1980	12/31/1988	1.07	3.35	3.13

B.4.3.3 Type S

The intake periods used in the type F and M fits were applied to the type S material fits. As discussed above, each chronic intake period for type S material was fit independently. Figures B-5 and B-6 show the individual fits for the 50th-percentile values. The 84th-percentile values were fit similarly. Table B-4 summarizes the intake rates for the 50th- and 84th-percentile values. The GSDs were determined as described for the type F intake rates.

Figures B-7 and B-8 show the 50th- and 84th-percentile predicted excretion rates, respectively, from all type S intakes.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA FOR PGDP
Page 12 of 15

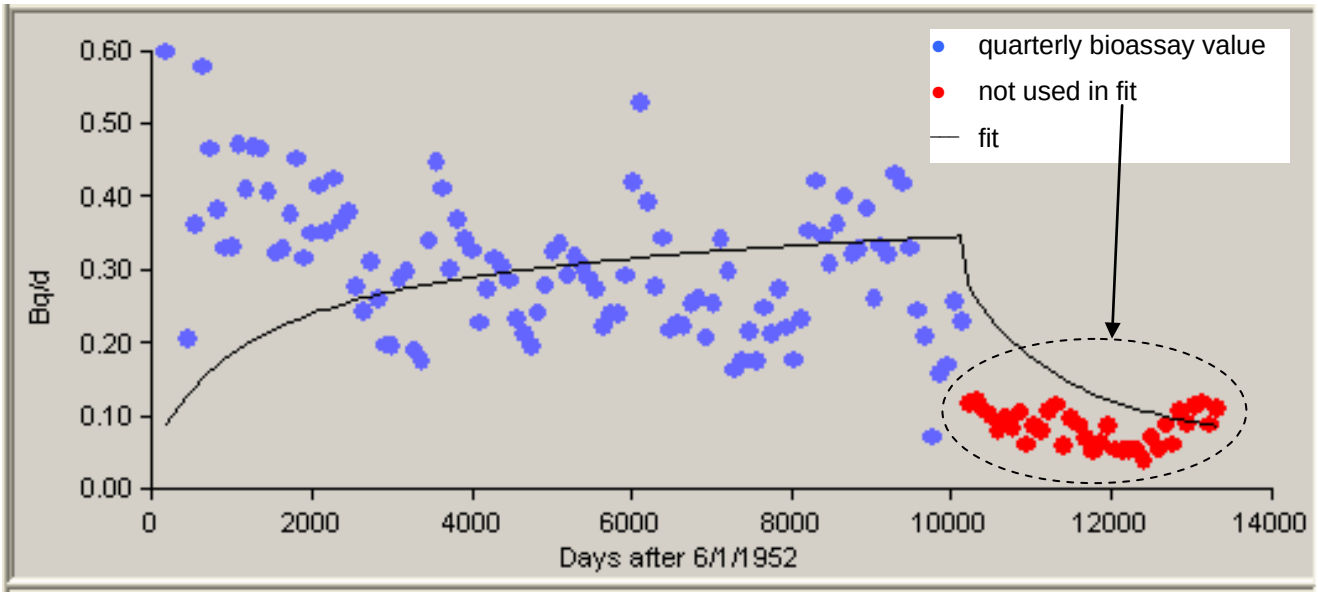

Figure B-5. 50th-percentile uranium urinalysis data used to estimate intakes of type S uranium occurring June 1, 1952, to March 31, 1980.

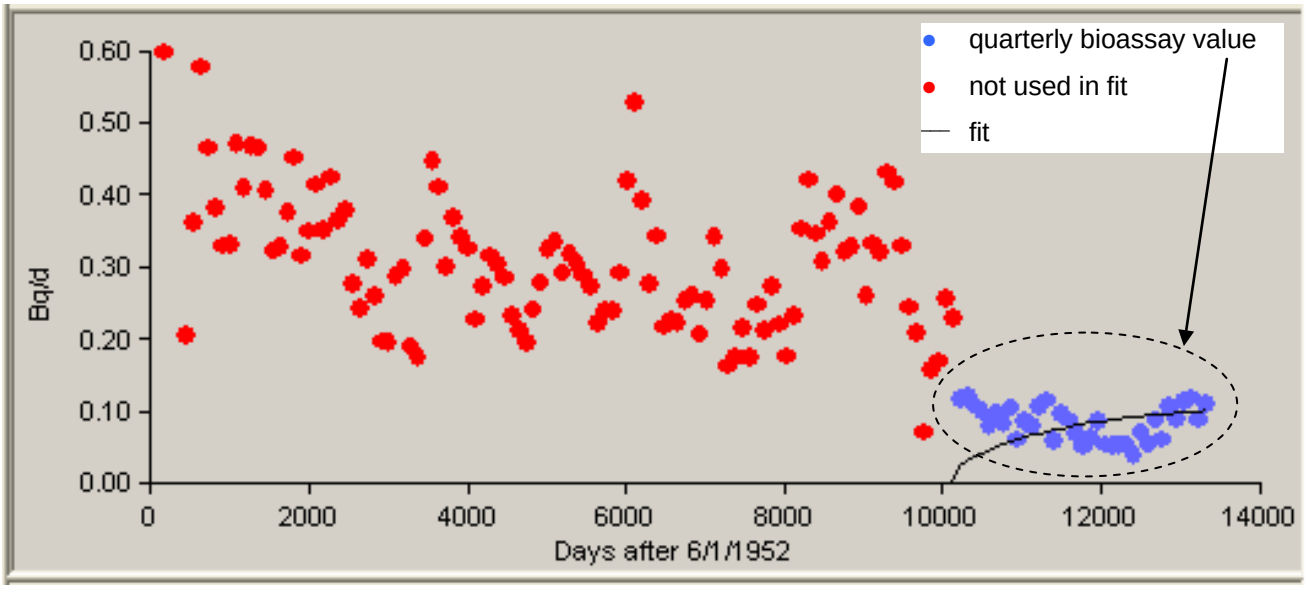

Figure B-6. 50th-percentile uranium urinalysis data used to estimate intakes of type S uranium occurring April 4, 1980 to December 31, 1988.

Table B-4. Type S uranium intake periods and rates.

		Uranium intake rate (Bq/d)		
Start date	**Stop date**	**50th percentile**	**84th percentile**	**GSD**
6/1/1952	3/31/1980	51.45	138.3	2.69
4/1/1980	12/31/1988	18.91	59.11	3.13

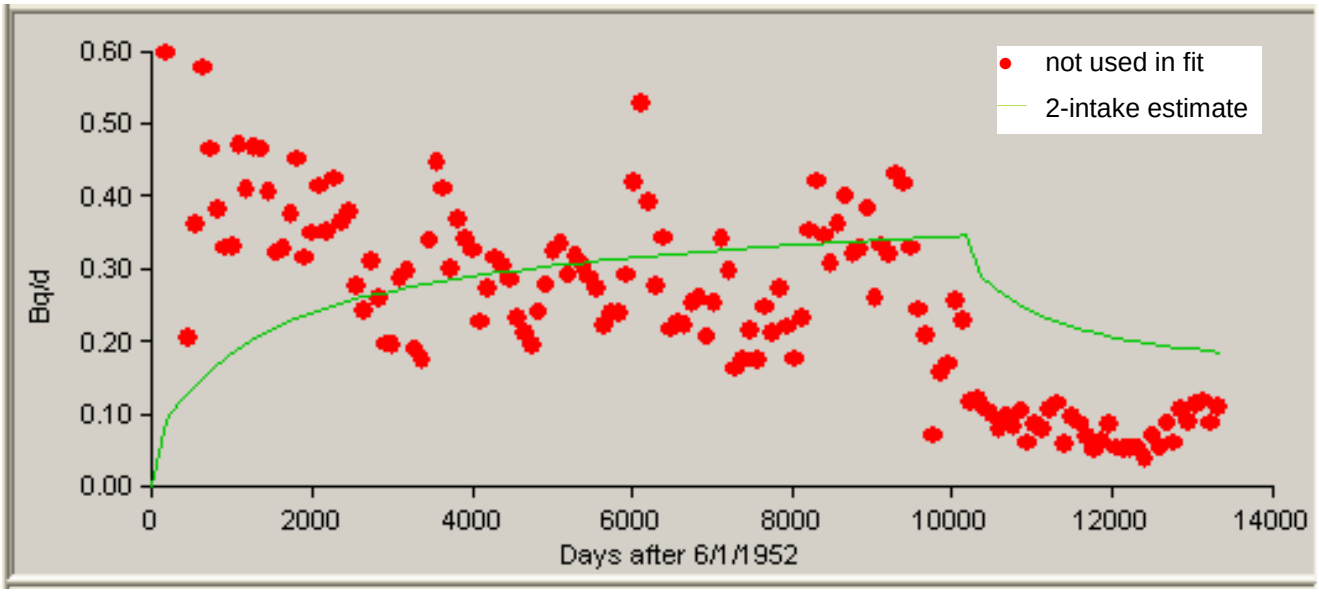

Figure B-7. Predicted 50th-percentile urinary excretion of type S uranium from 1952 to 1988 based on two independent intakes, compared with bioassay data.

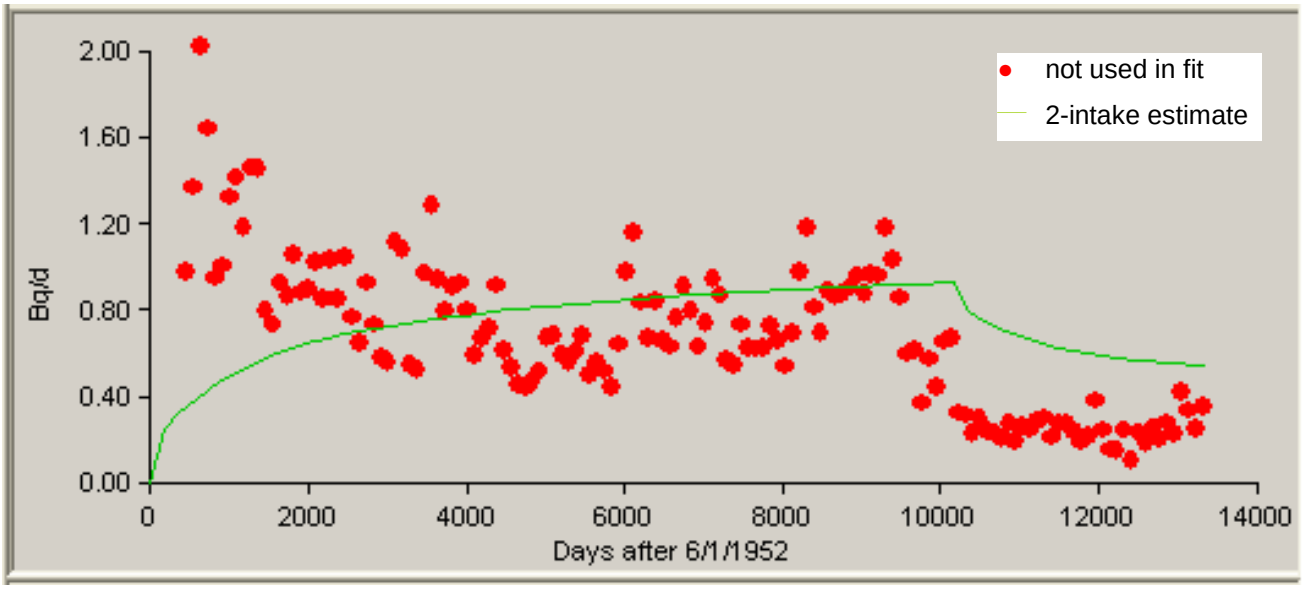

Figure B-8. Predicted 84th-percentile urinary excretion of type S uranium from 1952 to 1988 based on two independent intakes, compared with bioassay data.

B.5 ASSIGNMENT OF INTAKES AND DOSES

B.5.1 Intake Rate Summary

A summary of uranium intake rates and GSDs for periods from 1952 to 1988 is presented in Tables B-5, B-6, and B-7 for type F, M, and S material, respectively. These tables include potential intakes that may have occurred as early as June 1, 1952. When calculating doses to individuals from their own bioassay data, a GSD of 3 is assigned to account for biological variation and uncertainty in the

models. Therefore, a minimum GSD of 3 is assigned to the coworker intake distribution. Any calculated GSD less than 3 is adjusted upwards to 3.

The 50th-percentile intakes were used to calculate 95th-percentile intakes based on the following methodology:

$$\text{95th-percentile intake} = \text{50th-percentile intake} \times GSD^{1.645}$$

The tables listing the 50th- and 95th-percentile intakes are the values the dose reconstructor should use when assigning coworker intakes. The 50th-percentile intakes are assigned as a lognormal distribution with the associated geometric standard deviation (GSD) listed in each table. The 95th-percentile intakes are assigned using a constant distribution.

Generally speaking, the 50th-percentile coworker intakes are adequate to reconstruct a worker's unmonitored internal dose. However, there are isolated instances where a worker could be assigned the 95th percentile coworker intake rates. The following conditions should be met for a dose reconstructor to apply the 95th percentile coworker intake rates:

1) A worker does NOT have internal monitoring data or has voids in his monitoring data that cannot be adequately reconstructed using the available internal dosimetry data; **AND**

2) The worker was ROUTINELY exposed to elevated internal radiation hazards. (This may be determined by dosimetry records, telephone interviews, and/or DOE/DOL records.)

Table B-5. Type F uranium intake periods and rates.

		Uranium intake rate (Bq/d)			
Start date	Stop date	50th percentile	84th percentile	GSD	95th percentile (Bq/d)
6/1/1952	3/31/1980	1.11	3.05	3	6.8
4/1/1980	12/31/1988	0.279	0.862	3.1	1.8

Table B-6. Type M uranium intake periods and rates.

		Uranium intake rate (Bq/d)			
Start date	Stop date	50th percentile	84th percentile	GSD	95th percentile (Bq/d)
6/1/1952	3/31/1980	4.53	12.42	3	27.6
4/1/1980	12/31/1988	1.07	3.35	3.1	6.9

Table B-7. Type S uranium intake periods and rates.

		Uranium intake rate (Bq/d)			
Start date	Stop date	50th percentile	84th percentile	GSD	95th percentile (Bq/d)
6/1/1952	3/31/1980	51.45	138.3	3	313.5
4/1/1980	12/31/1988	18.91	59.11	3.1	121.6

B.5.2 Contribution from Contaminants in Recycled Uranium

Throughout the DOE complex, spent fuel from fission reactors has been processed to recover uranium for recycling. Because the uranium streams at PGDP could have contained recycled uranium, the dose from the added constituents, including plutonium, ^{237}Np, and ^{99}Tc, must be

included. See Section 5.2 of this TBD for information about intake values in relation to the uranium intake amounts.

B.5.3 Dose Assignment

For most cases, individual doses should be calculated from the 50th-percentile intake rates. For cases in which the 50th-percentile intake rates are not appropriate as discussed above, dose reconstructors should use the 95th-percentile intake rates applicable to the solubility class of material that are provided in Table B-5, B-6, or B-7. Dose reconstructors should select the material type that is the most favorable to claimants. A comparison of the intake rates shows that the intake rate substantially increases as the material solubility decreases. However, because lower solubility materials remain in the lungs for longer periods, while higher solubility materials are transferred to the systemic organs, it is necessary to compare the annual doses on a case-by-case basis to determine which will deliver the larger dose to the organ of interest. Recycled uranium contaminants, when appropriate for the period, are a factor in this comparison.

The lognormal distribution is selected in the Interactive RadioEpidemiological Program (IREP), with the calculated dose entered as Parameter 1 and the associated GSD as Parameter 2. The GSD is associated with the intake, so it is applied to all annual doses determined from the intake period. If the 95th-percentile intakes are used, they are assigned as a constant distribution.

Coworker intakes can be extended out to later years if needed. However, the vast majority of PGDP employees with a potential for radiological exposure are likely to have been monitored in recent years.

ATTACHMENT C
GENERAL INDICATION OF POTENTIAL FOR INTERNAL EXPOSURES
Page 1 of 2

The purpose of the following examples is to give the dose reconstructor a general idea of the types of job categories, work activities, and work locations for which internal dose assignment might be necessary.

Examples of job categories that might have little or no potential for workplace internal exposures:

Administrator	Draftsman	Program analyst
Assistant	Groundskeeper	Programmer
Business systems specialist	Instructor	Radio operator
Cafeteria worker	Manager	Recruiter
Checker	Medical technician	Scheduler
Clerk	Office supervisor	Secretary
Computer specialist	Planner	Telephone operator
Dispatcher	Quality assurance specialist	

Assignment of environmental internal dose might be appropriate for the job categories listed above on a case-by-case basis (ORAUT 2004b).

Examples of job categories that generally have some potential for workplace internal exposures depending on job specifics:

Biologist	Foreman	Patrolman
Boilermaker	Foundry worker	Photographer
Bricklayer	Heavy equipment operator	Scientist
Carpenter	HP analyst	Security guard
Construction worker	Instrument mechanic	Specialist
Driver	Insulator	Storekeeper
Electrician	Ironworker	Supervisor
Electronics technician	Janitor	Surveyor
Engineer	Laborer	Technician
Equipment operator	Mechanic	
Firefighter	Painter	

Examples of job categories that probably have some potential for workplace internal exposures:

Analytical chemist	Material handler	Radiation monitor
Assembly worker	Metallurgist	Radiochemist
Chemical operator	Millwright	Reactor operator
Fabricator	Pipefitter	Steamfitter
Glovebox worker	Plumber	Waste handler
HP technician	Processor	Welder
Machinist	Production worker	

In general, job categories that had increased potential for exposures from external radiation were similar to those for internal radiation. The jobs and tasks that appear to involve the greatest potential for radiation exposure included ash handling (Building C-410), cylinder heel cleaning (Building C-400), derby processing (Building C-340), pulverizer operations (Building C-400), certain maintenance operations on the fluorination towers (Building C-410), maintenance on cascade equipment (cascades), cleaning of air filters (baghouses for Buildings C-400, C-410, C-420, and C-340), converter maintenance (Building C-720), flange grinding (Buildings C-340, C-400, C-410, and C-420),

maintenance of the hydrogenation towers (Building C-340), and decontamination building cleaning operations (Building C-400). Jobs and tasks classified as having moderate potential radiation exposures include cascade operators and instrument mechanics (cascades), green salt sweeping (Buildings C-410 and C-420), disassembly of compressor (Building C-720), disassembly of block valves (Building C-720), drumming of green salt (Building C-340), and baghouse cleaning for cascades (Buildings C-310 and C-315). Other jobs and tasks had lower potential for worker radiation exposure. In addition, due to the lack of aggressive contamination control programs, contamination from the operations discussed above presented potential exposures to surrounding work areas. Further, legacy contamination generated from these operations could have posed a potential radiation exposure to individuals with access to the PGDP site in later years (PACE and University of Utah 2000).

Table C-1. History of significant incidents and events.[a]

Incident date	Incident description	Facility
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
April 1968	Worker overexposure	Unknown
January 1978	Fire	C-315
1958 to 1962	CIP	C-331, C-333, C-335, and C-337
1974 to 1982	CIP/CUP	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. Additional incident information can be found in Pace and University of Utah 2000 and BCJ 2000.