ORAU Team **Dose Reconstruction Project for NIOSH** Technical Basis Document for Portsmouth Gaseous Diffusion Plant – Occupational Internal Dose	Document Number: ORAUT-TKBS-0015-5 Effective Date: 11/24/2004 Revision No.: 00 Controlled Copy No.: _____ Page 1 of 27

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RECORD OF ISSUE/REVISIONS

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ACRONYMS AND ABBREVIATIONS

^{241}Am	Americium-241
^{237}Np	Neptunium-237
^{238,239,240}Pu	Plutonium-238, -239, -240
^{234m}Pa	Protactinium-234
^{99}Tc	Technetium-99
^{228,230,231,232,234}Th	Thorium-228, -230, -231, -232, -234
^{201}Tl	Thallium-201
^{234,235,236,238}U	Uranium-234, -235, -236, -238

AEC	Atomic Energy Commission
amu	atomic mass unit
AMAD	Activity Median Aerodynamic Diameter

Bq	Becquerel
BJ	Bechtel Jacobs Corporation

CsI	Cesium Iodide

DOE	U.S. Department of Energy
dpm	disintegrations per minute
DU	depleted uranium
DR	Dose Reconstructor

ERDA	Energy Research and Development Administration
EU	enriched uranium

g	gram
GAT	Goodyear Atomic Corporation Technical Document

HEU	highly enriched uranium
hr	hour

ICRP	International Commission on Radiological Protection
ICP/MS	Inductively Coupled Plasma Mass Spectrometry
IL	investigation level

KPA	kinetic phosphorescence analysis
keV	kilovolt electric

LSC	liquid scintillation counting

μCi	microcurie
μg	microgram
MDA	minimum detectable activity or, for elemental uranium, minimum detectable amount
MDC	minimum detectable concentration
mg	milligram
ml	milliliter
MMES	Martin Marietta Energy Systems

M   Molar
MTU   metric tons uranium

NaI   Sodium Iodide
Nat U   natural uranium
nCi   nanocurie
NCRP   National Council on Radiation Protection and Measurements
NIOSH   National Institute for Occupational Safety and Health
NO_3 nitrate ion

ORNL   Oak Ridge National Laboratory

PAS   portable air samples
PGDP   Paducah Gaseous Diffusion Plant
PORTS   Portsmouth Gaseous Diffusion Plant

RU   recycled uranium

SA   specific activity

TBD   Technical Basis Document
TRU   transuranic

UO_2 uranium dioxide
UO_3 uranium trioxide
U_3O_8 uranium oxide
UF_4 uranium tetrafluoride
UF_6 uranium hexafluoride
USEC   United States Enrichment Corporation

5.0   <u>INTRODUCTION – OCCUPATIONAL INTERNAL DOSE</u>

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

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

Portsmouth Gaseous Diffusion Plant (PORTS) operations, which involved several processes of the nuclear enrichment cycle, played a significant role in the U.S. energy program and the U.S. Department of Defense nuclear fuel program. These processes included nuclear fuel enrichment; radiochemical separations; refining, finishing, and storing uranium; and handling the associated radioactive waste.

PORTS workers, especially those employed during the production decades of the 1950s and 1960s, have been exposed to radiation types, absorption types, different enrichments and radionuclide matrixes and energies associated with nuclear energy development processes. PORTS used facility and individual worker monitoring methods to measure and control radiation exposures. Evaluations are difficult because the extensive scope of facility, process, and worker information relevant to an individual worker's potential dose might involve many years or even decades after employment.

The Portsmouth site internal dosimetry program started when the site initiated operation in 1954. Uranium isotopes were of primary concern. The primary method for monitoring of employees for radionuclide intake was urine bioassay. Fluorimetry for elemental uranium analysis and gas flow proportional counting for gross alpha counting was the cornerstone of the monitoring methodology for PORTS until very recent times. Technetium-99 (^{99}Tc) routine monitoring began in 1965 *in vivo* and 1975 *in vitro*. Transuranics have not been monitored routinely with the exception of Neptunium-237 (^{237}Np).

The radionuclides of concern for internal dosimetry include the following elements and isotopes:

- ^{237}Np
- ^{99}Tc
- Uranium (^{234}U, ^{235}U, ^{236}U, ^{238}U and mixtures) depleted uranium (DU), highly enriched uranium (HEU), enriched uranium (EU), natural uranium (Nat U)
- Plutonium (^{238}Pu, ^{239}Pu, ^{240}Pu)
- Thorium (^{228}Th, ^{230}Th, ^{231}Th, ^{232}Th, ^{234}Th)
- Protactinium (^{234m}Pa)
- Americium (^{241}Am)

These radionuclides are those identified by PORTS internal dosimetry program and the internal dose technical basis document (Hill and Strom, 1993). Only uranium, ^{99}Tc and ^{237}Np were monitored routinely.

5.1   *IN VITRO* MINIMUM DETECTABLE ACTIVITIES (MDAS) AND COUNTING METHODS

An example of probable decision level determination for urinalysis follows. These equations were used since about 1992.

$$D_L = \frac{\frac{4.65}{2} \times (C_{Background})^{1/2} + 3}{TREVA}$$ (Equation 5-1)

Where

D_L = decision level (dpm/L)
$C_{Background}$ = background counts in the region of interest
T = count time (min)
R = recovery fraction
E = average detector efficiency
V = sample volume (L), and
A = the alpha abundance for the radionuclide in question

The decision level is the level at which activity is considered present in a sample with a 95% confidence level.

$$MDA = \frac{4.65 \times (C_{Background})^{1/2} + 3}{TREVA}$$ (Equation 5-2)

5.1.1   *In Vitro* Urinalysis Records

The analysis records have been stored in several data electronic bases over PORTS site history. The databases include baseline (new hire), routine, special, recall and termination measurements. See section 5.8 for a description of database codes. The current database contains all of the information from the previous databases from 1954 to the present. The MDAs for the isotopes analyzed over the time frames are listed in Table 5.1.1-1. Activity fractions for uranium isotopes are listed in Table 5.1.1-2.

MDA values, flags or investigation levels (ILs) were used to determine the recorded values. The recall counts were counted for the same amount of time as the other counts. No change in MDA values would have occurred for these counts.

Table 5.1.1-1. *In Vitro* Measurement Frequencies, Measurement Types and MDCs for Urinalysis

Period	Frequency	Radionuclide	MDC	Record Level	Urinalysis Method	Volume
1954-3/31/1995	*Weekly to monthly*, weekly for suspected exposures > IL, monthly or bi monthly routine by work loc, function, etc.	Uranium Total	0.005 mg/L	0.01 mg/L	Fluorimetry.	Spot or 24-hr Simulated
1954-3/31/1995	*Weekly to monthly*, weekly for suspected exposures > IL, monthly or bi monthly routine by work loc, function etc.	Alpha Total	10 dpm/L	10 dpm/L	Ion Exchange/ Proportional Counter	Spot or 24-hr Simulated
4/1/95-Present	*Weekly to monthly* Did not measure total alpha anymore. ICP/MS replaced IC/Prop as well. Performed	U235, U238	0.047 μg/L 0.300 μg/L 0.1 pCi/L (all RNs)		ICP/MS	24-hr Simulated

Period	Frequency	Radionuclide	MDC	Record Level	Urinalysis Method	Volume
	Bechtel Jacobs Corporation (BJ) personnel until about 1998 when BJ decided to send samples to Wash or Oak Ridge National Laboratory (ORNL).					
10/0/98-6/30/01	*Weekly to monthly* Quanterra/BJ employees 24-hr sample.	Uranium Total	0.06 μg/L		Kinetic Phosphorescence Analysis (KPA)	24-hour Simulated
7/1/01-Present	*Weekly to monthly* ORNL/BJ employees 24-hour sample.	U234, U235 & U238	0.02 dpm/L, 0.02 dpm/L, 0.02 dpm/L		Alpha Spectrometry	24-hour Simulated
1/1/1994-Present	*Weekly to monthly* BJ & United States Enrichment Corporation (USEC).	^{237}Np	0.02 dpm/1 liter ORNL		Alpha Spectroscopy	24-hour Simulated
1/1/1994-Present	*Weekly to monthly* BJ & USEC.	^{241}Am	0.02 dpm/1 liter ORNL		Alpha spectroscopy	24-hour Simulated
1/1/1994-Present	*Weekly to monthly* BJ & USEC.	^{239}Pu	0.02 dpm/1 liter ORNL		Alpha spectroscopy	24-hour Simulated
1/1/1994-Present	*Weekly to monthly* BJ & USEC.	^{231}Th	8000 dpm/1 liter		Gas flow Proportional & LSC	24-hour Simulated
1/1/1994-Present	*Weekly to monthly* BJ & USEC.	^{234m}Pa	8000 dpm/1 liter		Gas flow Proportional & LSC	24-hour Simulated

Source: BJC, 1999; Mayfield, 1995; Hill and Strom, 1993; GAT, 1979; GAT, 1966a; GAT, 1956.

Table 5.1.1-2. Uranium Source Term Information

Uranium Source Term	Reference	Specific Activity pCi/µg	Activity Fractions			
			^{234}U	^{235}U	^{236}U	^{238}U
Natural Uranium	IMBA[a]	0.683	0.489	0.023	-	0.489
93.%	IMBA[a]	68.1	0.968	0.030	0.002	0.0003
3.5% Typical PORTS	IMBA[a]	2.20	0.818	0.034	-	0.147
2%	HPS[b]	1.20	0.648	0.041	0.0009	0.311
Typical DU	IMBA[a]	0.402	0.155	0.011	0.0005	0.834
Recycled Uranium	Hanford[c]	0.910	0.563	0.023	0.048	0.365
Uranium Source Term	**Reference**	**Specific Activity pCi/µg**	Specific Constituent Activity in Mixture (uCi/g, nCi/mg, or pCi/ug)			
Natural Uranium	IMBA[a]	0.683	0.334	0.016	-	0.334
93.%	IMBA[a]	68.1	65.9	2.04	0.136	0.020
3.5% Typical PORTS	IMBA[a]	2.20	1.80	0.075	-	0.323
2%	HPS[b]	1.20	0.778	0.049	0.001	0.373
Typical DU	IMBA[a]	0.402	0.062	0.004	0.0002	0.335
Recycled Uranium	Hanford[c]	0.910	0.563	0.023	0.048	0.365

a. IMBA computer software, and World Information Service on Energy (WISE) Uranium Project.
b. ANSI HPS N13.22-1995, *Bioassay Programs for Uranium,* (An American National Standard), October, 1995.
c. Hanford TBD Table 5.2.5-1 ORAUT-TKBS-0006-5.

5.1.2   *In Vitro* Methods for Individual Radionuclides

5.1.2.1   In Vitro Bioassay for uranium

Urinalysis for total alpha activity has been conducted since PORTS began operation in 1954. Urine samples were evaporated to dryness with an excess of concentrated nitric acid and ignited over a blast burner. The white salt residue was dissolved in distilled water and a double calcium precipitation was made with 0.4 M ammonium oxalate. The uranium was electroplated onto nickel disks from the combined supernates. The plated disks were ignited and the alpha activity was measured on a proportional counter in α counts/min / 100 ml urine {1 dpm/100 ml minimum detectable concentration (MDC)}. 100 ml of urine was needed for this technique. A result of 5 dpm/100 ml was considered the reporting level for the time-period of 1954–1993 (GAT, 1955; GAT, 1985a; Hill and Strom, 1993). Review of claim records reveals that sometimes values less than 5 dpm were recorded, probably down to the MDC of 1 dpm/100 ml.

To determine total uranium mass per unit volume a fluorimetry procedure was used. A fusion of uranium salts with sodium fluoride gives a characteristic yellow green fluorescence under UV light. The intensity of the fluorescence is proportional to the concentration of the uranium in the fused disk. The fluorescence is measured with a fluorimeter that has been calibrated with a series of standard uranium disks. Results exceeding 0.01 mg U/liter were re-run. This 0.01 mg/liter was considered a positive result from the time-period of 1954–1993 (GAT, 1956; Hill and Strom, 1993).

The calculation was as follows:

$$\frac{\text{Voltmeter reading x fluor scale x 1000}}{5 \times 0.2 \times 1000} = \text{mg uranium/ liter} \qquad \text{(Equation 5-3)}$$

5.1.2.2   Inductively Coupled Plasma Mass Spectrometry (ICP/MS)

As of 3/31/95, ICP/MS is being utilized for the analysis of ^{235}U and ^{238}U in urine. Urine is digested and wet oxidized with strong nitric and hydrochloric acids to solubilize uranium and to destroy the organic matter. Uranium is selectively separated from the chloride salts by an anion exchange resin and is extracted with dilute nitric acid. The uranium isotopes are then measured by ICP/MS.

The MDA for this process is 0.1 pCi/liter. The method can analyze directly for ^{235}U and ^{238}U. ^{234}U cannot be directly measured because of the 1 amu resolution of ICP/MS but can be estimated based upon knowledge of the enrichment. A 0.2 μg, standard with 3.0% by weight ^{235}U is counted daily. Typical results from ICP/MS are a 3.2% enrichment determination with a 1.8% relative standard deviation (Mayfield, 1995).

5.1.2.3   Frequency of Urine Bioassays

The frequency of urine bioassays could vary considerably at PORTS. As stated in one version of the urine bioassay procedure (circa 1971): "All individuals who may come in contact with toxic materials are placed on a routine urinary program and scheduled on a 1 week, 1 month, 3 month or 6 month frequency depending on the following" (shown below). A number of categories of urinalysis include routine, recall and a supervision requested analysis. Exact criteria were not established. It was based upon the coordinated judgment of the employee's supervision and the Industrial Hygiene and Health Physics Departments (GAT, 1966a, p2).

Type I - Routine

    A. Potential Exposure
        (a) High Daily – 1 week
        (b) Daily – 1 month
        (c) Weekly – 3 months
        (d) Occasional – 6 months

    B. Concentration involved
        (a) Planned and Unplanned High Airborne – 1 month
        (b) High (or high assay) – 1 month
        (c) Moderate – 3 months
        (d) Low – 6 months

    C. Past Medical History
        (a) Several Positive Exposures – 1 month
        (b) Few Exposures – 3 months
        (c) Infrequent or None – 6 months (GAT, 1966a, p3)

Type II - Recall

Recall samples were obtained to verify negative results, to replace a sample that was misplaced, spilled, or lost due to laboratory error, to replace an inadequate sample or whenever results indicate values above the recall limits:

Table 5.1.2.3-1. Recall Limits

Recall in	Uranium (mg/l)	Alpha (dpm/100 ml)
1 month	0.01	5.0
1 week	0.02	9.0
Immediately	0.06	30.0

    Source: GAT, 1966a, p4.

Type III – Supervision's request

Any time an individual is exposed to high concentrations of materials such as purge gas release, etc., a special urine sample should be submitted.

5.1.2.4   Work Restriction

Whenever a single sample or a series of samples indicate positive results equaling 0.3 mg/liter or 45 dpm/100 ml followed by a Monday sample of 0.06 mg U/l or 9 dpm/100 ml, the individual was considered for a work restriction. All individuals were then asked to submit daily samples until urinary values fell below 0.06 mg U/l and 9.0 dpm/100ml for two consecutive Mondays (GAT, 1966a, p 9-10).

A new list of personnel required to receive a urinalysis was compiled every January. Routine samples were submitted on Monday of every week and recorded on form A-551. A special sample was given 4 hours after exposure and one voiding for suspected inhalation incidents. Total uranium analysis required 2 ml and a total alpha analysis required a 100-ml volume. Until around 1995 spot samples were the norm at PORTS. One out of ten samples were controls.

5.1.2.5 Uranium MDCs or Reporting Levels

There has been a variety of methods used historically to analyze for uranium at PORTS. These methods and their associated detection capabilities are summarized in Table 5.1.1-1.

5.1.2.6 Source Term and Isotopic Determination

In order to use either the total alpha or elemental uranium results, use Table 5.1.1-2 to determine the actual radionuclide isotopic values. For gaseous diffusion enriched uranium, the approximate specific activity (SA) of a given enrichment over a 1-50% range is (DOE, 2000c):

Specific Activity of Enriched Uranium = $(0.4 + 0.38E + 0.0034E^2) \times 10^{-6}$ Ci/g $\qquad$ (Equation 5-4)

Where E = % ^{235}U by weight enrichment > or = 0.72

HEU production at PORTS ended in 1978. However, reintroduction of HEU from various sources to the cascade occurred up to 1998 (see Table 5.1.2.6-1 for further information). The increase in SA is mostly due to ^{234}U not ^{238}U replacement with ^{235}U. The above equation is not valid for recycled uranium (RU). (DOE, 2000c).

Table 5.1.2.6-1. Reactor Returns, RU or Tails Fed to the Cascade

Fiscal Year	Amount fed (MTU)	Enrichment (%^{235}U)	Source	Remarks
1955	105.8	0.64 – 0.68	Paducah	May – Sept. 1955
1956	54.5	0.64 – 0.68	Paducah	
1956	293.4	0.64 – 0.68	Oak Ridge	
1957	6.2	0.64 – 0.68	Paducah	
1958	64.2	0.64 – 0.68	Paducah	
1970	168.1	0.64 – 0.68	Paducah	Oct. & Nov. 1969
1974	398.8	0.64 – 0.68	Paducah	Jan. 1974
1974 – 1978	1.86	2 – 50	PORTS Oxide Conversion	
1968 – 1977	0.15	78 – 80	Division of International Affairs	
1977 – 1998	0.15	78 – 97	Babcock & Wilcox	
1969 – 1993	0.07	78	Atomic Energy Commission (AEC) Office of Safeguards & Materials Management	
1997 – 1998	1.10	56 – 82	France	

Source: Table 2.2.2.5-1 (BJC, 2000, p. 22).

PORTS mission was to enrich uranium in the form of UF_6 for use in domestic and foreign commercial power reactors from slightly enriched uranium of roughly 2.5% ^{235}U to about 5% enrichment of ^{235}U. Up until about 1992 PORTS enriched uranium to about 93% enrichment of ^{235}U. PORTS also accepted used or recycled uranium from domestic and foreign sources for enrichment. Compounds of uranium included UO_2, UO_3, U_3O_8, UF_4 and UF_6. The primary nuclides of concern for the plant are ^{238}U, ^{235}U and ^{234}U. The progeny of domestic interest for these radionuclides includes ^{230}Th and ^{234m}Pa. The major facilities where uranium was processed are listed in Table 5.1.2.6-2.

Table 5.1.2.6-2. Major Uranium Facilities at PORTS

Building No.	Name	Dates of operation	Activities
X-326	Gaseous Diffusion Process Bldg.	1954–1991	High Assay Product
X-330	Gaseous Diffusion Process Bldg.	1954–2001	Intermediate Process & Tails Withdrawal
X-333	Gaseous Diffusion Process Bldg.	1954–2001	Initial Enrichment & Reactor Product
X-342A	Fixed Feed Facility	1954–2001	Feed UF_6 to Process Line
X-342	Fluorine Generation Facility	1954–2001	Generate Elemental F_2 for Converters
X-343	Fixed Feed Facility	1954–2001	Feed UF_6 to Process Line
X-344	UF_6 Feed Manufacturing Plant	1958–1962	Conversion of UF_4 to UF_6
X-345	Special Nuclear Material Storage	1978–2003	Highly Enriched Uranium Storage
X-700	Maintenance Building	1954–2003	Large Component Repairs
X-705	Decontamination & Cleaning Bldg.	1954–2003	Equipment Wash & Uranium Recovery
X-705E	Oxide Conversion Plant	1957–1978	Convert U_3O_8 to UF_6
X-710	Labs, Electrical and I&C Shops	1954–2003	Testing, Calibration, & Repair
X-720	Compressor Shop	1954–2003	Disassembly & Repair of Compressors
X-744G	Smelter & Aluminum Recovery	1954–1978	Recover Aluminum from Scrap

Source: DOE, 2000b, p. 16.

Because of excretion of uranium from natural sources, the MDAs for natural or depleted uranium listed above should not be used to determine occupational intakes. Based on the Savanah River TBD report and an upper limit at 99% confidence, a 0.15 dpm/L DL was set based on analytical noise and background. (ORAUT-TKBS-0003-05, rev.1)

Hanford studies from 1985 and 1990 indicate a uranium urinary excretion of 0.05 to 0.5 μg/day in the Hanford area. The 1985 study (Sula et al, 1991) indicated that at the 99.9th percentile the daily uranium output was 0.2 μg/day (ORAUT-TKBS-0006-05, rev.1, p70). Any result greater than this value was considered to contain occupationally derived uranium. PORTS/Paducah Gaseous Diffusion Plant (PGDP) adopted this value according to the 1993 Internal Dosimetry PORTS/PDGP TBD (Hill and Strom, 1993). Uranium source term or isotopic breakdown information is presented in Table 5.1.1-2.

Because studies of the average daily uranium excretion on Portsmouth residents do not appear to be performed, it is not possible to make corrections for the contribution of nonoccupational intakes of uranium to a given urine sample result. However, to put a given result into perspective, a nominal activity of 0.43 μg (environmental decision level at 95% confidence) can be used (BJC, 1999). No correction for environmental levels of uranium is required for samples analyzed by fluorimetry, KPA, or total alpha because the MDAs were larger than the correction.

No record exists of attempts to characterize the size of particles at PORTS. For dose reconstruction purposes the default 5-μm AMAD particle size should be used.

Certain transuranic (TRU) radionuclides are present at PORTS including ^{239}Pu and ^{237}Np. These come from reactor tails or RU that were processed at PORTS. Refer to Table 5.1.2.6-1 "Reactor returns, RU or tails fed to the cascade". Only about 2% of the total feed was reactor tails. Table 5.1.2.6-3 gives the TRU and ^{99}Tc activities in dpm per gram of reactor tail uranium per year. This data is based upon the PORTS mass balance report (BJC, 2000). Note the information does not break down the TRU or ^{99}Tc information by facility or building. This information is presented in Table 5.1.2.6-4, "activity fraction" column and is based on air sampling results from 1992–1998, and thus is only a snap shot of the total time period of the PORTS operations.

Table 5.1.2.6-3. Uranium TRU and ^{99}Tc Source Term Information

Year	^{237}Np dpm/gRU	^{99}Tc dpm/gRU	^{238}Pu dpm/gRU	^{239}Pu dpm/gRU	^{240}Pu dpm/gRU
1955	1.20E+02	5.16E+01	0.00E+00	0.00E+00	0.00E+00
1956	5.54E+01	2.08E+01	0.00E+00	0.00E+00	0.00E+00
1957	1.08E+02	3.38E+01	0.00E+00	0.00E+00	0.00E+00
1958	9.52E+01	4.16E+01	0.00E+00	0.00E+00	0.00E+00
1959	9.52E+01	5.35E+01	0.00E+00	0.00E+00	0.00E+00
1960	9.52E+01	5.95E+01	0.00E+00	0.00E+00	0.00E+00
1961	9.52E+01	6.54E+01	0.00E+00	0.00E+00	0.00E+00
1962	9.52E+01	7.14E+01	0.00E+00	0.00E+00	0.00E+00
1963	9.52E+01	8.33E+01	0.00E+00	0.00E+00	0.00E+00
1964	9.52E+01	8.92E+01	0.00E+00	0.00E+00	0.00E+00
1965	9.52E+01	9.52E+01	0.00E+00	0.00E+00	0.00E+00
1966	9.52E+01	1.01E+02	5.95E-02	1.77E-01	5.26E-01
1967	9.52E+01	1.07E+02	5.95E-02	1.77E-01	5.26E-01
1968	1.04E+02	1.13E+02	5.95E-02	1.77E-01	5.25E-01
1969	1.09E+02	6.31E+01	5.74E-02	8.24E-02	1.18E-01
1970	1.09E+02	6.60E+01	5.74E-02	8.24E-02	1.18E-01
1971	1.09E+02	6.74E+01	8.61E-02	1.24E-01	1.77E-01
1972	1.09E+02	6.89E+01	8.61E-02	1.23E-01	1.77E-01
1973	1.09E+02	7.45E+01	8.59E-02	1.23E-01	1.76E-01
1974	6.44E+01	9.16E+01	1.00E-01	1.44E-01	2.06E-01
1975	6.58E+01	9.30E+01	1.00E-01	1.43E-01	2.05E-01
1976	6.76E+01	9.02E+01	3.24E-01	4.57E-01	6.44E-01
1977	6.90E+01	8.45E+01	3.24E-01	4.56E-01	6.43E-01
1978	7.04E+01	8.17E+01	3.24E-01	4.56E-01	6.43E-01
1979	7.04E+01	8.03E+01	3.24E-01	4.56E-01	6.43E-01
1980	7.04E+01	7.89E+01	3.24E-01	4.56E-01	6.43E-01
1981	7.04E+01	7.75E+01	3.24E-01	4.56E-01	6.43E-01
1982	7.04E+01	7.61E+01	3.24E-01	4.56E-01	6.43E-01
1983	7.04E+01	7.46E+01	2.25E-01	3.17E-01	4.47E-01
1984	6.90E+01	7.32E+01	2.25E-01	3.17E-01	4.47E-01
1985	6.90E+01	6.90E+01	2.25E-01	3.17E-01	4.47E-01
1986	6.90E+01	6.76E+01	1.83E-01	2.58E-01	3.63E-01
1987	6.90E+01	6.62E+01	1.83E-01	2.58E-01	3.63E-01
1988	6.90E+01	6.48E+01	1.83E-01	2.58E-01	3.63E-01
1989	6.90E+01	6.34E+01	1.83E-01	2.58E-01	3.63E-01
1990	6.90E+01	6.34E+01	1.83E-01	2.58E-01	3.63E-01
1991	6.90E+01	6.34E+01	1.83E-01	2.58E-01	3.63E-01
1992	6.90E+01	6.20E+01	1.83E-01	2.58E-01	3.63E-01
1993	6.90E+01	6.20E+01	1.83E-01	2.58E-01	3.63E-01
1994	6.90E+01	6.20E+01	1.83E-01	2.58E-01	3.63E-01
1995	6.76E+01	6.06E+01	1.83E-01	2.58E-01	3.63E-01
1996	6.62E+01	6.06E+01	1.69E-01	2.38E-01	3.35E-01
1997	6.48E+01	4.93E+01	1.69E-01	2.38E-01	3.35E-01
1998	6.48E+01	4.79E+01	1.41E-01	1.98E-01	2.79E-01

Based upon data from BJC, 2000, Table 3.2-1 and Figure 5.1-1.

If specific source term information to which the employee has been exposed is available, the dose reconstructor should utilize that information. However, if no source term information is available, the values and parameters in Table 5.1.2.6-4 provide an adequate input to the process. Note that the

activity fraction and contaminant ratio to uranium columns are based upon 1993 through 1999 PORTS air sampling data and is therefore a snapshot of the temporal radionuclide matrix of the facilities.

If isotopic data are available, it can be used to determine if a uranium urinalysis result represents an occupational uptake. The following rules apply:

- If ^{235}U or ^{236}U are detected, then ^{234}U must also be detected.

- If both ^{234}U and ^{238}U are detected, the uranium should be considered from occupational sources if the $^{234}U/^{238}U$ ratio is outside a range from 1 to 3, regardless of the total uranium excretion rate.

- If ^{238}U is detected, the uranium should be considered from an occupational source if the ratio of the ^{234}U critical level to the ^{238}U result is less than 1 and the total uranium excretion is greater than 0.43 µg/d.

- If ^{234}U is detected, the uranium should be considered from an occupational source if the ratio of the ^{234}U result to the ^{238}U critical level is greater than 3 (ORAUT-TKBS-0003-05, rev.1, p71).

Table 5.1.2.6-4. Lung Absorption Type and Fractional Activity of Radionuclides by Facility

Facility	Compound	RN	Suggested Absorption Type[d]	Absorption Type Range	Activity Fraction[f,h]	Intake Activity Fraction Relative to Total Uranium[j]
X-326	UF₆, U₃O₈, and uranyl fluoride (UO₂F₂) are likely (Hill and Strom 1993, p14.7).	$^{234}U^{a}$	F	F-S	9.12×10^{-1}	
		$^{235}U^{a}$	F	F-S	3.16×10^{-2}	
		$^{236}U^{a}$	F	F-S	2.64×10^{-3}	
		$^{238}U^{a}$	F	F-S	5.33×10^{-2}	
		$^{99}Tc^{g}$	F	F-M	6.43×10^{-3}	6.43×10^{-3}
		^{237}Np	M	M	9.58×10^{-5}	9.58×10^{-5}
		$^{238}Pu^{b}$	S	M-S	1.79×10^{-5}	
		$^{239}Pu^{b}$	S	M-S	1.79×10^{-5}	5.37×10^{-5}
		$^{240}Pu^{b}$	S	M-S	1.79×10^{-5}	
		$^{241}Am^{b}$	S	M-S	1.79×10^{-5}	1.79×10^{-5}
		$^{228}Th^{i}$	S	F-S	1.15×10^{-4}	1.15×10^{-4}
		^{230}Th	S	F-S	3.47×10^{-4}	3.47×10^{-4}
		^{231}Th	S	F-S	Trace	
		$^{232}Th^{i}$	S	F-S	3.6×10^{-6}	3.6×10^{-6}
		$^{234}Th^{c}$	M	F-S	Trace	
		$^{2134m}Pa^{c}$	M	M-S	(In equilibrium with ^{234}Th)	
X-330	UF₆, U₃O₈, and uranyl fluoride (UO₂F₂) are likely (High and Strom 1993, p14.7).	$^{234}U^{a}$	F	F-S	7.44×10^{-1}	
		$^{235}U^{a}$	F	F-S	3.33×10^{-2}	
		$^{236}U^{a}$	F	F-S	2.46×10^{-3}	
		$^{238}U^{a}$	F	F-S	2.20×10^{-1}	
		$^{99}Tc^{g}$	F	F-M	3.70×10^{-3}	3.70×10^{-3}
		^{237}Np	M	M	8.22×10^{-5}	8.22×10^{-5}
		$^{238}Pu^{b}$	S	M-S	1.03×10^{-5}	
		$^{239}Pu^{b}$	S	M-S	1.03×10^{-5}	3.09×10^{-5}
		$^{240}Pu^{b}$	S	M-S	1.03×10^{-5}	
		$^{241}Am^{b}$	S	M-S	1.03×10^{-5}	1.03×10^{-5}
		$^{228}Th^{i}$	S	F-S	1.51×10^{-5}	1.51×10^{-5}
		^{230}Th	S	F-S	8.36×10^{-5}	8.36×10^{-5}
		^{231}Th	S	F-S	Trace	
		$^{232}Th^{i}$	S	F-S	1.66×10^{-5}	1.66×10^{-5}
		$^{234}Th^{c}$	M	F-S	Trace	
		$^{234m}Pa^{c}$	M	M-S	(In equilibrium with ^{234}Th)	
X-333	UF₆, U₃O₈, and uranyl fluoride (UO₂	$^{234}U^{a}$	F	F-S	5.03×10^{-1}	
		$^{235}U^{a}$	F	F-S	2.83×10^{-2}	
	likely (High and	$^{236}U^{a}$	F	F-S	1.74×10^{-3}	

Facility	Compound	RN	Suggested Absorption Type[d]	Absorption Type Range	Activity Fraction[f,h]	Intake Activity Fraction Relative to Total Uranium[j]
	Strom 1993, p14.7).	$^{238}U^a$	F	F-S	4.67×10^{-1}	
		$^{99}Tc^g$	F	F-M	1.54×10^{-2}	1.54×10^{-2}
		^{237}Np	M	M	6.81×10^{-5}	6.81×10^{-5}
		$^{238}Pu^b$	S	M-S	4.29×10^{-5}	
		$^{239}Pu^b$	S	M-S	4.29×10^{-5}	1.29×10^{-4}
		$^{240}Pu^b$	S	M-S	4.29×10^{-5}	
		$^{241}Am^b$	S	M-S	4.29×10^{-5}	4.29×10^{-5}
		$^{228}Th^i$	S	F-S	4.42×10^{-5}	4.42×10^{-5}
		^{230}Th	S	F-S	8.23×10^{-5}	8.23×10^{-5}
		^{231}Th	S	F-S	Trace	
		$^{232}Th^i$	S	F-S	1.04×10^{-5}	1.04×10^{-5}
		$^{234}Th^c$	M	F-S	Trace	
		$^{234m}Pa^c$	M	M-S	(In equilibrium with ^{234}Th)	
X-342A and X-342	UF_6 (High and Strom, 1993, p14.7).	$^{234}U^a$	F	F-S	8.46×10^{-1}	
		$^{235}U^a$	F	F-S	3.48×10^{-2}	
		$^{236}U^a$	F	F-S	4.69×10^{-3}	
		$^{238}U^a$	F	F-S	1.14×10^{-1}	
		$^{99}Tc^g$	F	F-M	4.77×10^{-3}	4.77×10^{-3}
		^{237}Np	M	M-S	3.35×10^{-4}	3.35×10^{-4}
		$^{238}Pu^b$	S	M-S	1.33×10^{-5}	
		$^{239}Pu^b$	S	M-S	1.33×10^{-5}	3.99×10^{-5}
		$^{240}Pu^b$	S	M-S	1.33×10^{-5}	
		$^{241}Am^b$	S	M-S	1.33×10^{-5}	1.33×10^{-5}
		$^{228}Th^i$	S	F-S	1.05×10^{-4}	1.05×10^{-4}
		^{230}Th	S	F-S	1.06×10^{-4}	1.06×10^{-4}
		^{231}Th	S	F-S	Trace	
		$^{232}Th^i$	S	F-S	Trace	
		$^{234}Th^c$	M	F-S	Trace	
		$^{234m}Pa^c$	M	M-S	(In equilibrium with ^{234}Th)	(In equilibrium with ^{234}Th)
X-343	UF_6 (Hill and Strom, 1993, p14.7).	$^{234}U^a$	F	F-S	8.06×10^{-1}	
		$^{235}U^a$	F	F-S	3.45×10^{-2}	
		$^{236}U^a$	F	F-S	2.31×10^{-3}	
		$^{238}U^a$	F	F-S	1.57×10^{-1}	
		$^{99}Tc^g$	F	F-M	2.25×10^{-2}	2.25×10^{-2}
		^{237}Np	M	M	8.78×10^{-5}	8.78×10^{-5}
		$^{238}Pu^b$	S	M-S	6.26×10^{-5}	
		$^{239}Pu^b$	S	M-S	6.26×10^{-5}	1.88×10^{-4}
		$^{240}Pu^b$	S	M-S	6.26×10^{-5}	
		$^{241}Am^b$	S	M-S	6.26×10^{-5}	6.26×10^{-5}
		$^{228}Th^i$	S	F-S	2.49×10^{-5}	2.49×10^{-5}
		^{230}Th	S	F-S	1.95×10^{-4}	1.95×10^{-4}
		^{231}Th	S	F-S	Trace	
		$^{232}Th^i$	S	F-S	Trace	
		$^{234}Th^c$	M	F-S	Trace	
		$^{234m}Pa^c$	M	M-S	(In equilibrium with ^{234}Th)	
X-344	UF_6 (High and Strom 1993, p14.7) 1958-1962 UF_6, U_3O_8, and UO_2F_2 1962-2001.	$^{234}U^e$	S	F-S	7.44×10^{-1}	
		$^{235}U^e$	S	F-S	3.20×10^{-2}	
		$^{236}U^e$	S	F-S	4.9×10^{-4}	
		$^{238}U^e$	S	F-S	2.23×10^{-1}	
		$^{99}Tc^g$	F	F-M	1.25×10^{-3}	1.25×10^{-3}
		^{237}Np	M	M	2.37×10^{-5}	2.37×10^{-5}
		$^{238}Pu^b$	S	M-S	3.48×10^{-6}	
		$^{239}Pu^b$	S	M-S	3.48×10^{-6}	1.04×10^{-5}
		$^{240}Pu^b$	S	M-S	3.48×10^{-6}	
		$^{241}Am^b$	S	M-S	3.48×10^{-6}	3.48×10^{-6}
		$^{228}Th^i$	S	F-S	9.95×10^{-6}	9.95×10^{-6}
		^{230}Th	S	F-S	1.42×10^{-4}	1.42×10^{-4}
		^{231}Th	S	F-S	Trace	
		$^{232}Th^i$	S	F-S	Trace	

Facility	Compound	RN	Suggested Absorption Type[d]	Absorption Type Range	Activity Fraction[f,h]	Intake Activity Fraction Relative to Total Uranium[j]
		^{234}Th[c]	M	F-S	Trace	
		^{234m}Pa[c]	M	M-S	(In equilibrium with ^{234}Th)	
X-345	UF$_6$, UO$_3$, and UO$_2$F$_2$.	^{234}U[a]	F	F-S	9.78×10^{-1}	
		^{235}U[a]	F	F-S	1.68×10^{-2}	
		^{236}U[a]	F	F-S	1.19×10^{-3}	
		^{238}U[a]	F	F-S	2.16×10^{-3}	
		^{99}Tc[g]	F	F-M	Trace	
		^{237}Np	M	M	2.04×10^{-3}	2.04×10^{-3}
		^{238}Pu[b]	S	M-S	Trace	
		^{239}Pu[b]	S	M-S	Trace	
		^{240}Pu[b]	S	M-S	Trace	
		^{241}Am[b]	S	M-S	Trace	
		^{228}Th[i]	S	F-S	Trace	
		^{230}Th	S	F-S	1.2×10^{-1}	1.2×10^{-1}
		^{231}Th	S	F-S	Trace	
		^{232}Th[i]	S	F-S	Trace	
		^{234}Th[c]	M	F-S	Trace	
		^{234m}Pa[c]	M	M-S	(In equilibrium with ^{234}Th)	
X-700	UF$_6$, U$_3$O$_8$, and UO$_2$F$_2$.	^{234}U[e]	S	F-S	5.97×10^{-1}	
		^{235}U[e]	S	F-S	3.14×10^{-2}	
		^{236}U[e]	S	F-S	2.55×10^{-3}	
		^{238}U[e]	S	F-S	3.64×10^{-1}	
		^{99}Tc[g]	F	F-M	1.88×10^{-2}	1.88×10^{-2}
		^{237}Np	M	M	1.37×10^{-3}	1.37×10^{-3}
		^{238}Pu[b]	S	M-S	5.24×10^{-5}	
		^{239}Pu[b]	S	M-S	5.24×10^{-5}	1.57×10^{-4}
		^{240}Pu[b]	S	M-S	5.24×10^{-5}	
		^{241}Am[b]	S	M-S	5.24×10^{-5}	5.24×10^{-5}
		^{228}Th[i]	S	F-S	3.74×10^{-4}	3.74×10^{-4}
		^{230}Th	S	F-S	3.17×10^{-3}	3.17×10^{-3}
		^{231}Th	S	F-S	Trace	
		^{232}Th[i]	S	F-S	3.23×10^{-4}	3.23×10^{-4}
		^{234}Th[c]	M	F-S	Trace	
		^{234m}Pa[c]	M	M-S	(In equilibrium with ^{234}Th)	
X-705 & X-705E + H	UF$_6$, U$_3$O$_8$, and UO$_2$F$_2$ Includes MgF$_2$ traps, TRUs mostly in tower ash (705E 1957-1978).	^{234}U[e]	S	F-S	8.87×10^{-1}	
		^{235}U[e]	S	F-S	3.43×10^{-2}	
		^{236}U[e]	S	F-S	1.86×10^{-2}	
		^{238}U[e]	S	F-S	5.88×10^{-2}	
		^{99}Tc[g]	F	F-M	3.16×10^{-2}	3.16×10^{-2}
		^{237}Np	M	M	1.60×10^{-4}	1.60×10^{-4}
		^{238}Pu[b]	S	M-S	8.79×10^{-5}	
		^{239}Pu[b]	S	M-S	8.79×10^{-5}	2.64×10^{-4}
		^{240}Pu[b]	S	M-S	8.79×10^{-5}	
		^{241}Am[b]	S	M-S	8.79×10^{-5}	8.79×10^{-5}
		^{228}Th[i]	S	F-S	1.30×10^{-4}	1.30×10^{-4}
		^{230}Th	S	F-S	6.66×10^{-4}	6.66×10^{-4}
		^{231}Th	S	F-S	Trace	
		^{232}Th[i]	S	F-S	3.37×10^{-5}	3.37×10^{-5}
		^{234}Th[c]	M	F-S	Trace	
		^{234m}Pa[c]	M	M-S	(In equilibrium with ^{234}Th)	
X-710	UF$_6$, UO$_3$, and UO$_2$F$_2$.	^{234}U[a]	F	F-S	9.18×10^{-1}	
		^{235}U[a]	F	F-S	2.70×10^{-2}	
		^{236}U[a]	F	F-S	1.75×10^{-2}	
		^{238}U[a]	F	F-S	4.69×10^{-2}	
		^{99}Tc[g]	F	F-M	2.60×10^{-1}	2.60×10^{-1}
		^{237}Np	M	M	1.80×10^{-3}	1.80×10^{-3}
		^{238}Pu[b]	S	M-S	7.25×10^{-4}	
		^{239}Pu[b]	S	M-S	7.25×10^{-4}	2.18×10^{-3}

Facility	Compound	RN	Suggested Absorption Type[d]	Absorption Type Range	Activity Fraction[f,h]	Intake Activity Fraction Relative to Total Uranium[j]
		^{240}Pu[b]	S	M-S	7.25×10^{-4}	
		^{241}Am[b]	S	M-S	7.25×10^{-4}	7.25×10^{-4}
		^{228}Th[i]	S	F-S	3.49×10^{-4}	3.49×10^{-4}
		^{230}Th	S	F-S	3.94×10^{-3}	3.94×10^{-3}
		^{231}Th	S	F-S	Trace	
		^{232}Th[i]	S	F-S	4.61×10^{-5}	4.61×10^{-5}
		^{234}Th[c]	M	F-S	Trace	
		^{234m}Pa[c]	M	M-S	(In equilibrium with ^{234}Th)	
X-720	UF$_6$, UO$_3$, and UO$_2$F$_2$.	^{234}U[a]	F	F-S	7.77×10^{-1}	
		^{235}U[a]	F	F-S	3.45×10^{-2}	
		^{236}U[a]	F	F-S	8.9×10^{-4}	
		^{238}U[a]	F	F-S	1.86×10^{-1}	
		^{99}Tc[g]	F	F-M	1.11×10^{-1}	1.11×10^{-1}
		^{237}Np	M	M	Trace	
		^{238}Pu[b]	S	M-S	3.08×10^{-4}	
		^{239}Pu[b]	S	M-S	3.08×10^{-4}	9.24×10^{-4}
		^{240}Pu[b]	S	M-S	3.08×10^{-4}	
		^{241}Am[b]	S	M-S	3.08×10^{-4}	3.08×10^{-4}
		^{228}Th[i]	S	F-S	5.64×10^{-4}	5.64×10^{-4}
		^{230}Th	S	F-S	4.68×10^{-4}	4.68×10^{-4}
		^{231}Th	S	F-S	Trace	
		^{232}Th[i]	S	F-S	Trace	
		^{234}Th[c]	M	F-S	Trace	
		^{234m}Pa[c]	M	M-S	(In equilibrium with ^{234}Th)	
X-745G	UF$_6$, UO$_3$, and UO$_2$F$_2$.	^{234}U[a]	F	F-S	9.44×10^{-1}	
		^{235}U[a]	F	F-S	3.26×10^{-2}	
		^{236}U[a]	F	F-S	2.57×10^{-3}	
		^{238}U[a]	F	F-S	1.88×10^{-2}	
		^{99}Tc[g]	F	F-M	7.93×10^{-2}	7.93×10^{-2}
		^{237}Np	M	M	4.61×10^{-4}	4.61×10^{-4}
		^{238}Pu[b]	S	M-S	2.21×10^{-4}	
		^{239}Pu[b]	S	M-S	2.21×10^{-4}	6.63×10^{-4}
		^{240}Pu[b]	S	M-S	2.21×10^{-4}	
		^{241}Am[b]	S	M-S	2.21×10^{-4}	2.21×10^{-4}
		^{228}Th[i]	S	F-S	3.08×10^{-4}	3.08×10^{-4}
		^{230}Th	S	F-S	1.04×10^{-3}	1.04×10^{-3}
		^{231}Th	S	F-S	Trace	
		^{232}Th[i]	S	F-S	4.92×10^{-5}	4.92×10^{-5}
		^{234}Th[c]	M	F-S	Trace	
		^{234m}Pa[c]	M	M-S	(In equilibrium with ^{234}Th)	

a. UF$_6$ and UO$_2$F$_2$ are considered type F materials (Hill and Strom, 1993, p. 14.7). U$_3$O$_8$ is considered type S (Hill and Strom, 1993, p. 14.7). Most material should be UF$_6$ feed. Most bioassay results indicate that uranium acts as a type F material (Hill and Strom, 1993, p. 14.5).
b. Americium and plutonium not distinguished in analysis; use ^{239}Pu for the intake.
c. ^{234}Th/^{234m}Pa are found in equilibrium and are type M for GDPs.
d. Always use bioassay information to determine absorption class, when available.
e. Absorption type for this facility has indicated type S in many bioassay results.
f. Based upon air sampling data from 1993-1999 obtained by PORTS HP department
g. Based upon ^{99}Tc to ^{239}Pu ratio from Table 5.1.2.6-3.
h. Note some fractions may add to greater than 1 since the ^{99}Tc values (footnote g) were added to the air sampling data (footnote f).
i. ^{228}Th exceeds the value of ^{232}Th because averages are taken and the positive results for thorium analysis is near the LLD of the radionuclides.
j. Multiply the total uranium intake by this fraction to obtain the intake activity of the contaminant.

5.1.2.7   In Vitro Bioassay for Fission Products (^{99}Tc)

In vitro assessment of ^{99}Tc began in 1965, although until 1998 the analysis was not specific for ^{99}Tc. From at least 1965 to 1985, an addition of a liquid scintillator to the urine sample was used for the subsequent determination of total beta and alpha activity. Any urine sample greater than 1000 dpm/100 ml beta activity was re-analyzed to subtract any thorium/uranium complement. The remaining activity was reported as ^{99}Tc. There was at least one occurrence when ^{99}Tc was confused

with Thallium-201 (^{201}Tl) (GAT-S-52, 1986, p7. The Dose Reconstructor should use the ^{99}Tc to uranium ratios as indicated in Table 5.1.2.6-4 for determining intakes to ^{99}Tc.

Urinalysis for ^{99}Tc is the current bioassay method employed at PORTS. The stated ^{99}Tc bioassay MDA is 5,000 dpm L^{-1} (or about 80 Bq L^{-1}) (Hill and Strom, 1993).

As of 10/01/1998, Bechtel Jacobs (DOE) sent 24-hour samples to an outside lab yielding an MDA of 33 dpm/sample for Quanterra and 200 dpm/sample for ORNL.

5.1.2.8   *In Vitro* Bioassay for ^{237}Np

Although transuranics were suspected as a potential internal dosimetry issue routine monitoring was not conducted at PORTS. Alpha spectroscopy was performed on product analysis early in the site's history but the MDA of this system was considered too high for bioassay applications. In 1994, alpha spectroscopy was utilized for some suspect urine samples. A 0.5 pCi/liter MDC was reported. ICP/MS was available since 1995 but not calibrated for neptunium monitoring. In 1998, Bechtel Jacobs sent their samples offsite to ORNL or offsite vendors. ORNL had a 0.02 dpm/sample MDA for ^{237}Np. Np was analyzed in vivo at PORTS since about 1965.

5.1.2.9   *In Vitro* Bioassay for Trivalent Actinides (Americium)

Routine monitoring for ^{241}Am was not conducted. In 1994 alpha spectroscopy was utilized for some suspect urine samples. A 0.5 dpm/liter MDC was reported. In 1998, Bechtel Jacobs sent their samples offsite to ORNL or offsite vendors. ORNL had a 0.02 dpm/sample MDA for ^{241}Am.

5.1.2.10   *In Vitro* Bioassay for Plutonium

Routine monitoring for ^{238,239,240}Pu was not conducted. In 1994, alpha spectroscopy was utilized for some suspect urine samples. A 0.5 dpm/liter MDC was reported. In 1998, Bechtel Jacobs sent their samples offsite to ORNL or offsite vendors. ORNL had a 0.02 dpm/sample MDA for ^{238,239,240}Pu.

According to the Mass Balance report for PORTS (BJC, 2000, p 65), 0.23 grams of plutonium was processed from RU during the site's lifetime. Small amounts of TRU were also processed. Air sample analysis in the site indicated presence of TRUs. The site was broken up into two categories:

1. Process buildings with TRUs and thorium at < 0.05% alpha activity.

2. Non-process buildings with TRU =0.5% (default) to 2%. Only two facilities met this criteria – 705 & 710.

5.1.2.11   *In Vitro* Bioassay for Thorium

No bioassay for ^{231}Th was conducted at PORTS. The bioassay analysis for ^{234}Th has a MDC of 8000 dpm/liter. Urinalysis was performed for emergency situations (Hill and Strom 1993, p15.33).

5.1.2.12   *In Vitro* Bioassay for Protactinium

The bioassay analysis for ^{234m}Pa has a MDC of 8000 dpm/liter. Urinalysis was performed on emergency situations (Hill and Strom, 1993, p 15.37). Table 5.1.2.12 summarizes the MDC and urine analysis methods used for technetium, uranium daughters and the transuranics.

Table 5.1.2.12-1. Summary of ^{99}Tc, ^{237}Np, ^{241}Am, ^{239}Pu, ^{231}Th and ^{234m}Pa Urinalysis

Period	Personnel Organization	Radionuclide	MDC Urinalysis	Urinalysis Method
1/1/1965-Present	PORTS USEC	^{99}Tc	5000 dpm/liter	Gas Flow Proportional & LSC
10/1/1998-Present	PORTS BJ	^{99}Tc	200 dpm/liter ORNL 33 dpm/liter Quanterra	Chemical Separation and LSC
1/1/1998-Present	BJ &	^{237}Np	0.02 dpm/1 liter ORNL	Alpha Sspectroscopy
1/1/1994-12/31/97	BJ + USEC	^{237}Np	0.5 pCi/1 liter ORNL	Alpha Spectroscopy
1/1/1998-Present	BJ &	^{241}Am	0.02 dpm/1 liter ORNL	Alpha Spectroscopy
1/1/1994-12/31/97	BJ + USEC	^{241}Am	0.5 pCi/1 liter ORNL	Alpha Spectroscopy
1/1/1998-Present	BJ &	^{239}Pu	0.02 dpm/1 liter ORNL	Alpha Spectroscopy
1/1/1994-12/31/97	BJ + USEC	^{239}Pu	0.5 pCi /1 liter ORNL	Alpha Spectroscopy
1/1/1994-Present	BJ & USEC	^{234}Th	8000 dpm/1 liter	Gas flow Proportional & LSC
1/1/1994-Present	BJ & USEC	^{234m}Pa	8000 dpm/1 liter	Gas flow Proportional & LSC

5.2   CORRECTING FOR URINALYSIS VOLUME

At PORTS all on-site urine sample analysis are from spot samples collected at the end of shift on Friday (Hill and Strom, 1993, p 15.4). The samples, which have been sent off site since 1998, are 24-hour samples. All samples can be normalized to the reference volume of 1.4 liters/day by knowing either the period or volume collected. Normalization by volume would be appropriate for the spot samples and might be appropriate for the 24-hour samples if the volume is not within the range of normal variability for 24-hour samples.

Note that some of the urine samples are utilized for chemical monitoring: 50 ml for fluorides, 50 ml for mercury and 50 ml for other chemicals.

5.3   *IN VIVO* MDAS, COUNTING METHODS AND REPORTING PRACTICES

The whole body counting system was routinely started in 1965. The Y-12 mobile *in vivo* laboratory was operated by Martin Marietta Energy Systems (MMES) and utilized under contract with the Goodyear Atomic Corporation Technical Document. The mobile unit operated on the GAT site for between 12 to 16 weeks per year and was operated by Y-12 employees (GAT, 1985b, p1).

The *in vivo* unit employed two large 9" NaI (Tl) detectors for the detection of gamma and bremsstrahlung radiation in the chest cavity. Non-transportable or insoluble compounds such as U_3O_8 are retained in the lungs for extended periods of time. Therefore, this *in-vivo* monitoring technique is used for detecting these types of compounds. Monitoring for type M ^{99}Tc activity in the lung using bremsstrahlung radiation from the 85 keV (ave.) beta was also attempted. Soluble or transportable compounds such as UO_2F_2 and ^{99}Tc (Type F) are monitored via urinalysis (GAT, 1985b). The MDAs, recount values and restriction levels are summarized in Table 5.3-1. The results are given in mass or activity units.

Table 5.3-1. MMES Chest Counter /1965 –1990 MDAs and Restriction Levels

Measurement Category	Enriched Uranium (^{235}U)	Depleted Uranium (^{238}U)	Total Uranium (^{238}U + ^{235}U)	Tc-99	Np-237
Quantity Below Which Detection is Impossible	100 µg	4 mg 1.46 nCi	4 mg	1 µCi	0.2 nCi
Quantity at Which Recount is Required to Confirm Data	100 µg	4 mg	4 mg	1 µCi	1.7 nCi
Restrictive Value	240 µg	37 mg	27 mg	9 µCi	17 nCi

Source: GAT, 1985b, p 27.

A summary of *in vivo* monitoring at PORTS from 1965–1985 is presented in Table 5.3-2. The total number of counts above the MDA is given for each year. As can be seen, PORTS had an active *in vivo* monitoring program.

Table 5.3-2. *In Vivo* Summary (1965–1985)

Year	Total Counts Taken	Number of Counts >100 µg ^{235}U	Year	Total Counts Taken	Number of Counts >100 µg ^{235}U
1965	27	18	1976	411	58
1966	30	14	1977	971	96
1967	236	28	1978	542	29
1968	364	39	1979	497	15
1969	393	73	1980	924	4
1970	147	32	1981	868	2
1971	179	56	1982	910	1
1972	157	36	1983	632	3
1973	392	26	1984	613	0
1974	521	65	1985	798	4
1975	684	92			

Source: BJC, 2000, p 70.

A second chest counting unit, the Helgesson Phoswich detector system that was comprised of a 5" diameter, ½" NaI detector and a 1 ½" CsI thick detector, was used from 1991 through 1995. The chest counter had four phoswich detectors, two located over the whole body, two located over the thighs (background subtraction). Its detection capabilities are listed in Table 5.3-3.

The Helgesson system was used as a Quality Assurance check on the urine analysis and air sampling programs at PORTS and not as a routine bioassay system. The MDAs and restriction levels are summarized in Table 5.3-3.

Table 5.3-3. Helgesson Chest Counter /1991-1995

Compound	Enriched Uranium	Total Uranium
Quantity Below Which Detection is Impossible (^{235}U)	40 - 70 µg	2 – 4 mg
Quantity at Which Recount is Required to Confirm Data	40 - 70 µg	2 – 4 mg
Restrictive Value	240 µg	27 mg

^{137}Cs is reported in the chest counting database. The National Council on Radiation Protection and Measurements Report 94 (NCRP 94, p 160) reports body burdens in the United States from fallout. In the 1960s and 1970s, fallout contributed to the body burdens with the most peaking at 19 nCi in 1964. ^{137}Cs burdens would likely not have been due to occupational exposures at PORTS and can be ignored.

The uranium isotope ^{235}U decays with a physical half-life of 7.0 x 10^8 years to ^{231}Th by alpha emission. ^{231}Thorium, with a half-life of 25.5 hours, decays by beta emission and grows rapidly into equilibrium with ^{235}U. There is no significant in-growth of other daughters. The gamma emissions from ^{235}U that can be used for *in vivo* counting are 144 keV (10.5%), 186 keV (54.0%), and 200 keV (5.0%) (Hill and Strom, 1993, p 15.17).

The uranium isotope ^{238}U decays with a physical half-life of 4.5 x 10^9 year to ^{234}Th by alpha emission. Thorium 234, with a half-life of 24 days, decays by beta emission and grows into equilibrium with ^{238}U. The two low energy photons emitted from ^{238}U, namely 13.0 keV (2.96%), and 15.1 keV (4.47%), cannot be used for *in vivo* counting. In actuality the ^{234}Th photons are used to indirectly estimate the quantification of ^{238}U. The photons used are 63.3 keV (3.81%), 92.3 keV (2.73%) and 93.8 keV (2.69%) (Hill and Strom, 1993, p 15.29).

The uranium isotope ^{234}U decays with a physical half-life of 2.4 x 10^5 years to ^{230}Th by alpha emission. The long half-life of ^{230}Th prevents significant in growth of radioactive daughters. The gamma emissions cannot be utilized for *in vivo* counting. The isotope is quantified indirectly based on the amount of ^{235}U or ^{238}U present (Hill and Strom, 1993, p 15.17).

The uranium isotope ^{236}U decays with a physical half-life of 2.4 x 10^5 years to ^{232}Th by alpha emission. The long half-life of ^{232}Th precludes the in growth of radioactive daughters. The 13.0 keV (3.36%) and 16.1 keV (5.08%) gamma emissions are likely too low to be utilized for in vivo counting (Hill and Strom, 1993, p15.29).

In summary, ^{235}U was probably counted directly using the 186 kev photon; ^{238}U was probably counted via the 63.3 keV photon from ^{234}Th assuming equilibrium with ^{238}U; and ^{234}U and ^{236}U were not counted *in vivo*.

^{234m}Pa could be monitored through the gamma emissions of ^{234m}Pa for *in vivo* counting. This is not currently done at PORTS.

^{99}Tc is detected by Bremsstrahlung production by in-vivo counting for insoluble forms. (ORAUT-TKBS-0014-5, 5.3.2.3) However, urine analysis is the routine monitoring bioassay system at PORTS.

^{237}Np decays with a physical half-life of 2.14 x 10^6 years to ^{233}Pa by alpha emission. ^{233m}Pa, with a half-life of 27 days, decays by beta emission and grows into equilibrium with ^{237}Np. The low energy photons from ^{237}Np that can be used for *in vivo* counting are primarily the 86.5 keV (12.6%) and from ^{233}Pa, the 94.7 keV (10.1%), 98.4 keV (15.3%) and the 312 keV (36.0%) photons (Hill and Strom, 1993, p 15.43). Naturally-occurring radon daughters provide for a false positive for ^{237}Np.

5.4 ABSORPTION TYPES

Radioactive materials classified as F, M, or S (fast, medium or slow) depends on their retention time in the pulmonary region. These designations are similar to the classes D, W and Y, but refer strictly to the rate of absorption from the respiratory tract to the blood.

International Commission on Radiological Protection (ICRP) Publication 68 lists UF_6, UO_2F_2 and UO_2 (NO_3) (uranyl nitrate) as inhalation type F, UF_4 and UO_3 as type M and U_3O_8 and UO_2 as type S. UO_2 could be considered type S (ICRP 68, 1994, p 83).

The chemical form and the enrichment have varied over time at PORTS. From about 1975–1978 most of the HEU up to about 93% enrichment was produced at PORTS. Oxide conversion produced the

potential for insoluble uranium compound exposures in building X-705E from 1957-1978. Recovery and decontamination processes still evoke the potential of insoluble uranium exposures. In building X-344 a UF_4 to UF_6 conversion facility operated from 1958–1962. This also produced insoluble forms of uranium. Current air sampling analysis (1992-2003) and bioassays has indicated type S material in these two locations at PORTS. All other areas indicate a Type F form of uranium. Table 5.1.2.6-4 lists solubility types for the separate facilities and indicates a range of type F-S for all buildings with the exception of X-344 and X705 (E). Most urine analysis cases at PORTS indicated soluble uranium with a demonstrated 1.25-day effective half-life. Insoluble uranium with a 120-day effective half-life was used for dose calculations as indicated in the site literature (GAT, 1966a, p 1 supplement).

^{99}Tc is an inhalation hazard. The tendency of the ^{99}Tc is to accumulate at the higher stage levels at purge and vent areas. Detection levels at PORTS have been limited to about 5000 dpm/liter via urine analysis. ^{99}Tc has appeared to act like a type F material. Oxides, hydroxides, halides, and nitrates of technetium are assigned to inhalation type M; all others are assigned to type F (ICRP 68, 1994, p 82). According to Hill and Strom (1993), Tc at PORTS exists as Tc pertechnetate so type F is suggested. Classifying to type M should be done if the organs of interest are the ET airways, LLI wall, Colon or lungs, and classifying to type F should be done if the organs of interest are ST wall (ICRP 68, 1994).

^{237}Np is type M for all compounds (ICRP 68, 1994, p 83).

Plutonium oxides, carbides, and hydroxides are type S, nitrates and other compounds are type M. Where plutonium is a small contaminant in a uranium matrix, the plutonium might behave the same as the host uranium, so the exact absorption type for Pu at PORTS is not well known. The DR can assume either type M or S to maximize the dose to the organ of concern.

Americium is type M for all compounds; however, as for plutonium, trace amounts in a host matrix may behave the same as the host compound. Therefore, the DR can assume either type M or S.

Thorium oxides, carbides, and hydroxides are type S, nitrates and other compounds are assigned to type M. Because of the host matrix effect mentioned above, the DR can assume either M or S.

5.5   WORKPLACE AIR SAMPLING DATA AS APPLICABLE

PORTS has an extensive air sampling program. All process areas and most other work areas are sampled by low to high volume air samplers. Grab samples have also been taken from the beginning of operations. 50 μg/m^3 for uranium was the limit utilized from 1955 and the air concentration limits have gone through transitions. Despite the changes from AEC to the Energy Research and Development Administration (ERDA) to the U.S. Department of Energy (DOE) regulations the overall limit to uranium for both soluble and insoluble forms has remained about the same. For example, the calculated Derived Air Concentration based upon on-site air sampling and DOE 10 CFR 835 for Class Y uranium was equal to 1.9 x 10^{-11} μCi/ml (BJC, 2000, p 101) and on the older ERDA regulations 6 x 10^{-11} μCi/cc.

Portable air samples (PAS) have also been taken, but primarily limited to larger maintenance work. Area and PAS data have been used to establish airborne areas as well as characterizing the percentage of TRU in the air. If a positive bioassay occurs and work in a specified area can be established, more specific detail of the radionuclide make-up and chemical form can be determined by utilizing Table 5.1.2.6-4. In addition, removable contamination surveys have been useful in characterizing uptakes in the past at PORTS.

However, the air sampling data is very extensive and prone to larger bias and systemic errors. The air sample data from 1993 through 1999 was utilized to create the activity fraction column for the tables for each building in Table 5.1.2.6-4. In addition, site-specific bioassay and contamination monitoring aided in characterizing these areas. No formal attempt i.e. in the literature has been developed for the PORTS facility unlike other DOE sites to document workplace air monitoring results or practices. Air sample records can probably be located but are not currently stored in a centralized location.

5.6   INTERFERENCE AND UNCERTAINTY

5.6.1   Contamination of Samples

Excreta samples were collected off-site. There still may have been a possibility for cross-contamination from workers' hands to the samples. Laboratory contamination issues, sample count failures or mix-ups were possible, but not recurrent. Quality controls had been placed upon sample collection, data transferal and equipment operation since the start of operations at PORTS.

If there was a high sample count, a recall sample was requested and counted. Other samples may have been re-counted to ensure that possible laboratory error could be eliminated. If a sample seemed to be a true high count further urine samples were requested and an *in vivo* count requested and obtained. The solubility type of the radionuclide could then most likely be determined.

Contamination to the whole body could interfere with the lung counting. Workers were asked to shower and change prior to the chest count. If contamination was suspected, the worker would repeat the shower. A section in the database and on the data form indicates a "FB Ratio". This ratio was used to determine the possibility of external contamination as opposed to internal deposition. If the ratio was greater than one, the measured activity was closer to the surface and probably due to external contamination. A recount would be made after another shower. Subsequent re-counts and urine bioassays would be given to validate the positive chest count or urine count, or both.

5.6.2   Uncertainty

Uncertainties for bioassay measurements were not stated in the records or database.

PORTS had a practice of submitting routine urine samples at most within two days after the last work day. The sample was to be taken at the medical center within two hours of clocking in to minimize the possibility of inadvertent contamination from loose material on a person's clothing. If the individual comes to the hospital after the two hour period, the receptionist would reschedule the specimen for the next Monday. Non-routine samples had a greater possibility of cross contamination because the specimens were given either immediately or within anytime during the work day. (GAT, 1966a)

Dietary intakes of uranium pose a potential problem in interpreting urine bioassay results for PORTS workers. Because studies of the average daily uranium excretion on PORTS residents do not appear to have been performed, it is not possible to make corrections for the contribution of nonoccupational intakes of uranium to a given urine sample result. However, to put a given result into perspective, a nominal daily (24-hour) urinary excretion rate for uranium of 0.43 µg (environmental decision level at 95% confidence) can be used (BJC, 1999).

5.7 ASSESSMENT OF INTAKE FOR MONITORED AND UNMONITORED WORKERS

5.7.1 Monitored Workers with Measurable Intakes

PORTS had an extensive bioassay program from the beginning of its operations. Urine analysis started out in 1954 with the addition of chest counting in 1965. Although the earlier techniques had their sensitivity limitations, the detection sensitivity seemed to keep pace with the fast paced regulatory and safety changes. Seldom did workers achieve or surpass the body burden of the radionuclides of concern, as a whole. Some assumptions that a Dose Reconstructor can make:

- Time of intake – If the date or time of intake to a particular bioassay result is unknown, the intake occurred mid-way between prior and the current bioassay result. (ICRP 78, 1997).

- Radionuclide, particle size and solubility type- Tables 5.1.1-2, 5.1.2.6-3, and 5.1.2.6-3 list the values. If the work location of the employee is not known, default assumptions for radionuclide, particle size and solubility type are equal to those in the most closely related facility.

- When an intake of uranium has been determined, add intakes of the other radionuclide contaminants in proportions established in Table 5.1.2.6-4. If there are specific bioassay results for any of the contaminants that indicate a higher intake than determined using the ratio to uranium, use the specific bioassay results to determine the intake for that radionuclide.

- Uranium enrichment – 3.5% unless HEU of 93% is suspected.

- Uranium intake activity can be assumed to be ^{234}U.

- Route of intake – Intake is by inhalation. Note ingestion is possible.

5.7.2 Monitored Workers with Nothing Detected in Bioassay

For monitored workers with no positive results, a triangular distribution is used, with the mode determined using half of the MDC value and the maximum using the MDC as input into the dosimetry codes.

5.8 *IN VITRO* URINALYSIS AND *IN VIVO* LUNG COUNTING DATA CODE INFORMATION

The codes utilized for PORTS records and databases have been consistent over most of the history of the site. Since the 1950's both the industrial hygiene bioassay and health physics bioassay information was entered on the same form. The industrial hygiene information is not needed for radiation dose reconstruction. Table 5.8-1 lists a summary of known codes along with their interpretations.

Table 5.8-1. Internal Dosimetry Record Codes

Measurement Type	Column Identifier	Code	Interpretation
Urine Bioassay	Alpha (dpm/100 ml)		Total alpha counts per 100 ml of urine.
Urine Bioassay	Beta(Tc) (dpm/100ml)		Total beta counts per 100 ml of urine (all counts assumed beta).
Urine Bioassay	Column listed "A"	0	Not tested.
		1	Tested.
		2	Recall or above a flag.
		4	Restriction.
		Blank	< MDA.
Urine Bioassay	Alpha (dpm/dL)		Total alpha counts per 100 ml of urine (dL = deci-liter).
Urine Bioassay	URAN		Uranium in mg/liter by fluorometry.
Urine Bioassay	ALPHA		Total alpha counts per 100 ml of urine (dL = deci-liter) by Gas flow proportional.
Urine Bioassay	TECH		Total Beta counts per 100 ml of urine (dL = deci-liter) by Gas flow proportional.
Urine Bioassay	FLUOR, MERC, LEAD, ZINC & CAD		Fluorine, Mercury, lead, zinc and cadmium Chemical analysis not to be included in dose reconstruction.
Urine Bioassay	IV	IV	Insufficient urine volume.
Urine Bioassay	OE	OE	Operator error.
Lung Bioassay	FB Ratio	FB	This is a measure of how close to the front or back the internal contamination is. A ratio of greater than one may indicate external contamination.
Lung Bioassay	Negative value in Tot-U column		A negative value indicates less than the 4 mg MDA for total Uranium.

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