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Dose Reconstruction
Project for NIOSH

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EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
12/21/2004	00	New Technical Basis Document for the K-25 Site – Occupational Internal Dose. First approved issue. Initiated by Jay J. Maisler.
01/06/2006	01	Incorporates recent direction from NIOSH to include DOL review comments in the Introduction. Incorporates information on bioassay codes provided by Task 5 in sections 5.3.2 and 5.4.2; adds Attachments A, B, C. Deletes reference to Y-12 bioassay program from section 5.3. Approved issue of Revision 01. Training required: As determined by the Task Manager. Initiated by Jay J. Maisler.
10/04/2006	01-PC-1	Page change revision to update required language in the Introduction on page 5. Adds Purpose and Scope sections on pages 6 and 7. Incorporates Worker Outreach comments in on page 12 in Section 5.5 and page 14 in Section 5.6. Adds RU sentence on page 11 in Section 5.4. Adds two references to page 17. Incorporates NIOSH formal review comments. Incorporates internal formal review comments. Corrects dates on page 6 in Section 5.3 for consistency. Provides reference to Internal Dosimetry Coworker Data for K-25. This revision results in no change to the assigned dose and no PER is required. No sections were deleted. Training required: As determined by the Task Manager. Initiated by Paul A. Szalinski. Approval: Signature on File 10/02/2006 Paul A. Szalinski, Document Owner Signature on File 10/02/2006 John M. Byrne, Task 3 Manager Signature on File 10/03/2006 Edward F. Maher, Task 5 Manager Signature on File 10/02/2006 Kate Kimpan, Project Director Brant A. Ulsh Signature on File for 10/04/2006 James W. Neton, Associate Director for Science

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ACRONYMS AND ABBREVIATIONS

Bq	becquerel
dpm	disintegrations per minute
g	gram
hr	hour
ICRP	International Commission on Radiological Protection
IMBA	Interactive Modules for Bioassay Analysis
IRF	intake retention fraction
KPA	kinetic phosphorescence analysis
K-25	Oak Ridge Gaseous Diffusion Plant, or the K-25 Plant
L	liter
MDA	minimum detectable amount
MDC	minimum detectable concentration
mg	milligram
ml	milliliter
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
ORNL	Oak Ridge National Laboratory
pCi	picocurie
ppb	parts per billion
ppm	parts per million
RU	recycled uranium
TRU	transuranic
U.S.C.	United States Code
µCi	microcurie
µg	microgram
µm	micrometer

5.0 OCCUPATIONAL INTERNAL DOSE

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) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work.

As noted above, the statute 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 contains an exclusion with respect to the Naval Nuclear Propulsion Program, 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 dosimetry monitoring results are considered valid for use in 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:

- Radiation from naturally occurring radon present in conventional structures

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

- Radiation from diagnostic X-rays received in the treatment of work-related injuries

5.2 PURPOSE

The purpose of this document is to provide a uniform and consistent approach to assessing occupational internal dose at the Oak Ridge Gaseous Diffusion Plant, also called K-25. The document provides guidance to dose reconstructors on input parameters that are specific to employees of the Oak Ridge Gaseous Diffusion Plant (K-25), as well as the approach for employees with either missing or no monitoring information. *Internal Dosimetry Coworker Data for K-25* (ORAUT 2005) should be used for unmonitored workers who have the potential for intakes.

5.3 SCOPE

K-25 began gaseous diffusion enrichment production in 1945. Workers handled mainly uranium hexafluoride (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 93% (by weight) of ^{235}U from 1945 to 1964. After 1964, the highest enrichment was 5%.[2] The processing of recycled uranium (RU), which began in 1952, affected a broad range of processes and activities in facilities and locations at K-25 (BJC 2000). 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.

Facilities with significant involvement in RU processing included UF_6 feed production facilities, a waste ash pulverization and uranium recovery facility, decontamination and uranium recovery facilities, facilities fed by waste streams from decontamination facilities, other facilities performing more limited uranium recovery and decontamination activities, and the K-25 enrichment cascade, which operated in a variety of different configurations over time. In 1985, DOE had an overcapacity of enrichment capabilities and placed K-25 on standby status. DOE officially shut down the plant in 1987.

The primary method for monitoring employees for intakes of radionuclides at K-25 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 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)[3]. Data are available from 1952 to the present for both *in vivo* and *in vitro* analysis records and associated interpretations.

A review of in-house procedures used to assess the concentration of uranium in urine indicated 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*

[2] The predominant enrichment level was 3.0%. 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.

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

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 regarding enrichment, solubility, and the contribution of environmental uranium. In addition, samples were collected at work and during the middle of the workweek, so that cross-contamination and the inability to unfold soluble from insoluble intake fractions contributed to the uncertainty.

Nonetheless, dose reconstructors can prepare reliable, yet favorable to claimant 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 (see Section 5.8).

Section 5.2 provides guidance on selection of source terms. Sections 5.3 and 5.4 involve interpretation of *in vitro* and *in vivo* measurement results, respectively, each including instructions for assessing dose for both monitored and unmonitored employees. Section 5.5 summarizes the existing data analysis, and Section 5.6 identifies significant incidents with internal dose potential.

5.4 SOURCE TERM

The primary mission of K-25 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 93.5% ^{235}U (DOE 2000a). 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 2000a).

Certain TRU isotopes have been present at K-25 including ^{237}Np and ^{239}Pu. These resulted from the processing of reactor tails. Reactor tails were fed to the cascade from 1953 to 1964, and again from 1969 to 1976, with the exception of 1971 when none of the feed was of reactor origin (Smith 1984, p. 9). When this processing occurred, approximately 19% of the feed materials in use at the plant were reactor tails (DOE 2000a). The tails included 0.2 ppm of neptunium, and 4.0 ppb of plutonium.

At K-25, 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, dose reconstructors can use Table 5-1 to derive the isotopic fractions associated with each microgram of uranium reported in an analytical result. The default isotopic fraction provided in Table 5-1 is related to the typical enrichment used at K-25 over the lifetime of the plant operation.

Table 5-1. 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-02	4.90E-03	6.83E-04
4% enriched feed	6.42E-02	3.17E-03	6.67E-05	1.18E-02
Low-enrichment (2% enriched) feed	2.52E-02	1.58E-03	3.33E-05	1.21E-02
Default	2.52E-02	1.58E-03	3.33E-05	1.21E-02

a. Sources: ANSI (1995)

Table 5-2 lists the principal radionuclides that comprise the K-25 source term and their absorption types (lung solubilities). Table 5-3 provides that information for the various uranium compounds.

Table 5-2. Principal radionuclides found at uranium facilities and gaseous diffusion plants.

Nuclide	Absorption type
Th-230	M, S
U-234	F, M, S
U-235	F, M, S
U-236	F, M, S
U-238	F, M, S
Pu-238	M, S
Pu-239/240	M, S
Np-237	M
Cm-242	M
Cm-244	M
Am-241	M
Tc-99	F, M

Table 5-3. Absorption classification for some uranium compounds.[a]

Compound	Chemical notation	Absorption classification[b]
Uranium hexafluoride	UF_6	F
Uranyl fluoride	UO_2F_2	F
Uranyl nitrate	$UO_2(NO_3)_2$	F
Uranyl acetate	$UO_2(C_2H_3O_2)_2$	F
Uranyl chloride	UO_2Cl_2	F
Uranyl sulfate	UO_2SO_4	F
Uranium trioxide	UO_3	F
Uranium tetrafluoride	UF_4	M
Uranium oxide	U_3O_8	M
Uranium dioxide	UO_2	M
Uranium tetroxide	UO_4	M
Ammonium diuranate	$(NH_4)_2 + U_2O_7$	M
Uranium aluminide	UAl_x	S
Uranium carbide	UC_2	S
Uranium-zirconium alloy	Uzr	S
High-fired uranium dioxide	UO_2	S

a. Source: DOE (2000b).
b. The absorption classification is assumed to be comparable to the ICRP 30 classification system of D, W and Y.

If information on the source term to which the claimant was exposed is available, the dose reconstructor should use that source term. However, if no source term information is available, the values and parameters shown in Table 5-4 provide input to the process.

Table 5-4. Source term summary by location.[a]

Location	Form	Mass % ^{235}U	Concentration per gram of U	
K-1131 Chemical Plant Stream 1 & 2 (1) RU as UO3 to Feed Hopper (2) UO3 to UO3 Reduction	UO_3	0.64	520 ppb 4.4 ppb 7.8 ppm 170 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
K-1131 Chemical Plant Stream 5 Tower Ash Disposal	UF_4	0.64	13,000 ppb 440 ppb 40 ppm 100 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
K-1131 Chemical Plant Stream 5 Tower Ash Disposal Max case for Pu	UF_4	0.64	13,000 ppb 4000 ppb 40 ppm 100 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
K-1131 Chemical Plant Stream 6 UF_6 to UF_6 Collection	UF_6	0.64	393.94 ppb 0.04 ppb 7.47 ppm 170.71 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
K-1131 Chemical Plant Stream 7 UF_6 to UF_6 to Process Vent	UF_6	0.64	0.00 ppb 0.00 ppb 200 ppm 200 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
K-1131 Chemical Plant Stream 8 RU as UF_6	UF_6	0.64	394.34 ppb 0.04 ppb 7.280 ppm 170.88 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
K-1131 Chemical Plant Stream 9 UO_3 Reduction to Process Vent	UO_2	0.64	0.00 ppb 0.00 ppb 10 ppm 0.00 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
ORGDP Stream 12 UF_6 Cylinder Heel	UF_6	0.64	26,000 ppb 4.00 ppb 70 ppm 100.00 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
ORGDP Stream 13 Hanford & Savannah River UF_6 Feed Autoclave to Cascade	UF_6	0.64	132.79 ppb 0.00 ppb 6.64 ppm 171.60 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
ORGDP Stream 14 PGDP RU to UF_6 Feed Autoclave	UF_6	0.65	5.00 ppb 0.00 ppb 1.96 ppm 0.00 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
ORGDP Stream 15 PGDP UF_6 Cylinder Heels	UF_6	0.65	5.00 ppb 0.00 ppb 40 ppm 0.00 ppm	^{237}Np *Pu[b] ^{99}Tc ^{236}U
ORGDP Stream 16 PGDP UF_6 to Cascade	UF_6	0.65	0.00 ppb 0.00 ppb 1.75 ppm 109.29 ppm	Np-237 Pu[b] Tc-99 U-236
ORGDP Stream 18 UF_6 to Cascade	UF_6	0.66	30.66 ppb 0.00 ppb 2.29 ppm 86.79 ppm	Np-237 Pu[b] Tc-99 U-236

Table 5-4 (Continued). Source term summary by location.[a]

Location	Form	Mass % ^{235}U	Concentration per gram of U	
ORGDP Stream 19 Depleted UF_6	UF_6	0.30	0.00 ppb 0.00 ppb 0.00 ppm 39.51 ppm	Np-237 Pu[b] Tc-99 U-236
ORGDP Stream 20 Enriched UF_6 Product	UF_6	3.0	50.00 ppb 0.00 ppb 1.77 ppm 394.69 ppm	Np-237 Pu[b] Tc-99 U-236
ORGDP Stream 21 Cascade Accumulation Stream	UO_2F_2	1.0	130,000 ppb 4.00 ppb 7500 ppm 100.00 ppm	Np-237 Pu[b] Tc-99 U-236
ORGDP Stream 22 Purge Cascade Stream	UF_6	3.0	0.00 ppb 0.00 ppb 87 ppm 100.00 ppm	Np-237 Pu[b] Tc-99 U-236
ORGDP Stream 24 ^{99}Tc Chem Traps Spent Sorbent Stream	UF_6	3.0	5.00 ppb 0.00 ppb 100,000 ppm 100.00 ppm	Np-237 Pu[b] Tc-99 U-236
K-1420 Stream 26 Chem Decon K-1420 to Effluent Discharge B & C Ponds Stream	UO_2F_2	1.0	2.00 ppb 0.02 ppb 200 ppm 100.00 ppm	Np-237 Pu[b] Tc-99 U-236
B & C Ponds Sludge	UO_2F_2	0.7	2.00 ppb 0.02 ppb 200 ppm 100.00 ppm	Np-237 Pu[b] Tc-99 U-236
K-1131 Chemical Plant Stream 10 RU as UF_6 UO_2 Hydrofluorination to Process Vent	UF_4	0.64	0.00 ppb 0.00 ppb 10 ppm 0.00 ppm	Np-237 Pu[b] Tc-99 U-236

a. Source: BJC (2000).
b. Pu = total of all plutonium isotopes.

Table 5-5 provides conversion factors specific to each facility for the activity per unit mass for each nonuranium radionuclide of interest.

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 ^{99}Tc (Smith 1984, Appendix 12; DOE 2000a).

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 K-25 source term. Table 5-6 provides a default isotopic distribution to use when only total uranium results are available for a particular measurement.

For K-25, unless specific information to the contrary is available, this analysis assumes a particle size of 5 µm activity median aerodynamic diameter (AMAD), as recommended in International Commission on Radiological Protection (ICRP) Publication 68 (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. The default isotopic distribution listed in Table 5-6 is based on typical enrichments used at K25 over the course of the operation. The presence of other radionuclides not present in the uranium decay series is based on operating experience at Paducah Gaseous Diffusion Plant. The recycled uranium feed material was similar for the gaseous diffusion plants; K-25 shipped 11,629 metric tons of recycled uranium to Paducah (BJC 2000).

Table 5-5. Facility-specific radionuclide conversion factors.

Process description	Activity per unit mass (Bq/µg U)							
	Np-237	**Pu-238**	**Pu-239**	**Pu-240**	**Pu-241**	**Pu-242**	**Tc-99**	**U-236**
K-1131 Chemical Plant Stream 1 & 2 (1) RU as UO3 to Feed Hopper (2) UO3 to UO3 Reduction	1.36E-05	1.39E-06	9.42E-06	2.21E-06	6.71E-05	3.19E-10	4.89E-03	4.06E-04
K-1131 Chemical Plant Stream 1 & 2 (1) RU as UO3 to Feed Hopper (2) UO3 to UO3 Reduction Max case for Pu	1.36E-05	1.27E-05	8.56E-05	2.01E-05	6.10E-04	2.90E-09	4.89E-03	4.06E-04
K-1131 Chemical Plant Stream 3 UO2 to UO2 Hydrofluorination	1.36E-05	1.39E-06	9.42E-06	2.21E-06	6.71E-05	3.19E-10	4.89E-03	4.06E-04
K-1131 Chemical Plant Stream 4 UF_4 to UF_4 Fluorination	1.36E-05	1.39E-06	9.42E-06	2.21E-06	6.71E-05	3.19E-10	4.89E-03	4.06E-04
K-1131 Chemical Plant Stream 5 Tower Ash Disposal	3.39E-04	1.39E-04	9.42E-04	2.21E-04	6.71E-03	3.19E-08	2.51E-02	2.39E-04
K-1131 Chemical Plant Stream 5 Tower Ash Disposal Max case for Pu	3.39E-04	1.27E-03	8.56E-03	2.01E-03	6.10E-02	2.90E-07	2.51E-02	2.39E-04
K-1131 Chemical Plant Stream 6 UF_6 to UF_6 Collection	1.03E-05	1.27E-08	8.56E-08	2.01E-08	6.10E-07	2.90E-12	4.68E-06	4.08E-04
K-1131 Chemical Plant Stream 7 UF_6 to UF_6 to Process Vent	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.25E-01	4.78E-04
K-1131 Chemical Plant Stream 8 RU as UF_6	1.03E-05	1.27E-08	8.56E-08	2.01E-08	6.10E-07	2.90E-12	4.56E-06	4.08E-04
K-1131 Chemical Plant Stream 9 UO_3 Reduction to Process Vent	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.27E-03	0.00E+00
K-1131 Chemical Plant Stream 10 RU as UF_6 UO_2 Hydrofluorination to Process Vent	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.27E-03	0.00E+00
ORGDP 12 ORGDP Stream 12 UF_6 Cylinder Heel	6.79E-04	1.27E-06	8.56E-06	2.01E-06	6.10E-05	2.90E-10	4.39E-02	2.39E-04
ORGDP 13 ORGDP Stream 13 Hanford & Savannah River UF_6 Feed Autoclave to Cascade	3.47E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.16E-03	4.10E-04
ORGDP 14 ORGDP Stream 14 PGDP RU to UF_6 Feed Autoclave	1.31E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.23E-03	0.00E+00
ORGDP 15 ORGDP Stream 15 PGDP UF_6 Cylinder Heels	1.31E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.51E-02	0.00E+00
ORGDP 16 ORGDP Stream 16 PGDP UF_6 to Cascade	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.10E-03	2.61E-04
ORGDP 18 ORGDP Stream 18 UF_6 to Cascade	8.00E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.44E-03	2.07E-04
ORGDP 19 ORGDP Stream 19 Depleted UF_6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.44E-05
ORGDP 20 ORGDP Stream 20 Enriched UF_6 Product	1.31E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.11E-03	9.43E-04
ORGDP 21 ORGDP Stream 21 Cascade Accumulation Stream	3.39E-03	1.27E-06	8.56E-06	2.01E-06	6.10E-05	2.90E-10	4.70E+00	2.39E-04
ORGDP 22 ORGDP Stream 22 Purge Cascade Stream	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.45E-02	2.39E-04
ORGDP 24 ORGDP Stream 24 ^{99}Tc Chem Traps Spent Sorbent Stream	1.31E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.27E+01	2.39E-04
K1420 26K-1420 Stream 26 Chem Decon K-1420 to Effluent Discharge B&C Ponds Stream	5.22E-08	6.33E-09	4.28E-08	1.01E-08	3.05E-07	1.45E-12	1.25E-01	2.39E-04
B&C Pond Sludge	5.22E-08	6.33E-09	4.28E-08	1.01E-08	3.05E-07	1.45E-12	1.25E-01	2.39E-04

Table 5-6. Default isotopic distribution.

Radionuclide	nCi/g U
Pu-239	67.5
Am-241	67.5
U-236	0.93
U-235	43.9
U-234	702
U-238	337.5
Np-237	5.4
Th-230	18.9
Tc-99	0.12

5.5 *IN VITRO* MEASUREMENT METHODS

From the start of plant operations in 1945, samples of urine from workers involved in enrichment operations were analyzed for radionuclide content. As time progressed, other employees were included in the monitoring program. Spot urine samples (a single void) were used as a screening method for indication of uranium intake. The spot samples were collected often in the early years. By 1950, 24-hr sample collection was instituted at K-25. Spot urine samples show greater variability than 24-hr samples (Medley, Kathren, and Miller 1994).

The monitoring methodology during the early years of K-25 operations is unknown as of the date of this TBD, however existing data show results from a variety of analytical techniques that patterned those performed at the Y-12 plant during the same time periods. Because of the plant's relative proximity to the Y-12 plant, and because professional interactions between the two organizations were likely, for the purposes of this report a monitoring program history similar to that found at Y-12 is assumed.

As site-specific information becomes available, this assumption may change. In the interim, however, it is assumed that fluorometric analysis of urine was the monitoring method of choice beginning in the mid-1940's and continuing until the late 1980's (ORAUT 2004, Sect. 5.2.2). In addition, uranium analysis by electrodeposition and alpha counting may also have been performed in order to assess intakes of enriched uranium. After 1989, uranium analysis by alpha spectrometry was the predominant in vitro monitoring method. As of 1991, a routine schedule of weekly, monthly, or job-specific bioassays was established depending on work assignment (MMES 1991a).

5.5.1 Measurement Types and Detection Levels

Table 5-7 lists the *in vitro* measurement types and detection levels assumed to be applicable during various time periods at K-25. The expected intake pattern in most cases is acute. At K-25, airborne and surface contamination was typically controlled to prevent intakes, so most would have been the result of unexpected releases. It is possible that small, intermittent releases occurred that were not immediately detected, so an individual could have had multiple acute intakes.

In some cases, a detection level for a particular radionuclide or analysis method may not be available. In that case, dose reconstructors should use the nominal detection levels in Table 5-8.

Likewise, if a monitoring result does not match with the time periods shown in Table 5-7, the detection level given in Table 5-7 for the applicable monitoring method and for the closest time period should be used.

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 K-25) and the MDC from that measurement type should be used. 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.

5.5.2 Reporting Formats and Codes

A variety of codes occurs on various urine bioassay records for K-25. A summary of the codes for 1948 to 1988 is given in Attachment A (Wallace, 2005a).

5.5.3 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 K-25 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 K-25. 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, sample cross-contamination could have occurred during the first collection.

Dietary intakes of uranium and diurnal fluctuations in metals excretion pose a potential problem in interpreting urine bioassay results for K-25 workers. Because studies of the average daily uranium excretion on Oak Ridge 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. No correction for environmental levels of uranium is required for samples analyzed by fluorimetry or KPA because the MDC is larger than the correction.

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

Period	Measurement type	Radionuclide	MDC[b] (mg/L)
1945 to 1957	In-house fluorimetry	Total uranium	10 microgram/liter[c]
1958 to 1989	In-house fluorimetry	Total uranium	5 microgram/liter
1950 to 1965	Gross alpha counting	Enriched uranium	3.3 dpm/ 100 milliliter
1965 to 1989	Gross alpha counting	Enriched uranium	1.8 dpm/100 milliliter
1989 to present	Alpha spectrometry	Isotopic uranium	0.01 dpm/100 milliliter

a. ORAUT (2004), Section 5.2.1.3.
b. MDC = minimum detectable concentration.
c. No non-zero values below a reporting level of 0.01 milligrams of uranium per liter appear in the reported data until 1958. Therefore, while the methodology may have been capable of detecting 5 micrograms of uranium per liter prior to that time, the nominal MDC was rounded up to the applicable significant figure.

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).

5.6 *IN VIVO* MEASUREMENT METHODS

Whole-body counting and other *in vivo* methods were implemented after 1965. This measurement method was used primarily in response to incidents and for assessing the magnitude of insoluble material intakes (ORNL, 1981). Even when routine whole-body counting was instituted for certain K-25 employees in the late 1960s, the counting frequency was sporadic and seldom greater than once per year. As of 1991, annual routine whole-body counting was established (MMES 1991b). A comprehensive listing of scheduled whole-body or lung counts was not available.

5.6.1 Measurement Types and Detection Levels

At K-25 whole-body counting was performed using a mobile counter provided by the Y-12 plant (sometimes referred to as the MMES Counter.) Whole body counting was also performed at other facilities. Table 5-9 lists general information about the detection capabilities of the counting systems for various periods.

Table 5-9. *In vivo* measurement types and detection levels for various periods.[a]

Period	Equipment	Measurement type	Radionuclide	MDA (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 (2000b); Hill and Strom (1993); BJC (1999, Tables 3.2 and 3.5); Scott and West (1967); Bassett (1985, p. 27).

5.6.2 Reporting Formats and Codes

A variety of codes occurs on various *in vivo* bioassay records for K-25. Summaries of the *in vivo* codes for 1965 to 1988 and 1989 to 1991 are shown on Attachments B and C (Wallace, 2005b and 2005c.)

5.6.3 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 to *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.

Uncertainties in the bioassay measurements were not stated in the records. For results near or at the reporting levels, dose reconstructors 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 K-25 workers could have had body burdens of ^{137}Cs from nonoccupational sources (e.g., fallout and consumption of specific foodstuffs). There is evidence neither of ^{137}Cs in the source term at K-25 nor of occupational intakes of ^{137}Cs among K-25 workers; therefore, no dose of record should be associated with these measurement results.

5.6.4 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.

5.7 SIGNIFICANT INCIDENTS WITH INTERNAL DOSE POTENTIAL

At this time, no information is available about incidents where there was a potential for internal dose. If such information should become available, this section will be revised in subsequent versions of this document.

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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 in a short period and generally considered instantaneous for dose reconstruction purposes.

bioassay
Measurement of amount or concentration of radioactive material either in the body or in biological material excreted or removed from the body; also called *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 over a long period.

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 are 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 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 using 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, $\exists$, of nondetection (Type II error) while accepting a probability, $\forall$, of erroneously deciding that a positive (nonzero) 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 that relate 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.

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 suspected).

spot sample

A single void of urine.

ATTACHMENT A
URINE BIOASSAY CODES 1948 TO 1988

Data name	Value	Description
Payroll	W	Weekly
Payroll	M	Monthly
Payroll	H	Hourly
SamType	R	Routine
SamType	F	Follow-up
SamType	M	Medical
SamType	S	Supervisory
SamType	D	Delayed
Recall	A	First recall
Recall	B	Second recall
Recall	C	Third recall
Recall	D	Fourth recall
Recall	T	Termination
Freq	M	Monthly
Freq	Q	Quarterly
Freq	S	Semi-Annual
Freq	1	1 month sampling
Freq	2	2 month sampling
Freq	3	3 month sampling
Freq	6	6 month sampling
Freq	S	Special
Atype	01	Uranium analysis type..aka..contamination code
Atype	02	Fluoride
Atype	03	Mercury
Atype	04	Alpha Activity
Atype	05	Beryllium
Atype	06	Lead
Atype	07	Plutonium-239
Atype	08	Trichloracetic Acid
Atype	09	Nickel
Atype	10	Strontium-90
Atype	11	Neptunium-237
Atype	12	Cadmium
Atype	13	Barium
Atype	14	Carbon-14
Atype	15	Radon
Atype	16	Technetium-99
Atype	17	Copper
Atype	18	Phenol
Atype	19	Lead (blood)
Atype	20	Chromium
Atype	21	Iron
Atype	22	Benzene (phenol)
Atype	23	Magnesium
Atype	24	Calcium
Atype	25	Molybdenum
Atype	26	Manganese
Atype	27	Cobalt
Atype	28	Zinc
SamCode	I	Industrial hygiene
SamCode	H	Health physics
Exc	1	Used when result exceeded size of format. Note(only occurred once)
Pdate		Prior date
Ptime		Prior time
Vsam		Volume of sample
Tsam	1	AM time sample was taken
Tsam	2	PM
Tsam	0	Not used

ATTACHMENT B
IN VIVO BIOASSAY CODES 1965 TO 1988

Data name	Value	Description
Surface	1	No surface contamination if in first position
Surface	2 or 3	Surface contamination if in first position
Surface		Second through sixth are meter readings
Atype	1	Uranium 235
Atype	3	Thorium
Atype	4	U-238
Atype	6	Thorium and U-238
Atype	7	All codes
Mtype	01	Unknown or Mixed U enriched
FTB		Front to back ratio one decimal place
U235LB		U-235 lung burden in microgram
TLB		Thorium Lung Burden in milligram
CC		Cesium count in nanocuries
PC		Potassium (K-40) count in grams
U238LB		U238 Lung Burden in milligrams
AC		Actinium count in nanocuries(one decimal place)
LC		Lead 212 count in nanocuries(one decimal place)
TC		Technetium 99 count in microcuries (one decimal place)

ATTACHMENT C
IN VIVO BIOASSAY CODES 1988 TO 1991

Name of field	Value	Description
Plant		Plant
Plant	K	K25
Plant	Y	Y12
Plant	X	X10
DOB		Date of birth
CountDate		Date of count
Dept_No		Department number
EAC_Code		EAC Code
HP_Area_Code		HP Area Code
ReasonCode		Reason Code
ReasonCode	1R	Incident routine
ReasonCode	1S	Incident acute
ReasonCode	2R	Follow-up routine
ReasonCode	2S	Follow-up acute
ReasonCode	3	Routine
ReasonCode	4	Background
ReasonCode	5	Other
ReasonCode	6	Control
ReasonCode	7	Random
ReasonCode	8	Start date
ReasonCode	9	Stop date
ReasonCode	0	Acute
ReasonCode	1	Old value:::Incident
ReasonCode	2	Old value:::Followup
SampleReason		Reason for sample
WBC_Type		Type of whole body count
WBC_Type	LUN	Lung count
WBC_Type	O	Other
WBC_Type	RBC	Red blood cells
WBC_Type	THY	Thyroid count
WBC_Type	WBL	Whole body lung
WBC_Type	WBC	Whole body count