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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
10/07/2005	00	New technical basis document for the Lawrence Livermore National Laboratory – Occupational Internal Dose. First approved issue. Training required: As determined by the Task Manager. Initiated by Jay J. Maisler.
03/16/2010	01	Revision to incorporate Attributions and Annotations section and Advisory Board comments associated with the issuance of a LLNL SEC Class. Incorporates the internal coworker study (ORAUT-OTIB-0065) into the TBD as Attachment B. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Mutty M. Sharfi, Stephen Spanos and Ralph W. Kenning.
12/13/2010	02	Revision to update Table B-9 uranium 95th percentile coworker intakes for those time periods that were based on the 50th percentile intakes with a GSD that were rounded up to 3. The previous revision used the actual GSDs in Table B-8 that were less than 3 to determine the 95th percentile coworker intakes for those time periods. Added information pertaining to solubility types for aged related fission products for P-32 and C-14 in Section 5.3.2.3 and added associated footnotes to Tables B-25 and B-26. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Objective Manager. Initiated by Stephen Spanos and Ralph Kenning.
08/12/2016	03	Revision initiated to incorporate SEC-000221. Revised Section 5.2.3.1 to incorporate additional guidance for which tritium exposures in the form of metal tritides at LLNL should be considered. Revised Section 5.3.1 to incorporate additional information on gross alpha measurements, and more detailed guidance on the analysis of gross alpha, and plutonium bioassay results. Revised Section 5.3.2.1 to incorporate revisions to OTIB-0054. Added Section 5.3.2.4 to incorporate guidance on the analysis of tritium measurements. Revised Section B.4.2.3 to incorporate revisions to OTIB-0054. Revised Attachment C to incorporate additional guidance on the assignment of coworker intakes. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Objective Manager. Initiated by Stephen Spanos and Ralph W. Kenning.

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ACRONYMS AND ABBREVIATIONS

ALARA	as low as is reasonably achievable
AMAD	activity median aerodynamic diameter
ATR	Advanced Test Reactor
AVLIS	Atomic Vapor Laser Isotope Separation
AWE	Atomic Weapons Employer
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
cpm	counts per minute
CWT	chest wall thickness
d	day
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
DU	depleted uranium
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
F	fast (solubility rate)
g	gram
HEPA	high-efficiency particulate air
HEU	highly enriched uranium
HPGe	hyperpure germanium
hr	hour
HT	elemental tritium or tritiated gas
HTO	tritiated water vapor
ICP-MS	inductively coupled plasma–mass spectrometry
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt, 1,000 electron-volts
L	liter
LANL	Los Alamos National Laboratory
LLNL	Lawrence Livermore National Laboratory
LPTR	Livermore Pool-Type Reactor
LSC	liquid scintillation counting
M	moderate (solubility rate)
M&E	Mechanical and Electrical (engineering divisions)
MDA	minimum detectable amount
MeV	megaelectron-volt, 1 million electron-volts
MFP	mixed fission and activation product
mg	milligram
mrem	millirem

MT	metal tritide
nCi	nanocurie
NESHAPs	National Emission Standard for Hazardous Air Pollutants
NIOSH	National Institute for Occupational Safety and Health
NTS	Nevada Test Site
NU	natural uranium
OBT	organically bound tritium
ORAU	Oak Ridge Associated Universities
pCi	picocurie
RTNS	Rotating Target Neutron Source
RU	recycled uranium
R&D	research and development
S	slow (solubility rate)
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
SS	Super S absorption type
TBD	technical basis document
TRIGA	Training, Research, Isotopes, General Atomics
TRU	transuranic
Type SS	Super S absorption type
U.S.C.	United States Code
WB	whole-body
WBC	whole-body counter
yr	year
§	section or sections
µg	microgram
µm	micrometer

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 historical background information and guidance to assist in the preparation of dose reconstructions at particular Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies). These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used to refer to an area, building, or group of buildings that served a specific purpose at a DOE or AWE facility. It does not mean nor should it be equated to an "AWE facility" or a "DOE facility." The terms AWE and DOE facility are defined in sections 7384l(5) and (12) of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA), respectively. An AWE facility means "a facility, owned by an atomic weapons employer, that is or was used to process or produce, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining or milling." 42 U.S.C. § 7384l(5). On the other hand, a DOE facility is defined 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 [DOE] (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program);" and with regard to which DOE has or had a proprietary interest, or "entered into a contract with an entity to provide management and operation, management and integration, environmental remediation services, construction, or maintenance services." 42 U.S.C. § 7384l(12). The Department of Energy (DOE) determines whether a site meets the statutory definition of an AWE facility and the Department of Labor (DOL) determines if a site is a DOE facility and, if it is, designates it as such.

Accordingly, a Part B claim for benefits must be based on an energy employee's eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility's designated time period and location (i.e., covered employee). After DOL determines that a claim meets the eligibility requirements under EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and the types of radiation exposure to be included in an individual dose reconstruction. Under EEOICPA, eligible employment at a DOE facility includes individuals who are or were employed by DOE and its predecessor agencies, as well as their contractors and subcontractors at the facility. Unlike the abovementioned statutory provisions on DOE facility definitions that contain specific descriptions or exclusions on facility designation, the statutory provision governing types of exposure to be included in dose reconstructions for DOE covered employees only requires that such exposures be incurred in the performance of duty. As such, NIOSH broadly construes radiation exposures incurred in the performance of duty to include all radiation exposures received as a condition of employment at covered DOE facilities in its dose reconstructions for covered employees. For covered employees at DOE facilities, individual dose reconstructions may also include radiation exposures related to the Naval Nuclear Propulsion Program at DOE facilities, if applicable. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction.

NIOSH does not consider the following types of exposure as those incurred in the performance of duty as a condition of employment at a DOE facility. Therefore these exposures are not included in dose reconstructions for covered employees (NIOSH 2010):

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

5.1.1 Purpose

This technical basis document (TBD) discusses historical practices at the Lawrence Livermore National Laboratory (LLNL) and provides information for the evaluation of internal and external dosimetry data for unmonitored and monitored workers; it can serve as a supplement to, or substitute for, individual monitoring data.

This document provides a uniform and consistent approach to assessing occupational internal dose at LLNL for dose reconstructions for NIOSH in relation to EEOICPA. It provides guidance to dose reconstructors on input parameters that are specific to LLNL employees, as well as the approach for employees with either missing or no monitoring information.

This TBD includes guidance on selecting source terms, interpreting in vivo and in vitro measurement results, and instructions for assessing dose for monitored and unmonitored employees.

5.1.2 Scope

Occupational internal dose refers to exposures workers received inside the facilities at LLNL from ingestion, inhalation, or absorption of radioactive materials in the course of work. This section provides site-specific information to the dose reconstructor for use in reconstructing radiation doses for workers that were employed at LLNL, via multiple pathways of exposure including inhalation, ingestion, and absorption. In some cases, exposures occurred via wounds and were not restricted to intakes inside buildings.

Three classes of LLNL employees have been added to the SEC as described in the section below. The addition of these classes of employees has limited the scope of potential internal dose reconstructions at LLNL to the following.

- Internal dose from tritium from 1957 to the present;
- Internal dose from mixed fission and activation products (MFPs) from 1974 to the present;
- Internal dose from americium from 1957 to the present;
- Internal dose from plutonium from 1957 to the present;
- Internal dose from uranium, excluding ^{233}U, from 1958 to the present;
- Internal dose from other radionuclides at LLNL may be reconstructed from 1957 or 1958 to the present. The dose reconstructor should refer to the appropriate section later in this document for when the internal dose for that particular radionuclide can be reconstructed.

The internal dose from americium, plutonium, and uranium can be extended for employment before the above-listed periods based on a worker's later bioassay results and work history.

The remainder of this section describes the SEC, certain historical events that are important to internal dose reconstruction, and the overall approach to internal dose reconstruction for LLNL workers. Section 5.2 discusses source term information. Section 5.3 discusses in vitro and in vivo bioassay measurement methods. Section 5.4 describes significant incidents with internal dose potential. Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 5.5.

Attachment A provides materials for reconstruction of occupational internal dose for monitored workers. Attachment B provides coworker intakes. Attachment C describes scaling of coworker dose intakes consistent with type and duration of exposures.

5.1.3 Special Exposure Cohort

Classes Added to the SEC

NIOSH has determined, with concurrence from the Secretary of Health and Human Services (HHS), that mixed fission and activation doses at LLNL cannot be reconstructed with sufficient accuracy between 1950 and 1973, inclusive. For this reason, the following class of LLNL employees has been added to the SEC (Leavitt 2008):

> *Employees of the DOE, its predecessor agencies, and DOE contractors or subcontractors who were monitored for radiation exposure while working at the Lawrence Livermore National Laboratory from January 1, 1950, through December 31, 1973, for a number of workdays aggregating at least 250 workdays or in combination with workdays within the parameters established for one or more other classes of employees in the Special Exposure Cohort.*

NIOSH has also determined that mixed fission and activation doses for workers not monitored for radiation exposure at LLNL also cannot be reconstructed with sufficient accuracy between 1950 and 1973, inclusive. For this reason, the following class of LLNL employees has been added to the SEC (Sebelius 2010):

> *All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Lawrence Livermore National Laboratory in Livermore, California from January 1, 1950 through December 31, 1973, for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort.*

NIOSH has further determined that internal doses from ^{233}U at LLNL cannot be reconstructed with sufficient accuracy between 1974 and 1989, inclusive. For this reason, a class of LLNL employees has been added to the SEC (Burwell 2016).

> *All employees of the Department of Energy, its predecessor agencies, and its contractors and subcontractors who worked in any area at the Lawrence Livermore National Laboratory in Livermore, California, during the period from January 1, 1974 through December 31, 1989, for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees in the SEC.*

Each SEC class includes all workers during the SEC period. Because of the identified dose reconstruction infeasibility, all dose reconstructions for all workers having employment during an SEC period are considered partial dose reconstructions. If monitoring data are available for workers in the SEC, dose is to be assigned as appropriate based on such data. However, such dose reconstructions are still considered partial dose reconstructions because HHS has determined that exposure to mixed fission and activation products, or ^{233}U during the LLNL SEC periods cannot be bounded.

5.2 SOURCE TERM

LLNL workers handled a variety of radionuclides as part of their routine work. The key elements in the source term were plutonium and tritium, although others were used at various times and in various forms. Gloveboxes, extensive filtered ventilation systems, and other engineering controls were used to minimize personnel exposures from the outset of operations at LLNL (Balanda 1966). Administrative controls were implemented to minimize the potential for an intake from handling

radioactive materials. A majority of the buildings at LLNL handled or processed unsealed sources of radioactive materials; persons who were assigned to operate processes or laboratory research had a potential for exposure (ORAUT 2005a; Mansfield 1989). Administrative and other support personnel in the individual buildings had lesser potential for exposure because of engineering and administrative controls. For dose reconstruction, it can be assumed that internal source terms were introduced at LLNL when it began operations on September 2, 1952.

Before the early 1960s, the only bioassay methodology LLNL used to monitor employees for intakes of radionuclides was urine bioassay, and the primary focus was excreted tritium. A review of previous bioassay samples indicates that bioassay monitoring started in March 1957. The analysis of bioassay samples continues to this day for plutonium, americium, uranium, MFPs, a variety of tracer radionuclides, iodine, and tritium (Mansfield 2000). Air monitoring in workplaces has been a common surveillance method. A brief description of the building processes is provided in ORAUT-TKBS-0035-2, *Lawrence Livermore National Laboratory – Site Description* (ORAUT 2005b, Table 2-2). The buildings that were assigned to handle plutonium are noted. Buildings in which uranium was processed and handled are also described. ORAUT (2005b, Table A-1) provides a cross reference of main site building number changes that took place in 1966. The dose reconstructor can refer to this list if needed when assessing building location information for a worker's claim.

In vivo methodologies began on an investigational basis in 1964 with focus on high-energy gamma emitters (i.e., greater than 200 keV). In the 1970s there were attempts to detect low-energy photon emitters (i.e., 60-keV gamma radiation from the decay of ^{241}Am and even plutonium L-shell X-rays). LLNL can provide a broad spectrum of in vivo counting services with varying degrees of detectability (Mansfield 2000).

A review of in-house procedures LLNL used to assess the concentration of radioactivity in urine indicates that quality control steps were an integral part of the process (Miller 1979). For example, LLNL ran duplicates consistently, and comparisons of results to "known quantities" were a critical step. Therefore, in vitro results from in-house processing are generally reliable. However, interpretation of those results can be difficult, primarily because they might not have considered the contribution of environmental radioactivity (i.e., uranium, thorium). Because sample collection could have occurred at work (e.g., "in-field" tritium analyses, Monday morning urines), cross-contamination could be an issue.[1]

The primary LLNL missions have been weapons research and development (R&D), controlled nuclear weapons research, peaceful uses of nuclear explosives, biomedical research, and laser fusion research. In addition, work in nonnuclear technologies and materials testing has been ongoing since at least 1976 (LLL ca. 1978). LLNL conducted aboveground (atmospheric) and underground tests at both the Pacific Proving Ground and the Nevada Test Site (NTS) from 1953 to 1992. Although the tests were conducted off site, test planning and preparation occurred at LLNL. In 1971, LLNL managed the CANNIKIN test at Amchitka Island, Alaska.

Unless site-specific information is available, the particle size is assumed to be 5-µm activity median aerodynamic diameter (AMAD), as recommended in ICRP (1995, paragraph 5).

5.2.1 Plutonium

If a monitoring result refers to "weapons-grade plutonium" or plutonium, dose reconstructors can use the isotopic mix listed in Tables 5-1 and 5-2 for material aging times (Mansfield 2000). From September 1952 through 1957, fresh plutonium should be assumed. For 1958 through 1962, assume

[1] Twenty-four-hour collections typically took place at the employee's home on weekends. However, spot samples were often collected in site restrooms (Mansfield 1989).

a 5-year-old plutonium mixture. From 1963 forward, 10-year-old plutonium should be assumed. For assessing type SS solubility, refer to ORAUT-OTIB-0049, *Estimating Doses for Plutonium Strongly Retained in the Lung* (ORAUT 2010).

Table 5-1. Specific activity (Ci/g) for specified ages of weapons-grade plutonium (Mansfield 2000).

Nuclide	Fresh	5-yr	10-yr	20-yr
Pu-238	6.85E-03	6.59E-03	6.33E-03	5.85E-03
Pu-239	5.81E-02	5.81E-02	5.80E-02	5.80E-02
Pu-240	1.37E-02	1.37E-02	1.37E-02	1.37E-02
Pu-241	5.98E-01	4.70E-01	3.69E-01	2.28E-01
Pu-242	1.57E-06	1.57E-06	1.57E-06	1.57E-06
Am-241	0.00E+00	4.24E-03	7.54E-03	1.21E-02
Pu-239+240	7.18E-02	7.18E-02	7.17E-02	7.17E-02
Pu-alpha	7.87E-02	7.84E-02	7.80E-02	7.76E-02

Table 5-2. Activity ratios in mixture for specified ages of weapons-grade plutonium (Mansfield 2000).

Nuclide	Fresh	5-yr	10-yr	20-yr
Pu-239+240:Am-241	N/A	1.69E+01	9.51E+00	5.93E+00
Pu-239+240:Pu-238	1.05E+01	1.09E+01	1.13E+01	1.23E+01
Pu alpha:Pu-239+240	1.10E+00	1.09E+00	1.09E+00	1.08E+00
Pu alpha:Pu-238	1.15E+01	1.19E+01	1.23E+01	1.33E+01
Pu-241:Pu alpha	7.60E+00	6.00E+00	4.73E+00	2.94E+00

5.2.2 Uranium

If a monitoring result refers to “uranium,” dose reconstructors can use the isotopic mix in Table 5-3 for enrichment levels (Mansfield 2000). LLNL received uranium metal from the Y-12 plant that was known to contain recycled uranium (RU) contaminants including technetium, neptunium, thorium, and plutonium [1]. ORAUT-TKBS-0014-5, *Y-12 National Security Complex – Occupational Internal Dose*, (ORAUT 2012) discusses the typical concentration of RU contaminants. Table 5-4 provides the list of RU ratios to be applied for uranium material from Y-12. The RU ratios in Table 5-4 reflect upper bound values. Given that there is uncertainty in the specific batches of uranium LLNL received, the upper bound values in Table 5-4 should be used for dose reconstruction. LLNL also used natural uranium (NU) from the Feed Materials Production Center for the Atomic Vapor Laser Isotope Separation (AVLIS) project. The AVLIS project converted kilogram quantities of NU to slightly enriched uranium, up to 5%, as a demonstration plant. The AVLIS project operated in Buildings 175, 177, and the 482 complex from 1973 through 1999. Table 5-5 provides the list of RU ratios to be applied for uranium associated with the AVLIS project. In the absence of specific information, the Y-12 RU ratios in Table 5-4 should be used.

If the bioassay results are expressed in units of mass, the dose reconstructor should select the isotopic mix equivalent to NU in Table 5-3 in almost all cases at LLNL, in the absence of claim specific information. The uranium in question for bioassay results expressed in units of mass would typically be either DU or NU.

Highly enriched uranium (HEU) came from Y-12 and was used in Building 231 and the Oralloy Shop in Building 321C. HEU was restricted at LLNL, both in areas and amounts, due to criticality concerns. HEU machining was done in Building 321C in either enclosures/hoods or close-capture ventilation systems. Building 231 is an experimental, manufacturing, assembly, test, and materials-handling facility. Building 231 also housed research and development activities. Building 241, which conducted materials science research, had an old glovebox line in Room 1600 that contained EU. As a result, internal exposure potential to HEU in Building 241 would have been remote. Building 327 is

a radiography facility that conducted nondestructive testing of materials including HEU. Because of the nature of the work in Building 327, internal exposure potential to HEU would be low. LLNL performed in vitro and in vivo analysis for HEU. Uranium bioassay results for HEU would be expressed in activity.

Table 5-3. Isotopic information for various uranium enrichment levels (Mansfield 2000).[a,b]

Mixture	Radionuclide	Mass fraction	Activity fraction	Specific activity of mix (Ci/g)
DU	U-238	9.975E-1	9.02E-1	3.72E-7
DU	U-235	2.50E-3	1.45E-2	3.72E-7
DU	U-234	5.00E-6	8.4-E-2	3.72E-7
NU	U-238	9.93E-1	4.77E-1	7.00E-7
NU	U-235	7.20E-3	2.23E-2	7.00E-7
NU	U-234	5.60E-5	5.00E-1	7.00E-7
AVLIS <5% EU	U-238	9.50E-1	4.11E-1	7.78E-7
AVLIS <5% EU	U-235	5.00E-2	1.39E-1	7.78E-7
AVLIS <5% EU	U-234	5.60E-5	4.50E-1	7.78E-7
95% HEU	U-238	3.96E-2	1.98E-4	6.72E-5
95% HEU	U-235	9.50E-1	3.06E-2	6.72E-5
95% HEU	U-234	1.04E-2	9.69E-1	6.72E-5

a. When "U238," "U28," or "U38" was reported, these were units of total U mass (Mansfield 2006a).
b. Notations in bioassay records indicate that starting in 2000 a mass spectroscopy system was used at LLNL. Results from the mass spectroscopy system are reported as mass of U-238. These U-238 bioassay results need to be adjusted for total uranium by dividing by the appropriate U-238 activity fraction, and multiplying by the appropriate specific activity..

Table 5-4. Y-12 recommended RU contaminant ratios default levels based on upper levels of expected ranges (ORAUT 2012).

Isotope	HEU & Oralloy (pCi/µg U)	DU, NU, and 2% EU (pCi/µg U)	HEU & Oralloy (dpm/µg U)	DU, NU, and 2% EU (dpm/µg U)	HEU & Oralloy (pCi/pCi or dpm/dpm total U)	DU, NU, and 2% EU (pCi/pCi or dpm/dpm total U)
Tc-99	4	0.3	8.88	0.666	4.00E-02	0.15
Th-228	0.5	0.003	1.11	0.00666	5.00E-03	1.50E-03
Np-237	0.6	0.003	1.332	0.00666	6.00E-03	1.50E-03
Pu-238	0.2	N/A	0.444	N/A	2.00E-03	N/A
Pu-239	0.2 (Oralloy only)	0.003	0.444 (Oralloy only)	0.00666	2E-03 (Oralloy only)	1.50E-03

Table 5-5. Feed Materials Production Center RU contaminant ratios for NU (ORAUT 2016).

Isotope	pCi/µg U	dpm/µg U	pCi/pCi or dpm/dpm total U
Pu-239	0.00629	0.01396	0.00921
Np-237	0.00250	0.00555	0.00366
Tc-99	0.15435	0.34266	0.22599

5.2.3 Tritium

Tritium was encountered in several forms: tritium oxide as water or gas (HTO), elemental tritium or tritiated gas (HT), organically bound tritium (OBT), and metal tritides (MTs). Each form has unique characteristics. Tritium handling at LLNL was first performed in Building 231 in the early to mid-1950s. Gas fill operations were performed along with other experimental activities. Tritium handling at Building 331 began in 1958. The second phase of Building 331 was completed in 1962. Tritium-filled microspheres were used at LLNL in Buildings 381 and 391 for the Shiva Nova II experiment in 1977. Stable MTs were handled in Building 331, Rooms 154, 150, 138, 145 (Odell and Arthur 1975),

Building 212, and Building 292. Outside these buildings, in cases where exposure to OBT or MTs is not implied by case-specific information, assume HTO. Tritium gas work was done in processing hoods. Compounds of tritium, namely MTs, were usually worked in gloveboxes. The glovebox air was filtered because experimentation with MTs required dry, inert environments to prevent contamination of the material under experimentation with water from the environment. Guidance for estimating doses for OBT and MTs is available in ORAUT-OTIB-0066, *Calculation of Dose from Intakes of Special Tritium Compounds* (ORAUT 2007a).

Site locations for potential exposure to tritium are listed in Attachment A. Bioassay results for tritium are usually available in the dosimetry records.

5.2.3.1 Metal Tritides

Tritium exposures in the form of MT aerosols were primarily titanium tritide targets for the accelerator beams in Buildings 212 and 292. Operations at Building 212 were the forerunner to Rotating Target Neutron Source II (RTNS II) (Building 292). Tritium operations were conducted at Building 212 from 1966 until 1978 when the RTNS II became operational. Operations at Building 212 were significantly reduced after this time and were terminated in 1987. Operations at Building 292 began in 1978 and were terminated in 1987. This facility took over tritium operations that previously had been conducted at Building 212. The titanium MT targets were handled in Building 331, Rooms 150, 138, and 145. The personnel who would have had the potential of exposure to the MT targets are the chemists, researchers, health physics technicians, and accelerator operators and mechanical technicians who performed work on the accelerators.

The targets in Buildings 212 and 292 were changed primarily by the accelerator operators. Occasionally, some of the facility support staff, usually mechanical technicians, changed the targets. The titanium tritide targets for the RTNS I accelerator in Building 212 were changed inside a hood that was exhausted through a high-efficiency particulate air (HEPA) filter. The target was inserted in one side of the hood and changed through glove ports on the other side of the hood. Individuals involved in target change wore gloves, protective clothing (typically a laboratory coat), and a half-mask respirator. A Hazards Control Monitor (Health and Safety Technician) was required to be present for target change. The Hazards Control Monitor was responsible for making radiation measurements and taking air samples to check for airborne contamination. Bioassay samples were collected within 4 hours after the target handling operation. Because of the larger physical size of the titanium tritide targets for RTNS II in Building 292, they were not changed in a hood enclosure as were the RTNS I targets. The workers changing the targets wore anticontamination clothing, plastic shoe covers, a paper hood, gloves, a plastic face shield, and a half-mask respirator. Health and Safety Technicians provided radiation monitoring support (making radiation measurements, taking swipes, and collecting air samples) during the operation. After removing protective clothing, workers were swiped for clothing and skin contamination by the Health and Safety Technician. Bioassay samples were collected within 4 hours after the target handling operations (Myers 2008).

Solubility type M should be assumed for assigning internal dose from titanium tritides. Based on available information, there is no indication that MT solubility type S was a possibility at LLNL.

ORAUT-OTIB-0066 (ORAUT 2007a) provides guidance on the evaluation of MT intakes. MTs are referred to as “tritium particulates.” In addition to potential exposure in the above-listed facilities, the claimant telephone interviews for the claim, if any, can provide indications that a person was exposed to MTs. Building exposure locations are also available in tritium dosimetry records as an aid to the dose reconstructor in assigning tritium dose from MTs.

Based on the above, the following is a list of buildings and job categories for which tritium exposures in the form of MT at LLNL should be considered.

1. Building 212 (RTNS I accelerator):
    a. Accelerators operators,
    b. Researchers,
    c. Mechanical technicians who performed work on the accelerators, and
    d. Health and Safety Technicians (health physics technicians, hazard control monitors).
2. Building 292 (RTNS II accelerator):
    a. Accelerators operators,
    b. Researchers,
    c. Mechanical technicians who performed work on the accelerators, and
    d. Health and Safety Technicians (health physics technicians, hazard control monitors).
3. Building 331 (Tritium Facility), Rooms 150, 138, and 145:
    a. Chemists, and
    b. Health and Safety Technicians (health physics technicians, hazard control monitors).
4. Cases where a claimant telephone interview provide indications that a person was exposed to MTs.

5.2.3.2 Organically Bound Tritium

Tritium exposures in the form of OBT are primarily found in pump oils associated with fume hoods. Exposure to OBT would occur during maintenance if contact with the skin was made. If claim information indicates the need for assigning tritium dose based on OBT, guidance on the calculation of doses from OBT is given in ORAUT-OTIB-0066 (ORAUT 2007a).

5.2.4 Other Limited-Exposure Radionuclides

LLNL has always been a center for research. As such, small-scale use of various radionuclides (in terms of either the number of persons or the activity of the source) that is not addressed above has occurred throughout the history of LLNL. The internal dose potentials due to these nuclides were remote because of the requirements for handling unsealed sources in fume hoods, hot cells, gloveboxes, and dryboxes. LLNL can provide a broad spectrum of in vivo counting services with varying degrees of detectability (Mansfield 2000). This is also evident by the radionuclides assessed by lung counting or whole-body (WB) counting (see Table 5-9 in Section 5.3.3). The dosimetry records should provide this information to the dose reconstructor as part of the case-specific information. In addition, Attachment C provides a method to scale the provided intakes to estimate a credible exposure to individuals who were exposed to rarely encountered radionuclides.

5.3 MEASUREMENT METHODS

In vivo measurement methods began at the site as far back as 1964 when LLNL used sodium iodide detectors connected to simple multichannel analyzers to evaluate ^{40}K and fission products in humans. In 1964, a scanning bed, referred to as the "RIDL," was in use at LLNL. This device was a shadow shield counter equipped with a 3- by 2-in. sodium iodide detector for detection and quantification of relatively high energy gamma photons (i.e., greater than 200 keV). It had some capability for determining the source of radionuclide deposition in subjects. This device was calibrated with a water tank phantom spiked with ^{137}Cs and ^{40}K (Hickman undated).

In April 1965, thin crystal studies began (Schmidt and Anderson 1965). In 1965 and 1966, LLNL identified ^{65}Zn in people working at the Building 153 cyclotron, which dominated the study groups at that time (Author unknown ca. 1988).

Workers were selected for WB or specific organ counting based on direction by the program supervisor and health physics staff. Baseline and termination counts occurred as early as 1966 (Balanda 1966).

From at least 1957, in vitro measurement methods were in use including routine monitoring occurring annually and semiannually during those early days [2]. LLNL selected workers for internal monitoring based on the potential for internal exposure. Employees whose work with radionuclides warranted routine testing for internal radiation were determined based on the supervisor of each operation and the Radiation Safety Section Health Physicist assigned to the area (Balanda 1966). Building 331 had a monthly monitoring program in place. During weapons test support periods, routine sampling frequencies appear to have been increased to at least quarterly for involved employees, with weekly sampling for Building 331 (for tritium) and monthly sampling for Building 251 (Balanda 1966). Through the 1980s, the program supervisor and the area Health Physicist selected workers for internal monitoring based on the potential for internal exposure (Mansfield 1989). The selection of workers for internal monitoring based on the potential for internal exposure continues at LLNL (Mansfield 2000).

Bioassays were supplemented with workplace airborne monitoring (Mansfield 1989). In addition, nasal smears were collected after incidents as a means of assessing intake potential. The dose reconstructor might come across some of these data in worker dosimetry records (Mansfield 1989, Appendix A).

Site 300 had a routine bioassay monitoring program as early as 1971 in which all Site 300 personnel received an employment start and termination WB count. In addition, persons who worked in the firing areas and other radioactive materials handling (i.e., process) areas were bioassayed for gross alpha and beta and underwent fluorometric analysis for uranium. Additional WB count or bioassays were requested at the discretion of the Health Physicist (NAG 1971). Starting in 1980 at Site 300, all persons working in, or who can be expected to have spent a considerable portion of their time in, areas where an airborne uranium hazard might exist were required to receive an annual lung count and a semiannual urine analysis. Other schedules might have been required if radionuclides other than uranium were used and considered a potential internal hazard (Straume 1980). Tables 5-6 and 5-7 list in vivo and in vitro sampling frequencies for various employment periods. In the case of in vitro methods, sample sizes could have been spot (single void), 24-hour collections, or simulated 24-hour collections with creatinine corrections performed if the results were indicative of "significant" intakes (Mansfield 2000).

Selected records report the presence of ^{239}Pu and ^{240}Pu as determined by accelerator mass spectrometry (also called CAMS – Center for Accelerator Mass Spectrometry) and are reported in units of activity that were converted from units of mass. The accelerator mass spectrometry data in a

worker's records could come primarily from two sources (or both): (1) For select individuals with known or suspected uptakes, an intercomparison was performed between alpha spectrometry data and accelerator mass spectrometry data to validate the mass spectrometry techniques, and (2) in the case of suspected uptakes or for follow-up, mass spectrometry was used because it typically has a detection limit 10 times to as much as 100 times lower than alpha spectrometry. Where two positive

Table 5-6. Routine in vitro sampling frequencies.[a,b]

Radionuclide	Period	Building	Measurement type	Frequency
Gross alpha, beta (LSC)	2000–present	Sitewide	Urine	As applicable
HTO	1989[c]	Sitewide	Urine	Weekly, monthly
HTO	2000–present	Sitewide	Urine	Quarterly, monthly, biweekly, or weekly
Workplace-specific	1966[c]	101	Urine	Quarterly, annual
Workplace-specific	1966[c]	102	Urine	Annual
Workplace-specific	1966[c]	110	Urine	Annual
Workplace-specific	1966[c]	112	Urine	Annual
Workplace-specific	1966[c]	114C	Urine	Annual
Workplace-specific	1966[c]	117	Urine	Quarterly, annual
Workplace-specific	1966[c]	121	Urine	Semiannual
Workplace-specific	1966[c]	125	Urine	Semiannual
Workplace-specific	1966[c]	127	Urine	Semiannual
Workplace-specific	1966[c]	170	Urine	Annual
Workplace-specific	1966[c]	171	Urine	Annual
Workplace-specific	1966[c]	172	Urine	Weekly, monthly
Workplace-specific	1966[c]	173B	Urine	Annual
Workplace-specific	1966[c]	190	Urine	Semiannual, monthly
Workplace-specific	1966[c]	193	Urine	Annual
P-32	1989[c]	Sitewide	Urine	Monthly
P-32, C-14 (LSC)	2000–present	Sitewide	Urine	As applicable
Pu	2000–present	Sitewide	Urine	Semiannual
TRU	2000–present	Sitewide	Urine	Annual, semiannual
U	1980[c]	231	Urine	Quarterly
U	1983[c]	251	Urine	Quarterly
U	1985[c]	Site 300	Urine	Semiannual (collected on Mondays)
U	2000–present	Sitewide	Urine	Quarterly, monthly
Weapons-grade Pu	1989[c]	Sitewide	Urine	Semiannual

a. Sources: Mansfield (2000); Balanda (1966, Appendix A); LLNL (1987a); Ozaki (1980); Wilson (1982); Leahy (1983); LRL (1961, 1964); Gibson (1985).
b. These are typical in vitro monitoring frequencies; they varied depending on workplace conditions, suspected inhalation events, and positive or suspected positive bioassay results.
c. Periods listed as a single year were taken from annual reports that did not indicate a beginning date for the stated bioassay period.

values are reported for the same sample, the higher of the two should be used. If two values are reported at the minimum detectable activity for the same sample, the lower of the two values should be used. If reported as ^{239}Pu, the alpha spectrometry value is ^{239}Pu and ^{240}Pu. In the case of accelerator mass spectrometry, the sum of the ^{239}Pu and ^{240}Pu should be used to compare to the alpha spectrometry value.

5.3.1 Analysis of Gross Alpha Measurements

Urine samples were analyzed for the presence of alpha radiation from about 1957 through 1996 [3] using three possible procedures, including “Analytical Method for Determining Gross Alpha Activity in Urine,” “Determination of Plutonium and Americium in Urine by Separation,” and “Urinalysis for Curium by Electrodeposition” (Miller 1979). The results were reported as gross alpha, but all three

involved the chemical separation of several different elements depending on the method. The records are not clear about which method was used and require some review by the dose

Table 5-7. In vivo sampling frequencies.[a,b]

Radionuclide	Period	Building	Measurement type	Frequency
As detected	2000–present	Sitewide	WB scans	Annual
I-131	1989-1999[c]	Sitewide	Thyroid	Monthly
I-131	2000–present	Sitewide	Thyroid count	Coordinated with work schedule
MFPs	1964–present	Sitewide	WB scans	Not available
Pu, Am	1970–present	Sitewide	Lung count	Annual
Pu, weapons-grade Pu	1989[c]	Sitewide	Lung	Annual
Pu-239	1961–present	Sitewide	Wound	Incident-specific
U	1985[c]	Site 300	Lung count	Annual
U, Th	1987	Bldg. 321	Lung count	Annual
U, Th	2000–present	Sitewide	Lung count	Annual, semiannual

a. Sources: Mansfield (2000); Balanda (1966, Appendix A); LLNL (1987a); Ozaki (1980); Wilson (1982); Leahy (1983); LRL (1961, 1964); Gibson (1985).
b. These are typical in vivo frequencies; they varied depending on workplace conditions, suspected inhalation events, and positive or suspected positive bioassay results.
c. Periods listed as a single year were taken from annual reports that did not indicate a beginning date for the stated bioassay period.

reconstructor. The methods were capable of separating uranium from the sample before analysis to reduce the potential interference (Miller 1979).

The three urine procedures involving alpha (not including uranium) are described below:

1. "Analytical Method for Determining Gross Alpha Activity in Urine" (Miller 1979, p. 43). This method involved bismuth phosphate coprecipitation and then coprecipitation with lanthanum fluoride. Thorium, americium, curium, actinium, and neptunium are carried through to determination with this procedure. This procedure also captured plutonium.

2. "Determination of Plutonium and Americium in Urine by Separation" (Miller 1979, p. 7). This method was specific for plutonium and americium. It involved precipitation with ammonium hydroxide, dissolution in nitric, and then adsorption on an anion exchange column. The americium did not stick on the column so it came right through with the wash. If gross alpha was to be measured as well as the plutonium, the effluent was counted on a proportional counter. The plutonium was eluted from the column, electroplated, and counted in a scintillation counter.

3. "Urinalysis for Curium by Electrodeposition" (Miller 1979, p. 32). This method had the same first steps as the first procedure above, but included a third precipitation using lanthanum hydroxide. This separated curium from all the other alpha emitters. The precipitate was electroplated, then counted using pulse-height analysis, low-background proportional counting, or autoradiography.

The gross alpha procedure described above was used until around mid-2002.

Notations in bioassay records indicate that in mid-2002 the LLNL gross alpha procedure was significantly changed to increase the sensitivity and usefulness of the method. A sequential procedure in which plutonium was first separated from the sample by anion exchange chromatography, followed by separation of the remaining actinides from the sample matrix by extraction chromatography, is now employed. The resulting sample is then analyzed by alpha spectroscopy.

From mid-2002 to late 2003, bioassay records contained an analyte termed “GA-#.” These samples, indicated by GA-1, GA-2, or GA-3, represent low, middle, and high alpha energies and were reported along with a total gross alpha that represented the sum of activity in all three regions. In late 2003, the low- and middle-energy regions (GA-1 and GA-2) were combined into a single region labeled GA-1. The new GA-1 region spanned alpha energies from about 3.9 MeV to 5.1 MeV, and would normally include ^{232}Th, ^{238}U, ^{235}U, ^{234}U, and ^{237}Np. The GA-3 region spanned alpha energies from about 5.3 MeV to 6.1 MeV, and would normally include ^{241}Am, ^{228}Th, ^{244}Cm, ^{249}Cf, and some higher energy naturally occurring radionuclides [4]. As stated previously, plutonium was separated from these “GA-X” samples.

Claims that have "GA-1", "GA-2", and "GA-3" bioassay results are to be assessed on a case by case basis. The dose reconstructor should contact the site lead for any claim that has "GA-1", "GA-2", and "GA-3" bioassay results for further guidance.

The method described below applies only to gross alpha results prior to the “GA-X” analyte method implemented in mid-2002.

Gross alpha results, prior to the “GA-X” analyte method in mid-2002, were a mix of procedures 1 and 2 above. The database does not indicate which procedure was used. Sometimes the gross alpha analysis might have been specific for americium, other times it included natural thorium and plutonium. Separated americium was used in Building 251 as early as 1955 for tracer studies. Separated americium is assessed as ^{241}Am solubility Type M. To account for either possibility [5]:

1. If a worker has gross alpha and plutonium results on the same day (i.e., one sample analyzed sequentially):

    a. Assess the plutonium results with any other plutonium results that might have been collected (regardless of whether they were collected on the same day as a gross alpha sample).

        i. Assign the plutonium mixture to the plutonium results, and assign the dose.

    b. Assume there is no plutonium in the gross alpha result and assess it as ^{241}Am, solubility type M, and assign the dose.

2. If there is a gross alpha result without a plutonium result on the same day, assume it contains both plutonium and americium:

    a. If there are also plutonium results for the individual:

        i. Assess the plutonium results first for solubility types M, S, and SS:

            (1) Assess missed dose.
            (2) Assess fitted dose.
            (3) Assign the plutonium mixture to the plutonium results.
            (4) Compare missed versus fitted dose on an annual basis and assign the larger of the two in each year.

        ii. Assess the gross alpha results:

            (1) If there are gross alpha results <MDA, assess missed dose:

(a) Subtract the plutonium MDA from the gross alpha MDA at the time of the last gross alpha sample.

(b) Use the adjusted sample result to assess ^{241}Am solubility type M missed dose (i.e., use one-half the result to assess a chronic intake).

(2) If there are gross alpha results >MDA, assess fitted dose:

(a) If there are plutonium results >MDA:

(i) Run a projection from the plutonium intakes calculated in 2.a.i.2 to the dates of the positive gross alpha results. Use whichever solubility type was determined to give the largest dose in that step.

(ii) Subtract the projected plutonium results from the gross alpha results and assess the remainder as ^{241}Am solubility type M.

(b) If there are no plutonium results >MDA, do not adjust the gross alpha results; use them as reported and assess as ^{241}Am solubility type M.

(3) Compare missed versus fitted dose on an annual basis and assign the larger of the two in each year.

3. If there are gross alpha results and no plutonium results for the individual:

a. If the work location is Building 251:

i. Assess the gross alpha results twice, once assuming plutonium and once assuming ^{241}Am solubility type M:

(1) Assign the plutonium mixture to the plutonium results.

(2) Compare the plutonium mixture doses (solubility types M, S, and SS) and the the ^{241}Am solubility type M doses. Select whichever yields the largest dose to the organ of interest.

b. If the work location is not Building 251 or unknown, assess the gross alpha results assuming plutonium:

i. Assign the plutonium mixture to the plutonium results.

Exceptions occur if there is something in the file that shows the worker has been exposed to neptunium or curium. Another possible exception would involve thorium. LLNL performed some machining of natural thorium. Thorium machining was done in either enclosures/hoods or close-capture ventilation systems. Natural thorium, along with DU and NU, was also used at Site 300 in high explosive tests. Indications are that WB counting was used to monitor for thorium intakes. If a worker was getting regular uranium analyses, and then had a gross alpha bioassay analysis, this suggests that the worker had been working with the uncontained thorium and the gross alpha result should be interpreted as thorium. The worker's job description should be evaluated to determine if

this is appropriate. If there is a WB count, the thorium intake calculated from the gross alpha urinalysis could be compared with the detection limit of the WB count.

5.3.2 Analysis of Gross Beta Measurements

Urine samples were also analyzed for beta-emitting radionuclides. The gross beta procedure included some separation chemistry that removed ^{40}K (ORAUT 2006a) before counting. The procedure included a broad spectrum of other beta-emitting radionuclides that would have been captured and counted, although the percent recovery might have differed among the elements. Radioelements likely to have been captured and counted would have been strontium, barium, radium, all the rare earths, all the actinides, and most of the transition metals unless they form strong amine complexes (such as cobalt, nickel, copper, zinc, and silver). This means that beta-emitting progeny of thorium and uranium were likely to be included in the gross beta results. Ruthenium-103 and -106 were likely captured and counted (ORAUT 2006a). LLNL used other methods to separate ^{32}P, ^{14}C, and ^{90}Sr and listed the results for these specific nuclides in the results. A result described as gross beta represented many potential nuclides. Where gross beta samples were reported in cpm, the dose reconstructor should assume a conversion of 0.4 cpm/dpm (ORAUT 2013).

There are three different source terms for beta/gamma-emitting radionuclides:

- Fission and activation products from a reactor (through 1980),
- The residual from weapons tests (through 1992), or
- Aged fission products and research-related radionuclides (after 1992).

Note that for intakes before 1981 both reactor mixtures through 1980 and weapons residue mixtures should be considered. The mixture that gives the highest dose to the organ of interest should be assigned for the relevant work periods. From 1981 through 1992, only weapons residue should be assumed for the gross beta results. Reactor mixtures should only be assumed through 1980 because the Livermore Pool-Type Reactor (LPTR), which began operations in December of 1957, was permanently shut down on March 31, 1980.

Dosimetry records may contain whole-body counter (WBC) information for a worker. WBC information can be used to limit the intakes determined by the three gross beta methods listed below. This can be done by multiplying the gross beta intakes determined using the three gross methods listed below by the ratio of the specific WBC radionuclide intake to the specific gross beta radionuclide intake.

5.3.2.1 Reactor Mixtures (1974–1980)

ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2015), discusses how to use gross beta urine bioassay results to assign intakes of fission and activation products. Of the three types of gross beta analysis discussed in OTIB-0054, the gross beta analysis at LLNL is considered to be best described as "Gross beta analysis that includes all fission and activation products except radioiodines and beta emitters naturally in urine such as ^{40}K" (see Section 7.1.3 of OTIB-0054). The fission and activation product decay times assessed should be 10-day-old, 40-day-old, 180-day-old, and 1-year-old reactor fuel (Table 7-1a in OTIB-0054). Therefore, the following method is used to assign internal dose from gross beta results from reactor mixtures:

1. Determine the ^{90}Sr intake by multiplying the gross beta results by the ^{90}Sr average fraction of beta activity in urine sample for 10-day-old, 40-day-old, 180-day-old, and 1-year-old reactor fuel for the ATR, N Reactor, and TRIGA reactor types in Table 7-1a in OTIB-0054.

2. Determine the associated radionuclides, excluding ^{131}I, using the ratios for estimating intakes for intakes based on intake activity relative to ^{90}Sr for 10-day-old, 40-day-old, 180-day-old, and 1-year-old reactor fuel for the ATR, N Reactor, and TRIGA reactor types in Tables 7-3a through 7-3c, and Tables 7-3f through 7-3i in OTIB-0054.

3. Assign the highest dose determined for the above decay times.

5.3.2.2 Weapons Residue (1974–1992)

A significant amount of the work involving beta-emitting radionuclides that would have been detected by the gross beta procedure involved analysis of soil cores from the Nevada Test Site (NTS) (ORAUT 2007b). The fission and activation product mixtures from atomic weapons testing generally were not the same as those for reactor fuel.

Some results of analyses of drill-back core samples from NTS test shots are documented in Hicks (1981) in terms of activity per square meter. Results from five shots were analyzed to determine the more abundant contributors to beta exposure in the core samples. The five shots analyzed were SEDAN (1962), SCHOONER (1968), SULKY (1964), CABRIOLET (1968), and PALANQUIN (1965) (ORAUT 2007b). All were underground shots. The cores usually arrived within a few days, as opposed to months, after the shot [6]. Exposure to workers depended not only on the age of the samples being worked but also, presumably, from some build-up of contamination in hoods and ventilation systems. Therefore, exposure was probably a mix of contamination from current cores and older contamination from previous cores. For that reason, the analysis compared ratios for 100 days after each shot as being representative of the age of contamination accessible to the LLNL workers (ORAUT 2007b).

The analysis revealed that even among these five shots, all taken in the 1960s, the radionuclides of greatest abundance were not consistent. Some shots showed a high abundance of activation products and others showed mostly fission products. For the SEDAN, SCHOONER, AND CABRIOLET shots, isotopes of tungsten were most predominant, especially ^{185}W. Iron-55 was also noticeably present in the SCHOONER and CABRIOLET shots. In the SULKY and PALANQUIN shots, these radionuclides were minor with the usual list of fission products dominating (e.g., ^{91}Y, ^{89}Sr, ^{95}Nb, ^{95}Zr, ^{144}Ce, ^{103}Ru, and ^{106}Ru). When the activity per square meter values were totaled for the five shots and presented as a fraction of the total activity, the fractions of the more significant radionuclides were as follows:

- ^{185}W, 0.58
- ^{181}W, 0.36
- ^{103}Ru, 0.0070
- ^{188}W, 0.0064
- ^{91}Yt, 0.0051
- ^{55}Fe, 0.0046
- ^{95}Nb, 0.0044
- ^{89}Sr, 0.0041
- ^{141}Ce, 0.0036
- ^{95}Zr, 0.0033
- ^{106}Ru, 0.0022
- ^{7}Be, 0.0019
- ^{54}Mn, 0.0017
- ^{144}Ce, 0.0013
- ^{58}Co, 0.0013

Tungsten-181 is not a beta emitter and produces little organ dose per unit activity in the organ. The same is true for ^{7}Be (ORAUT 2007c). It is unlikely the tungsten isotopes or ^{7}Be were source terms for workers other than for the NTS core analysis program. For example, they were not likely sources for reactor workers or accelerator workers.

Ruthenium-103 is the third most abundant radionuclide in the five NTS shots, is fairly volatile, and would apply well to the reactor workers. Its half-life compares reasonably well to that of ^{185}W (39 versus 75 days, respectively) (Kocher 1981), and it produces higher dose in all organs (except ET1) per unit activity in urine than ^{185}W (ORAUT 2007c). It also produces higher dose per unit intake in all organs than ^{181}W and ^{7}Be. Overall, considering abundance, biokinetics, dose per unit activity in urine, and universal application, ^{103}Ru was considered a reasonably representative beta-emitting radionuclide for all five shots. However, some of the other major radionuclides concentrate in different organs. Because there was a mix of radionuclides in any one sample and there was a change in the mix of predominant radionuclides from core to core, it was considered important not to assume that the gross beta urinalysis values were represented by a single radionuclide.

To determine a representative mix, consideration was given to how the major radionuclides concentrate in organs. For instance, barium, beryllium, strontium, and niobium are bone seekers; cerium concentrates mostly in the liver and to some extent bone; yttrium deposits primarily in bone and to some extent the liver; zirconium deposits half to bone and half to all other tissues; ruthenium and iron are mostly distributed evenly to all tissues. Based on all the above considerations, it is recommended that the intake be estimated as consisting of 50% type F ^{103}Ru, 25% type S ^{141}Ce, and 25% type F ^{89}Sr. These values should be used for 1974 through 1992. In terms of activity in urine at equilibrium from chronic exposure, 59% is from ^{103}Ru and 41% is from ^{89}Sr. The ^{141}Ce contribution would be negligible. Beta yield per disintegration from both ^{103}Ru and ^{89}Sr is 100%, so no adjustment is necessary for beta yield. Additional intakes of type S ^{141}Ce and type F ^{89}Sr should be included at 50% each of the ^{103}Ru intake. Tungsten-181 is not a beta emitter, so it would not have been detected in the gross beta urinalysis; however, the additional dose to each organ from using ^{103}Ru accounts for the unmeasured intake (or dose) from ^{181}W. Therefore, the following method is used to assign internal dose from gross beta results from weapons residue:

1. Multiply the gross beta results by 0.59 to obtain the ^{103}Ru contribution.

2. Use the adjusted results to calculate a type F ^{103}Ru intake.

3. Assign additional intakes relative to the ^{103}Ru intake:

    a. Type S ^{141}Ce intake = 0.5 × ^{103}Ru intake, and

    b. Type F ^{89}Sr intake = 0.5 × ^{103}Ru intake.

5.3.2.3 Aged Fission Products and Research-Related Radionuclides (Post 1992)

If specific beta bioassay results are available (i.e., ^{90}Sr, ^{32}P, ^{137}Cs, or ^{14}C), the dose reconstructor should use that information for determining the internal dose.

If no specific beta bioassay results are available, intakes should be modeled as type F ^{90}Sr using 55% of the gross beta result. Additional intakes of ^{137}Cs, ^{32}P, ^{14}C, and ^{90}Y should be included, each equal to the ^{90}Sr intakes. This was based on the following methodology. For after 1992, the radionuclides in Attachment A, which apply to 1992 to 2001, were qualitatively used. Of the radionuclides in Attachment A that would have been measured by the gross beta procedure, ^{90}Sr and ^{137}Cs were considered representative. They tended to be among the more significant radionuclides in many of the beta-exposure facilities, and one is a bone seeker while the other is a WB seeker. In terms of

activity in urine at equilibrium from chronic exposure, 38% is from ^{90}Sr and 62% is from ^{137}Cs. However, beta yield per disintegration from ^{90}Sr is 200% because the ^{90}Y beta would have been counted as well. The beta yield from ^{137}Cs is 100%, so ^{90}Sr would have constituted 55% of the total beta counts and ^{137}Cs would have constituted 45% of the total beta counts. Therefore, the gross beta results should be assessed based on 55% type F ^{90}Sr. The ^{137}Cs intake is assumed to be equal to the ^{90}Sr intake as favorable to the claimant. In addition, the listing in Attachment A for the waste operations buildings (513, 514, 514 tank farms, 613, and 625) showed that ^{32}P, and ^{14}C also constituted significant fractions of the total activity. To account for these sources, the activity intake of ^{32}P and ^{14}C is assumed to be equal to the ^{90}Sr intake. Yttrium-90 was undoubtedly present with any intake of ^{90}Sr, and it is not known if it was accounted for as part of the reported activity of ^{90}Sr. Therefore, an intake of ^{90}Y equal to the intake of ^{90}Sr should be included. The following method should be used to assign internal dose from gross beta results from aged fission products and research-related radionuclides:

1. Determine the type F ^{90}Sr intake assuming 55% of the gross beta results.

2. Type F or M ^{32}P intake = ^{90}Sr intake.

3. Type F, M, or S ^{14}C intake = ^{90}Sr intake.

4. Type F ^{137}Cs intake = ^{90}Sr intake.

5. Type F ^{90}Y intake = ^{90}Sr intake.

5.3.2.4 Analysis of Tritium Measurements

External dose records for LLNL might contain tritium doses without actual tritium bioassay records for a worker from 1958 through 1961. The following method should be used when assessing internal tritium dose for a worker.

1. If there are reported positive or zero tritium doses in the external dosimetry records from 1958 through 1961, but no bioassay results for the worker, the following should be done:

    a. If all tritium doses are zero, do not report any tritium dose for these time periods, and assign environmental internal tritium dose.

    b. If any of the tritium doses are positive, report the positive doses for these periods in the output from the Interactive RadioEpidemiological Program (IREP). Assign the positive tritium doses as a lognormal distribution with a GSD of 3.0.

2. If there are tritium bioassay results for the worker:

    a. Assign tritium dose for years for which there is tritium bioassay. For remaining years, assign only environmental internal tritium dose.

    b. If there is no tritium bioassay, excluding (1.b) above, assign environmental internal tritium dose.

5.3.3 Measurement Types and Detection Levels

A variety of program documents report detection levels for the various in vivo and in vitro bioassay methods. These methods are listed in Tables 5-8 and 5-9, and the documents are cited in the table footnotes.

Table 5-8. Bioassay in vitro detection levels.[a]

Measurement type	Analysis method	Selected radionuclide[d]	Period	MDA
Fecal	Radiometric alpha spectrometry	Pu-239, 238	1977–1980	1 nCi/sample
Fecal	Coprecipitation and proportional counting	Pu-239, 238	1990–1999	0.04 dpm
Urine	Coprecipitation, separation (gross alpha method)	Am	1957–1989	0.718 dpm/sample
Urine	Coprecipitation, separation (gross alpha method)	Am	1990–1999	0.365 dpm/sample
Urine	24-hr sample, alpha spec	Am-241	2000–present	0.01 dpm
Urine	Liquid scintillation	C-14	2000–present	0.002 µCi/L
Urine	Electrodeposition and proportional counting	Cm-244	1960–1999	0.2 dpm
Urine	24-hr sample, alpha spec	Cm-244	2000–present	0.01 dpm
Urine	Liquid scintillation	HTO	1955–1978	1 µCi/L
Urine	Liquid scintillation	HTO	1979–present	0.01 µCi/L
Urine	Liquid scintillation	I-125, I-131, S-35	2000–present	0.002 µCi/L
Urine	24-hr sample, alpha spec	Np-237	2000–present	0.01 dpm
Urine	Liquid scintillation	P-32, P-33	2000–present	0.002 µCi/L
Urine	Extraction and gross alpha[b]	Po-210	1955–1978	0.1 pCi/L
Urine	24-hr sample, alpha spec	Po-210	2000–present	0.01 dpm
Urine	Coprecipitation	Gross alpha	1957–1989	0.78 dpm/sample
Urine	Coprecipitation	Gross alpha	1990–1999	0.365 dpm/sample
Urine	Coprecipitation, separation, and proportional counting	Pu-total alpha	1957–1989	0.051 dpm/sample
Urine	Coprecipitation, separation, and proportional counting	Pu-total alpha	1990–1999	0.012 dpm/sample[f]
Urine	24-hr sample, alpha spec	Pu	2000–present	0.01 dpm
Urine	Gross beta	MFP	1957–1989	600 dpm/L[g]
Urine	Gross beta	MFP	1990–1999	32 dpm/L
Urine	TRU-spec column with alpha spectrometry	Th-232, Th-228	1989–present	0.2 dpm/sample (with Th-232 being equivalent to Th-228)
Urine	Fluorometric	DU, NU, U AVLIS	1958–1973	5 µg/L
Urine	Fluorometric	DU, NU, U AVLIS	1974–1978	1 µg/L
Urine	Fluorometric	DU, NU, U AVLIS	1979–1989	0.3 µg/L
Urine	Spot sample KPA	DU, NU, U AVLIS	1990–1999	0.05 µg/L
Urine	Spot sample ICP-MS	U	2000–present	0.002 µg/L
Urine	Extraction/alpha proportional counting (U-234 alphas measured)[b]	HEU	1955–1970	100 dpm/L[e]
Urine	Extraction/alpha proportional counting[b]	HEU	1971–1976	15 dpm/L
Urine	Extraction/alpha proportional counting[b]	HEU	1977–1986	4 pCi/L

Measurement type	Analysis method	Selected radionuclide[d]	Period	MDA
Urine	Alpha spectrometry[c]	HEU	1987–1999	0.3 pCi/L

a. Sources: Mansfield (2000); Dupzyk and Biggs (1960); LRL (1964, 1970); Griffith (1980); ORAUT (2005c); Schmidt and Anderson (1965); ORAUT (2013, 2007a); LLL ca. 1978; Miller (1979); Dupzyk (1990); Simpson (1989), Author unknown (2006).
b. Based on information from Los Alamos National Laboratory (LANL).
c. Based on ICRP (1989) information.
d. Other nuclides might have been detected by the method; these might be reported in worker files.
e. Based on information from LANL. Decision level of 50 dpm/L reported. Assume the MDA is 2 times the decision level.
f. MDA range of 0.012 to 0.051 dpm/sample was indicated. Dose reconstructors should use available sample-specific information.
g. Gross beta method MDA was 3.84 cpm/sample with a detector efficiency range of 20% to 31% and sample recovery of 60%. This results in an MDA range of approximately 400 to 600 dpm/sample depending on detector efficiency (Miller 1979). Dose reconstructors should use available sample-specific information.

Table 5-9. Bioassay in vivo detection levels.[a]

Measurement type	Analysis method	Radionuclide	Period	MDA
Lung	Phoswich[b]	Pu-239, Am-241	1966–1969	400 nCi, 2 nCi
Lung	Phoswich[c]	Pu-239, Am-241	1970–1979	54 nCi, 0.5 nCi
Lung	Phoswich	Pu-239	Circa 1978	19 nCi
Lung	Phoswich	Am-241	Circa 1979	0.3 nCi
Lung	Phoswich	Am-241	1984–1985	0.1–0.3 nCi (2.3–5.25 cm CWT)
Lung	Dual phoswich	Pu-239	1980–present	>16.3 nCi (2.1 cm CWT)
Lung	Dual phoswich	“Pure” Pu-239	1984–1985	9–200 nCi (1.45–5.95 cm CWT)
Lung	HPGe arrays	“Pure” Pu-239	2000–present	200 nCi
Lung	HPGe arrays	“Pure” Pu-238	1984–1985	4–100 nCi (1.5–4.1 cm CWT)
Lung	ACTII	“Pure” Pu-238	1999	433 nCi (3.33 cm CWT)
Lung	HPGe arrays	“Pure” Pu-238	2000–present	150 nCi
Lung	ACTII	“Pure” Pu-238	2006	87 nCi (3.6 cm CWT)
Lung	ACTII	Weapons-grade Pu	1984–1985	1–4 nCi (1.45–5.85 cm CWT)
Lung	HPGe arrays (Am-241)	Weapons-grade Pu	2000–present	0.15 nCi
Lung	ACTII	Am-241	1999	0.14 nCi (3.33 cm CWT)
Lung	HPGe arrays	Am-241	2000–present	0.15 nCi
Lung	ACTII	Am-241	2006	0.11 nCi (3.6 cm CWT)
Lung	ACTII	DU, NU	Circa 1977	5 nCi
Lung	ACTII	DU	1984–1985	0.5–2 nCi
Lung	ACTII	NU	1984–1985	0.7–2 nCi
Lung	ACTII	U-235	1984–1985	~100 µg (depending upon enrichment)
Lung	ACTII	U-238 (Th-234)	1999	1.27 nCi (3.33 cm CWT)
Lung	HPGe arrays	DU, NU	2000–present	1 nCi
Lung	ACTII	U-238 (Th-234)	2006	0.68 nCi (3.6 cm CWT)
Lung	ACTII	U-235	1999	0.07 nCi (3.33 cm CWT)
Lung	HPGe arrays	AVLIS U, Th, U-235	2000–present	1 nCi
Lung	ACTII	U-235	2006	0.05 nCi (3.6 cm CWT)
Lung	HPGe arrays	40% U, 95% U	2000–present	0.1 nCi
Lung	HPGe arrays	Cm-244	2000–present	100 nCi
Lung	HPGe arrays	Np-237	2000–present	0.35 nCi
Lung	ACTII	Np-237	2006	0.3 nCi (3.6 cm CWT)
Lung	HPGe arrays	Pa-233	2000–present	1 nCi
Lung	ACTII	Pa-233	2006	0.13 nCi (3.6 cm CWT)
Thyroid count	ACTII	I-131	1984–1985	1 nCi
Thyroid count	ACTII	I-131	2000–present	0.01 nCi
Thyroid count	ACTII	I-131	2006	0.04 nCi

Measurement type	Analysis method	Radionuclide	Period	MDA
Thyroid count	Portable NaI	I-131	2006	0.27 nCi
Thyroid count	Portable NaI	I-125	2000–present	0.02 nCi
Thyroid count	ACTII	I-125	2006	0.31 nCi
Thyroid count	Portable NaI	I-125	2006	0.02 nCi
WB scan	Shadow shield counter	K-40	1964–1995	150 g K in 20 min (~120 nCi K-40)
WB scan	Shadow shield counter	150- to 300-keV gamma emitters	1964–present	500 gammas per second between 150 and 300 keV
WB Scan	Shadow shield counter	Th-232	1970–present	1 nCi (based on Tl-208 with no correction for branching ratio)
WB scan	Shadow shield counter	Natural Th (Tl-208)	2000–present	1 nCi
WB scan	Shadow shield counter	I-131 (TB)	1970–1995	1 nCi
WB scan	Shadow shield counter	Cs-137	1970–1995	1 nCi
WB scan	Shadow shield counter	Co-60	1970–1995	1 nCi
WB scan	Shadow shield counter	Ce-144	1970–present	8 nCi
WB scan	Shadow shield counter	Ce-141	1970–present	2 nCi
WB scan	Shadow shield counter	Ru-103	1970–present	3 nCi
WB scan	Shadow shield counter	Ru-106	1970–1995	5 nCi
WB scan	Shadow shield counter	Zr-95	1970–present	1 nCi
WB scan	Shadow shield counter	Mn-54	1970–1995	1 nCi
WB scan	Shadow shield counter	Zn-65	1970–present	1 nCi
WB scan	Shadow shield counter	Na-22	1970–1995	1 nCi
WB scan	Shadow shield counter	K-40	1996–1999	4.46 nCi
WB scan	Shadow shield counter	Mn-54	1996–1999	0.63 nCi
WB scan	Shadow shield counter	Fe-59	1996–1999	0.98 nCi
WB scan	Shadow shield counter	Co-60	1996–1999	0.66 nCi
WB scan	Shadow shield counter	Ru-106	1996–1999	6.53 nCi
WB scan	Shadow shield counter	I-131	1996–1999	1 nCi
WB scan	Shadow shield counter	Cs-134	1996–1999	0.7 nCi
WB scan	Shadow shield counter	Cs-137	1996–1999	0.89 nCi
WB scan	Shadow shield counter	Co-57	2000–2005	0.97 nCi
WB scan	Shadow shield counter	Co-60	2000–2005	0.51 nCi
WB scan	Shadow shield counter	Cs-137	2000–2005	0.7 nCi
WB scan	Shadow shield counter	I-131	2000–2005	0.66 nCi
WB scan	Shadow shield counter	K-40	2000–2005	2.6 nCi
WB scan	Shadow shield counter	Na-22	2000–2005	0.44 nCi
WB scan	Shadow shield counter	K-40	2006	2.6 nCi
WB scan	Shadow shield counter	Cs-137	2006	0.72 nCi
WB scan	Shadow shield counter	Co-60	2006	0.43 nCi
WB scan	Shadow shield counter	I-131	2006	0.67 nCi
WB scan	Shadow shield counter	Na-22	2006	0.42 nCi
WB scan	Shadow shield counter	Pa-233	2006	1.5 nCi
WB scan	Shadow shield counter	Ac-228	2007–present	2.74 nCi
WB scan	Shadow shield counter	Ba-133	2007–present	0.85 nCi
WB scan	Shadow shield counter	Bi-207	2007–present	0.47 nCi
WB scan	Shadow shield counter	Bi-212	2007–present	7.71 nCi
WB scan	Shadow shield counter	Bi-214	2007–present	1.82 nCi
WB scan	Shadow shield counter	Cf-249	2007–present	0.83 nCi
WB scan	Shadow shield counter	Cm-243	2007–present	3.59 nCi
WB scan	Shadow shield counter	Co-57	2007–present	0.97 nCi
WB scan	Shadow shield counter	Co-60	2007–present	0.37 nCi
WB scan	Shadow shield counter	Cs-134	2007–present	0.56 nCi
WB scan	Shadow shield counter	Cs-137	2007–present	0.7 nCi
WB scan	Shadow shield counter	Eu-152	2007–present	1.75 nCi

Measurement type	Analysis method	Radionuclide	Period	MDA
WB scan	Shadow shield counter	Eu-154	2007–present	1.25 nCi
WB scan	Shadow shield counter	Eu-155	2007–present	2.58 nCi
WB scan	Shadow shield counter	I-131	2007–present	0.68 nCi
WB scan	Shadow shield counter	K-40	2007–present	2.52 nCi
WB scan	Shadow shield counter	Na-22	2007–present	0.4 nCi
WB scan	Shadow shield counter	Np-237	2007–present	6.38 nCi
WB scan	Shadow shield counter	Np-239	2007–present	4.73 nCi
WB scan	Shadow shield counter	Pa-231	2007–present	21.83 nCi
WB scan	Shadow shield counter	Pa-233	2007–present	1.6 nCi
WB scan	Shadow shield counter	Pa-234	2007–present	3.93 nCi
WB scan	Shadow shield counter	Pb-212	2007–present	2.23 nCi
WB scan	Shadow shield counter	Pb-214	2007–present	2.36 nCi
WB scan	Shadow shield counter	Po-210	2007–present	3.40E+04 nCi
WB scan	Shadow shield counter	Ra-224	2007–present	20.6 nCi
WB scan	Shadow shield counter	Ra-226	2007–present	23.9 nCi
WB scan	Shadow shield counter	Th-231	2007–present	11.6 nCi
WB scan	Shadow shield counter	Th-234	2007–present	15.8 nCi
WB scan	Shadow shield counter	U-235	2007–present	1.61 nCi

a. Sources: Mansfield (2000); LRL (1964, 1970); Schmidt and Anderson (1965); Griffith (1980); ORAUT (2005c, 2013, 2007a); LLL (ca. 1978); Anderson, Campbell, and Griffith (1979); King (1977); Author unknown (ca. 1985); Sundsmo and Hickman (2009a, 2009b, 2009c); Hickman (2008).
b. Based on LANL information.
c. Based on ICRP (1989) information.

In some cases, there is evidence that LLNL implemented monitoring using a particular methodology, but site-specific detection levels for that methodology were not available. In these cases, detection levels for comparable LANL methods were used to complete Tables 5-8 and 5-9 because there is evidence of interlaboratory communications on analytical methods over the years, and because both LLNL and LANL were operated by the same contractor, which increases the likelihood for shared procedures (ORAUT 2005c).

In a few instances, neither LLNL nor LANL detection levels were available, requiring a surrogate source of information. ICRP Publication 54 (ICRP 1989) specified nominal detection levels for a variety of measurement methods that were comparable to those used at LLNL.

A review of in vivo bioassay records provided by LLNL indicates that the data typically include the assessed radionuclide, MDA, activity result, height, weight, and chest wall thickness (CWT). In most cases, the dose reconstructor will have all the information necessary to assess the internal dose from in vivo bioassay data. In the absence of specific in vivo MDA information in the worker's dosimetry records, the dose reconstructor should refer to Table 5-9. The expected intake pattern in most cases is an acute intake. At LLNL, airborne and surface contamination was typically controlled to prevent intakes, so most intakes would have been the result of unexpected releases. Small intermittent releases that were not immediately detectable could have occurred, so an individual could have had multiple acute intakes.

5.3.4 Reporting Formats and Codes

There are a variety of entries provided in LLNL urine bioassay records. These bioassay records exist in several bioassay databases named MAPPER, SYMPHONY, and BLIMS used by LLNL throughout its history. Table 5-10 provides a lisiting of the bioassay information reported in the ouputs of these databases provided in claim dosimetry record files.

Table 5-10. Bioassay database record entries.

Type of entry	Entry description
Sample number	Number assigned to sample by Bioassay Laboratory
Employee number	6-digit integer
Employee name	Last name, comma, other names and/or initials
Nuclide analyzed	MFP, ALPHA, PU239, PU238, U238, AM241, CM244, or other nuclide
Sample date	Month, day, and year
Sample result	Floating point number or exponential number
Sample error	One standard deviation as a percentage of the result
Sample media	U = urine, B = blood, f = feces, n = nose swipe
Sample type	R = routine; S = special
Result units	Dpm, dpm/L, dpm/sample, µg/L, µCi/L,µCi/sample, or other unit
Sample volume or mass	ml, or g
Decision level	Decision level
MDA	Minimum detectable activity

5.3.5 Assessment of Intake for Unmonitored Workers

LLNL had an established in vitro and in vivo bioassay sampling program, and there was clear direction to supervisors on individual employee participation in the program. In the case where a worker might not have been monitored but should be assigned an intake based on job description and information in the claimant telephone interview, a means of assessing intakes for unmonitored workers is necessary. There is little correlation between external dosimetry and the bioassay program. From 1953, external dosimetry was administered as a sitewide requirement, whereas bioassay was determined based on the potential of exposure by facility supervisors and Health Physics. Dose reconstructors should refer to the coworker data in Attachment B to assign intakes for unmonitored workers. Attachment C provides guidance on assigning the 50th- or 95th-percentile coworker intakes based on job category and level of contact with the radioactive material.

5.3.6 Unmonitored Intakes of LLNL Radionuclides Other than Primary Radionuclides

A short description of the work in relation to each of the listed radionuclides follows. Most were handled as part of basic research, and the number of exposed individuals would have been small.

Actinium-227 was used in Building 251, which housed basic research in the chemical and nuclear behavior of the transuranic (TRU) elements starting in 1955. Building 251 was restricted to “an as needed basis” (LLL 1980). Given the nature of the work, access and radiological controls, and an in vitro and in vivo bioassay monitoring program in place for workers in areas where radioactive material was routinely handled, unmonitored intakes were unlikely. Starting around mid-1978, a solution of ^{227}Ac stored in a glovebox in Building 251 was “milked” periodically for its ^{223}Ra, which was used as an alpha-emitting contaminant in exercises (Gibson 1980). Actinium-227 was also stored in a cave in Room 1235 of Building 251 (Meadows, Sisson, and Hayes 1981). Given the containment and restrictive use controls, the potential for exposure to unmonitored individuals was remote. In 1992 all U.S. nuclear testing ceased. In 1993 LLNL decided to discontinue programmatic operations in Building 251 based on new regulatory requirements. Building 251 was placed in standby mode in 1995. During the standby period, Building 251 was staffed with three people. Building 251 transitioned from Category II Nonreactor Nuclear Facility in 2002 to a radiological facility in 2005 where equipment and inventory were removed (Gray et al. 2006).

Thorium-232 (natural thorium) was associated primarily with Site 300 and two other facilities, Buildings 231 and 321C. Building 231 is a large experimental, manufacturing, assembly, test, and materials-handling facility, and Building 321C machined depleted uranium (DU) and thorium. Machining only occurred in 321C. Site 300 was a high-explosive test facility that opened in mid-1955. DU, NU, and natural thorium were used with high-explosive materials tests at Site 300.

Curium-244 was associated primarily with Building 251, starting in 1955 when the building became operational. As indicated previously, access to Building 251 was restricted to "an as needed basis." Given the nature of the work, access and radiological controls, and an in vitro and in vivo bioassay monitoring program in place for workers in areas where radioactive material was routinely handled, unmonitored intakes were unlikely. Curium-244 was also used in Building 332 (Plutonium Facility) with similar controls in place. Building 332 became operational in 1961.

Neptunium-237 work was conducted primarily in Building 251 and, as stated above, unmonitored intakes were unlikely. Research on ^{237}Np was conducted in Room 1319 of Building 281, which included gloveboxes. Building 281 conducted tracer work, dissolution studies, and flow studies. As a result, the potential for internal exposure to ^{237}Np in Building 281 would have been remote.

Based on the foregoing, the potential for an unmonitored intake of the above-listed radionuclides is limited. However, it is necessary to demonstrate that maximum doses can be assigned when presumptive intakes, beyond those that might be characterized by bioassay records, are implied by case-specific information. In the absence of case-specific information, the dose from unmonitored intakes of LLNL radionuclides other than primary radionuclides should not be assigned. Based on the foregoing information on Building 251, no dose from any of the above-listed radionuclides should be assigned after 1995.

Table 5-11 lists the radionuclides to consider along with the work locations. Workers who are assigned doses from unmonitored intakes of LLNL radionuclides other than primary radionuclides based on the coworker intakes described in the table below will not be assessed for the period before 1957 for plutonium and americium, and before 1958 for uranium, because they are included in the designated SEC period.

Table 5-11. Dose assignment for radionuclides other than primary radionuclides.

Radionuclide	Building	Likely exposed individuals	Period of potential exposure	Dose estimation techniques (in order of preference)
Ac-227	251	Chemists, researchers, health physics technicians, technicians, and maintenance	1978–1995	1. Base on bioassay results. 2. Assign dose based on coworker intakes for Pu-239 assuming Ac-227 solubility types F, M, and S.
Np-237	251, 281	Chemists, researchers, health physics technicians, technicians, and maintenance	1955–1995 (Building 251) 1952–present (Building 281)	1. Base on bioassay results. 2. Assign dose based on coworker intakes for Pu-239 assuming Np-237 solubility type M.
Cm-244	251, 332	Chemists, researchers, health physics technicians, technicians, and maintenance	1955–1995 (Building 251) 1961–present (Building 332)	1. Base on bioassay results. 2. Assign dose based on coworker intakes for Am-241 assuming Cm-244 solubility type M.
Th-232 (natural thorium)	231, 321C, Site 300	Chemists, researchers, machinists, health physics technicians, and technicians	1952–present (Buildings 231, 321C) 1955–present (Site 300)	1. Base on bioassay results. 2. Assign dose assuming a natural thorium mass intake equal to the coworker uranium mass intake. Assume a natural thorium specific activity of 4.88E+02 dpm/mg to determine the total thorium activity intake. Assume 50% of the total thorium activity intake is Th-232 and 50% of the total thorium activity intake is Th-228. Assume solubility types M and S.

5.4 SIGNIFICANT INCIDENTS WITH INTERNAL DOSE POTENTIAL

During operations at LLNL, a number of incidents increased the potential for intakes of radioactive materials. If a worker recalls involvement in one or more of those incidents, dose reconstructors can use the information in Table 5-12 as input to an incident-specific assessment. This list is not all encompassing, and many other incidents probably occurred; these are the incidents identified from review of the data capture records. Individual worker records could provide documentation of involvement in other incidents.

Table 5-12. Input parameters for significant incidents.

Incident date	Incident description	Facility	Other information
03/26/1963	Criticality	Bldg. 110	Potential for internal and external exposure (I, Kr, Xe); monitored participants listed by name in 1963 reference (see below).
11/06/1975	Radium contamination	Bldg. 343, Rm. 1005	Crystal was removed from radium dial resulting in spread of contamination.
11/23/1971	Pu-contaminated squib valve	Bldg. 343	Direct and indirect bioassays performed on 14 individuals

a. Sources: LRL (ca. 1963a, 1963b); Leahy (1975); AEC (1971).

5.5 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section, with information to identify the source and justification for each associated item. Conventional References, which are provided in the next section of this document, link data, quotations, and other information to documents available for review on the Project's Site Research Database (SRDB).

[1] Szalinski, Paul A. Oak Ridge Associated Universities (ORAU) Team. Health Physicist. March 2007.
The LLNL Technical Basis Manual for Internal Dosimetry (Mansfield 2000) discusses RU in the DOE complex, so the internal dosimetrists at LLNL are aware of the issues.

[2] Thomas, Bill R. ORAU Team. Health Physicist. March 2007.
Results for gross alpha in urine were reviewed in the claimant files. The frequency of sample collection was dependent on the particular building where the worker was assigned. The frequency ranged from samples several times per year to a single sample per year.

[3] Szalinski, Paul A. ORAU Team. Health Physicist. March 2007.
In ORAUT (2006a), when asked if the procedures were in place from 1957 to 1996, an LLNL employee replied that they were, at least to 1987. Because these procedures were based on DOE Environmental Measurement Laboratory Procedures Manual HASL-300 (Cheico 1997), which did not undergo its next significant revision until 1997, it is likely that the procedures were unchanged through 1996.

[4] Thomas, Bill R. ORAU Team. Health Physicist. March 2007.
The information in this paragraph comes from Explanatory Notes on "Gross Alpha" Results in LLNL claimant files (Author unknown 2006) for Summary of Bioassay Results that were reported after approximately 2002.

[5] Szalinski, Paul A. ORAU Team. Health Physicist. March 2007.
Because the first two gross alpha analysis methods might contain the same radionuclides and there is no information as to which method was used on a given sample, the steps are given to assess the various possibilities that might be encountered.

[6] Bihl, Donald E. ORAU Team. Principal Health Physicist. April 2007.
Personal communication with M. H. Chew, October 16, 2006.

[7] LaBone, Thomas R. ORAU Team. Deputy Principal Internal Dosimetrist. February 2007.
The uranium results were recorded as µg/L and the statistical analyses were performed in those units. However, the Integrated Modules for Bioassay Analysis (IMBA) computer program requires that all excreta data be entered as total excretion per day; hence, the statistical parameters were converted to excretion per day before intake calculations were made using IMBA.

[8] Brackett, Elizabeth M. ORAU Team. Principal Internal Dosimetrist. February 2007.
The uniform absolute error of 1 weights all results equally; other fitting schemes weight high values or low values disproportionately. Because the median and 84th-percentile values were determined from statistical analysis of many samples in each interval, there was no *a priori* reason to weight results from one interval over another.

[9] LaBone, Thomas R. ORAU Team. Deputy Principal Internal Dosimetrist. February 2007. The breathing rate and particle size distribution are project default values to be used unless site-specific information indicates otherwise. No information has been found concerning intakes at LLNL that shows that the default values should not be used. See, for instance, OCAS-IG-002, *Internal Dose Reconstruction Implementation Guide* (NIOSH 2002), and ICRP Publication 66, *Human Respiratory Tract Model for Radiological Protection* (ICRP 1994).

[10] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007. The minimum geometric standard deviation (GSD) of 3 is established in ORAUT-OTIB-0060, *Internal Dose Reconstruction* (ORAUT 2014). It reflects the overall uncertainty associated with biokinetic modeling as well as usual radiochemical analysis, and indicates that even though the spread in coworker excreta results for a given population (e.g., a year of excreta samples) can have a GSD of <3, the uncertainty of intakes determined using the biokinetic models is no less than 3.

[11] LaBone, Thomas R. ORAU Team. Deputy Principal Internal Dosimetrist. February 2007. Use of the midpoint of the period represented by the excreta data point is established in ORAUT-OTIB-0019, *Analysis of Coworker Bioassay Data for Internal Dose Assignment* (ORAUT 2005a), and is standard practice for assigning chronic intakes or acute intakes with unknown dates. See, for instance, ANSI/HPS N13.39-2001, *Design of Internal Dosimetry Programs* (HPS 2001).

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GLOSSARY

absorption type
Categories for materials according to their rates of absorption from the respiratory tract into the blood, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F: deposited materials that are readily absorbed into blood from the respiratory tract (fast solubilization), M: deposited materials that have intermediate rates of absorption into blood from the respiratory tract (moderate rate of solubilization), and S: deposited materials that are relatively insoluble in the respiratory tract (slow solubilization). Also called solubility type.

activity
Amount of radioactivity. The International System unit of activity is the becquerel (1 disintegration per second); the traditional unit is the curie [37 billion (3.7×10^{10}) becquerels].

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

acute exposure
Radiation exposure to the body delivered in a short period. See *chronic exposure*.

air sampling
Collection of samples of the ambient atmosphere to detect or measure the presence of radioactive material in the air.

alpha radiation
Positively charged particle emitted from the nuclei of some radioactive elements. An alpha particle consists of two neutrons and two protons (a helium nucleus) and has an electrostatic charge of +2.

atomic vapor laser isotope separation (AVLIS)
Process for enriching ^{235}U by ionizing uranium compounds in a vapor based on a laser tuned to the electronic structure of uranium compounds.

beta radiation
Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron.

bioassay
Measurement of amount or concentration of radionuclide material in the body (in vivo measurement) or in biological material excreted or removed from the body (in vitro measurement) and analyzed for purposes of estimating the quantity of radioactive material in the body. Also called radiobioassay.

bremsstrahlung
Electromagnetic radiation released as a result of inelastic scattering of a moving charged particle within the nucleus of an atom. X-rays produced in a typical medical X-ray tube frequently originate from inelastic scattering of accelerated electrons in the anode material.

chronic exposure

Radiation dose to the body delivered in small amounts over a long period (e.g., days or years). See *acute exposure*.

claim

Petition filed by a covered employee or survivor(s) thereof and submitted to the U.S. Department of Labor for compensation under the Energy Employees Occupational Illness Compensation Program Act of 2000. There can be multiple claims filed per worker case.

claimant

Individual who has filed for compensation under the Energy Employees Occupational Illness Compensation Program Act of 2000. This individual can be the energy employee (worker), a survivor, or the legal representative of the energy employee.

contamination

Radioactive material in undesired locations including air, soil, buildings, animals, and persons.

curie (Ci)

Traditional unit of radioactivity equal to 37 billion (3.7×10^{10}) becquerels, which is approximately equal to the activity of 1 gram of pure ^{226}Ra.

depleted uranium (DU)

Uranium with a percentage of ^{235}U lower than the 0.7% found in natural uranium.

dose

In general, the specific amount of energy from ionizing radiation that is absorbed per unit of mass. Effective and equivalent doses are in units of rem or sievert; other types of dose are in units of roentgens, rad, rep, or grays.

dose equivalent

In units of rem or sievert, product of absorbed dose in tissue multiplied by a weighting factor and sometimes by other modifying factors to account for the potential for a biological effect from the absorbed dose. See *dose*.

dosimeter

Device that measures the quantity of received radiation, usually a holder with radiation-absorbing filters and radiation-sensitive inserts packaged to provide a record of absorbed dose received by an individual

electron-volt (eV)

Unit equal to the energy of one electron moving through a potential difference of 1 volt (1.602×10^{-19} joules). The common units in nuclear physics and radiology are kiloelectron-volts (thousands) and megaelectron-volts (millions).

Energy Employees Occupational Illness Compensation Program Act of 2000, as amended (EEOICPA; 42 U.S.C. § 7384 *et seq.*)

Law that provides for evaluation of cause and potential compensation for energy employees who have certain types of cancer.

exchange period (frequency)

Period (weekly, biweekly, monthly, quarterly, etc.) for routine exchange of dosimeters.

exposure

(1) In general, the act of being exposed to ionizing radiation; see *acute exposure* and *chronic exposure*. (2) Measure of the ionization produced by X- and gamma-ray photons in air in units of roentgens. In this document, exposure does not refer to the radiological physics concept of charge liberated per unit mass of air.

gamma radiation

Electromagnetic radiation (photons) of short wavelength and high energy (10 kiloelectron-volts to 9 megaelectron-volts) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma photons are identical to X-ray photons of high energy; the difference is that X-rays do not originate in the nucleus.

glovebox

Enclosure with special rubber gloves through which an operator can handle radioactive or toxic material without risk of injury or contamination, normally operated at a slightly reduced pressure so that air leakage, if any, is inward.

half-life

Time in which half of a given quantity of a particular radionuclide disintegrates (decays) into another nuclear form. During one half-life, the number of atoms of a particular radionuclide decreases by one half. Each radionuclide has a unique half-life ranging from millionths of a second to billions of years.

highly enriched uranium (HEU)

Uranium enriched to at least 20% ^{235}U for use as fissile material in nuclear weapons components and some reactor fuels. Also called high-enriched uranium.

intake

Radioactive material taken into the body by inhalation, absorption through the skin, injection, ingestion, or through wounds.

Integrated Modules for Bioassay Analysis (IMBA)

Computer program that uses bioassay results and other information to calculate intakes of radionuclides and subsequent doses.

internal dose

Dose received from radioactive material in the body.

internal dose assessment

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

in vitro bioassay

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 bioassay

Measurements of radioactive material in the human body using instrumentation that detects radiation emitted from the radioactive material in the body.

ionizing radiation

Radiation of high enough energy to remove an electron from a struck atom and leave behind a positively charged ion. High enough doses of ionizing radiation can cause cellular damage.

Ionizing particles include alpha particles, beta particles, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, photoelectrons, Compton electrons, positron/negatron pairs from photon radiation, and scattered nuclei from fast neutrons. See *alpha radiation*, *beta radiation*, and *X-ray radiation*.

isotope

One of two or more atoms of a particular element that have the same number of protons (atomic number) but different numbers of neutrons in their nuclei (e.g., ^{234}U, ^{235}U, and ^{238}U). Isotopes have very nearly the same chemical properties.

kiloelectron-volt (keV)

Unit of particle energy equal to 1,000 (1×10^3) electron-volts.

lung solubility type

See *absorption type.*

megaelectron-volt (MeV)

Unit of particle energy equal to 1 million (1×10^6) electron-volts.

minimum detectable amount (MDA)

Smallest amount (activity or mass) of an analyte in a sample that can be detected with a probability β of nondetection (Type II error) while accepting a probability α of erroneously deciding that a positive (nonzero) quantity of analyte is present in an appropriate blank sample (Type I error).

monitoring (personnel)

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

National Institute for Occupational Safety and Health (NIOSH)

U.S. agency responsible for dose reconstruction under the Energy Employees Occupational Illness Compensation Program Act of 2000. Part of the Centers for Disease Control and Prevention, which is part of the U.S. Department of Health and Human Services, NIOSH is the Federal agency responsible for conducting research and making recommendations for the prevention of work related injury and illness.

natural uranium (NU)

Uranium as found in nature, approximately 99.27% ^{238}U, 0.72% ^{235}U, and 0.0054% ^{234}U by mass. The specific activity of this mixture is 2.6×10^7 becquerel per kilogram (0.7 microcuries per gram).

occupational dose

Internal and external ionizing radiation dose from exposure during employment. Occupational dose does not include that from background radiation or medical diagnostics, research, or treatment, but does include dose from occupationally required radiographic examinations that were part of medical screening.

radiation

Subatomic particles and electromagnetic rays (photons) with kinetic energy that interact with matter through various mechanisms that involve energy transfer. See *ionizing radiation*.

rem

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

routine monitoring

Monitoring carried out at regular intervals during normal operations. See *special monitoring*.

sievert

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

Special Exposure Cohort (SEC) [42 U.S.C. § 7384l(14)]

… "member of the Special Exposure Cohort" means a Department of Energy employee, Department of Energy contractor employee, or atomic weapons employee who meets any of the following requirements:

(A) The employee was so employed for a number of work days aggregating at least 250 work days before February 1, 1992, at a gaseous diffusion plant located in Paducah, Kentucky, Portsmouth, Ohio, or Oak Ridge, Tennessee, and, during such employment—

(i) was monitored through the use of dosimetry badges for exposure at the plant of the external parts of employee's body to radiation; or

(ii) worked in a job that had exposures comparable to a job that is or was monitored through the use of dosimetry badges.

(B) The employee was so employed before January 1, 1974, by the Department of Energy or a Department of Energy contractor or subcontractor on Amchitka Island, Alaska, and was exposed to ionizing radiation in the performance of duty related to the Long Shot, Milrow, or Cannikin underground nuclear tests.

(C) (i) Subject to clause (ii), the employee is an individual designated as a member of the Special Exposure Cohort by the President for purposes of the compensation program under section 7384q of this title.

(ii) A designation under clause (i) shall, unless Congress otherwise provides, take effect on the date that is 180 days after the date on which the President submits to Congress a report identifying the individuals covered by the designation and describing the criteria used in designating those individuals.

special monitoring

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

spot sample

In relation to bioassay, a single void of urine.

whole-body counter (WBC)

Equipment used to perform in vivo bioassay. Radiation emitted from radioactive material deposited throughout the body is measured.

whole-body exposure

Exposure of the entire body to radiation rather than an isolated part. When an ingested radioisotope is uniformly distributed throughout the body tissues rather than being concentrated in certain parts, the irradiation is a whole-body exposure.

X-ray radiation

Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstrahlung and electron shell transitions within atoms (characteristic X-rays). Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001

Table A-1. Environmental releases by building, 1993 to 2001.[a]

(Current) building no.	Building activity	Radionuclide	Activity fraction
131	M&E Divisions	Th-232	2.29E-10
		U-234	8.47E-02
		U-235	1.18E-02
		U-238	9.04E-01
132	Analytical & Nuclear Chemistry Laboratories; Forensic Sciences Center	H-3	7.3E-09
		Co-60	1.5E-07
		Th-228	1.2E-10
		Th-230	3.5E-10
		Th-232	7.9E-05
		U-234	3.1E-04
		U-235	7.2E-03
		U-238	9.9E-01
		Pu-238	4.4E-03
		Pu-239	6.7E-06
		Pu-240	1.7E-06
		Pu-241	1.6E-05
		Pu-242	2.0E-10
		Am-241	3.1E-07
151	Isotope Sciences, Environmental Services Laboratory	Ni-63	1.1E-01
		Zn-65	5.1E-07
		Sr-90	3.8E-05
		Y-90	3.7E-06
		Tc-99	1.3E-10
		Tc-99m	5.3E-08
		Ru-106	4.6E-09
		Sn-113	1.4E-07
		Sb-125	1.6E-06
		I-129	4.6E-10
		Ba-133	5.1E-08
		Cs-134	5.1E-08
		Cs-136	7.1E-07
		Cs-137	9.5E-05
		Ce-144	8.0E-07
		Pm-147	7.6E-05
		Eu-152	6.5E-08
		Eu-154	6.5E-08
		Eu-155	1.6E-12
		Bi-207	4.2E-05
		Po-209	1.9E-12
		Ra-226	2.5E-08
		Pa-231	2.7E-07
		Th-228	1.4E-09
		Th-229	2.2E-12
		Th-230	1.5E-09
		Th-232	6.4E-08
		U-232	2.7E-11
		U-233	1.3E-07

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
151 (continued)	Isotope Sciences, Environmental Services Laboratory	U-234	2.0E-03
		U-235	3.0E-04
		U-236	7.3E-06
		U-238	1.9E-02
		Np-237	8.9E-04
		Pu-236	5.3E-06
		Pu-238	2.0E-02
		Pu-239	1.7E-01
		Pu-240	3.8E-02
		Pu-241	6.1E-01
		Pu-242	9.5E-05
		Pu-244	5.1E-05
		Am-241	1.0E-02
		Am-243	7.6E-04
		Cm-242	5.1E-06
		Cm-244	1.5E-05
		Cm-246	1.2E-05
		Cm-248	8.0E-06
		Cf-249	1.5E-04
175	U-AVLIS	U-234	3.11E-01
		U-235	9.50E-03
		U-238	6.79E-01
177	U-AVLIS	U-234	3.05E-01
		U-235	1.04E-02
		U-238	6.85E-01
179	Unknown	U-234	4.38E-01
		U-235	1.24E-01
		U-238	4.38E-01
212	Physics and Space (rotating target neutron source)	H-3	1
222	Unknown	H-3	6.96E-01
		C-14	5.44E-10
		Ni-63	3.04E-01
		Th-232	6.86E-08
		U-234	2.84E-08
		U-235	1.25E-09
		U-238	7.13E-06
		Pu-239	6.53E-08
223	Unknown	Pu-238	1.09E-02
		Pu-239	8.78E-01
		Am-241	1.25E-04
		Am-243	1.11E-01
226	Unknown	H-3	1.00E+00
		U-238	9.67E-06
227	Unknown	U-234	5.05E-01
		U-235	2.26E-02
		U-238	4.73E-01
231	Safeguards and engineering	Th-232	1.3E-02
		U-234	8.5E-02
		U-235	1.2E-02
		U-238	8.9E-01

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
235	Characterization studies and ion beam experiments	Th-232	4.0E-08
		U-234	2.7E-01
		U-235	1.6E-02
		U-238	6.7E-01
		Pu-238	3.0E-04
		Pu-239	1.1E-02
		Pu-240	2.5E-03
		Pu-241	2.9E-02
235	Characterization studies and ion beam experiments	Pu-242	1.7E-07
		Am-241	1.4E-03
241	R&D	C-14	2.1E-07
		P-32	8.7E-08
		Th-232	1.9E-12
		U-234	1.0E+00
		U-235	1.8E-06
		U-238	4.0E-05
251	Heavy element facility	U-233	6.58E-09
		Pu-238	2.17E-04
		Pu-239	5.54E-09
		Pu-243	1.45E-06
		Am-241	3.29E-01
		Cm-243	1.69E-01
		Cm-244	5.02E-01
		Cm-248	4.18E-05
		Cf-252	1.37E-05
251	Laboratories and counting rooms	H-3	1.1E-04
		C-14	1.0E-07
		P-32	1.0E-06
		Sr-90	5.5E-06
		Sr-90/Y-90	2.8E-08
		Y-90	3.7E-08
		Cs-137	1.4E-07
		Bi-214	3.3E-01
		Pb-214	3.3E-01
		Po-218	3.3E-01
		Ra-226	1.5E-02
		Th-230	2.0E-08
		U-234	3.7E-04
		U-235	1.6E-05
		U-238	3.5E-04
		Np-237	4.3E-08
		Pu-238	2.6E-06
		Pu-239	6.6E-05
		Pu-240	1.5E-05
253	Unknown	Pu-241	6.5E-04
		Pu-242	9.9E-10
		Am-241	3.1E-06
254	Bioassays and analytical services	H-3	4.4E-02
		C-14	1.4E-02
		P-32	4.0E-03
		P-33	4.0E-01

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
254 (continued)	Bioassays and analytical services	S-35	4.1E-02
		Sr-90	9.9E-02
		Y-90	9.9E-02
		I-125	6.3E-02
		Po-209	2.6E-05
		Th-230	6.0E-03
		U-232	2.3E-04
		U-233	1.0E-05
		U-234	4.1E-04
		U-235	1.0E-05
		U-236	3.2E-03
		U-238	8.9E-07
		Pu-238	0.0E+00
		Pu-239	9.0E-05
		Pu-240	0.0E+00
		Pu-241	0.0E+00
		Pu-242	5.5E-04
		Np-237	4.4E-02
		Np-239	1.7E-01
		Am-241	3.2E-05
		Am-243	9.2E-04
		Cm-242	2.5E-04
		Cm-244	3.3E-04
		Cf-249	2.5E-04
		Cf-252	5.4E-03
255	Calibration laboratory	H-3	1.00E+00
		C-14	3.50E-10
		P-32	2.80E-08
		S-35	1.20E-09
		Sr-90	4.89E-13
		Y-90	4.89E-13
		I-125	5.91E-08
		I-131	1.80E-07
		Th-230	9.53E-11
		Th-232	2.50E-15
		U-233	2.50E-10
		U-234	1.10E-11
		U-235	2.90E-13
		U-236	8.69E-09
		U-238	4.19E-14
		Pu-239	2.77E-12
		Pu-242	1.40E-11
		Np-237	1.70E-09
		Np-239	1.70E-09
		Am-241	9.54E-13
		Am-243	2.54E-11
		Cm-242	9.09E-12
		Cm-244	9.49E-13
		Cf-252	8.39E-11

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
281	Tracer and dissolution studies	H-3	3.7E-02
		Be-10	1.5E-02
		C-14	2.8E-01
		Na-22	1.2E-04
		Cl-36	1.5E-02
		Ca-41	1.5E-01
		Mn-54	4.9E-06
		Ni-59	1.0E-04
		Ni-63	3.1E-01
		Co-60	5.0E-06
		Sr-90	1.5E-02
		Tc-99	1.5E-04
		Sb-125	4.9E-08
		Eu-152/Tm-171	1.7E-08
		Eu-154/Eu-155	2.1E-05
		Th-232	1.1E-06
		U-233	1.9E-04
		U-234	1.9E-04
		U-235	8.1E-06
		U-238	1.8E-04
		U-233/U-238	1.9E-06
		Pu-239	6.3E-04
		Pu-242	2.0E-03
		Pu-244	2.4E-06
		Np-237	1.7E-01
		Am-241	5.4E-04
		Am-241/Np-237	1.4E-04
282	Residual contamination	H-3	1.0E+00
		Rb-86/87	7.5E-09
292	Residual contamination rotating target neutron source	H-3	1
298	Laser fusion program	H-3	9.9E-01
		U-234	4.1E-04
		U-235	1.9E-05
		U-238	6.0E-03
321	Milling and shaping	DU Thorium[b]	unknown
321A	Milling and shaping	U-234	1.8E-01
		U-235	2.4E-02
		U-238	7.9E-01
321C	Milling, machining, and shaping	U-234	6.5E-01
		U-235	8.1E-03
		U-238	3.4E-01
322	M&E	U-234	8.1E-01
		U-235	1.2E-02
		U-238	1.8E-01
327	M&E	U-234	7.7E-02
		U-235	8.8E-02
		U-238	8.4E-01
331	Research and laboratories	H-3 (HTO)	4.69E-01
		H-3 (HT)	5.19E-01
		H-3	1.24E-02

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
332	Gloveboxes, HEPA filters	Pu-239 TRU	Unknown
341	Lasers Directorate	U-234	5.70E-02
		U-235	7.69E-03
		U-238	9.35E-01
361	R&D	H-3	5.6E-06
		C-14	3.3E-03
		P-32	8.3E-01
		P-33	1.1E-01
		S-35	5.4E-02
362	R&D	H-3	7.22E-01
		C-14	2.78E-01
363	R&D	H-3	9.99E-01
		C-14	1.11E-07
		P-32	1.42E-03
364	R&D	H-3	1.11E-03
		C-14	2.76E-03
		P-32	9.96E-01
365	R&D	H-3	4.04E-03
		C-14	9.96E-01
366	R&D	H-3	5.00E-01
		P-32	4.38E-01
		P-33	6.25E-02
377	R&D	H-3	7.66E-04
		C-14	3.83E-03
		P-32	9.95E-01
		Ni-63	1.15E-04
378	Unknown	Co-57	1.44E-03
		Co-60	5.59E-02
		Sr-85	6.52E-02
		Cd-109	3.58E-02
		Cs-134	6.98E-01
		Cs-137	6.98E-02
		U-233	2.14E-04
		U-234	8.66E-07
		U-235	4.84E-08
		U-238	2.00E-06
378	Unknown	Pu-236	2.13E-03
		Pu-239	6.99E-05
		Pu-240	5.59E-05
		Pu-242	1.05E-02
		Pu-244	3.58E-04
		Np-237	1.44E-06
		Am-241	6.00E-02
		Am-243	4.01E-04
381	Unknown	H-3	1
391	Unknown	H-3	1
412W	Unknown	Ni-59	3.50E-07
		Ni-63	1.00E+00
446	Unknown	C-14	1

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
513	Waste processing	H-3	9.85E-01
		C-14	2.49E-04
		P-32	3.17E-06
		K-40	5.79E-04
		Mn-54	8.34E-06
		Co-57	7.28E-07
		Co-60	1.03E-04
		Sr-90	7.27E-04
		Nb-95	8.34E-06
		Zr-95	1.30E-05
		Ru-106	6.07E-09
		I-125	7.49E-07
		I-131	2.70E-08
		Ba-133	6.09E-07
		Cs-134	4.13E-06
		Cs-137	2.61E-05
		Cs-138	1.52E-10
		Ce-141	6.07E-09
		Ce-144	9.56E-06
		Eu-152	1.67E-04
		Eu-154	2.12E-07
		Eu-155	3.79E-08
		Tl-208	6.22E-08
		Bi-212	8.50E-08
		Bi-214	5.92E-09
		Pb-210	7.89E-06
		Pb-212	1.43E-07
		Pb-214	2.12E-08
		Ra-223	4.55E-07
		Ra-226	8.65E-08
		Ra-228	4.55E-16
		Ac-228	1.52E-09
		Pa-231	8.50E-07
		Th-226	9.71E-07
		Th-227	1.67E-06
		Th-228	4.04E-05
		Th-232	3.51E-06
		Th-234	3.03E-06
		U-233	1.52E-05
		U-234	1.08E-03
		U-235	1.52E-04
		U-238	1.04E-02
		Pu-238	1.74E-09
		Pu-239	2.28E-04
		Pu-240	2.03E-08
		Pu-241	1.27E-06
		Pu-242	6.87E-07
		Am-241	7.62E-04
		Cm-244	3.03E-05
		Cf-249	2.28E-07

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
514	Waste processing	H-3	8.61E-01
		Be-7	7.53E-06
		C-14	2.40E-02
		Na-22	9.05E-05
		P-32	6.53E-02
		S-35	8.42E-03
		K-40	6.11E-05
		Sc-46	5.46E-06
		Cr-51	5.59E-06
		Fe-55	5.42E-06
		Mn-54	1.66E-05
		Co-56	1.82E-07
		Co-57	7.56E-05
		Co-58	5.46E-06
		Co-60	1.09E-04
		Ni-63	1.45E-07
		Zn-65	1.92E-08
		Y-88	5.33E-05
		Sr-89	4.96E-11
		Sr-90	9.13E-04
		Nb-94	1.19E-06
		Nb-95	8.46E-06
		Zr-95	1.20E-07
		Tc-99	6.05E-05
		Ru-103	2.18E-09
		Ru-106	1.33E-06
		Cd-109	2.89E-08
		Sb-125	2.08E-05
		I-125	1.39E-04
		I-131	3.63E-06
		Ba-133	1.02E-04
		Cs-134	6.37E-05
		Cs-137	1.16E-03
		Ce-139	7.14E-13
		Ce-141	1.82E-09
		Ce-144	4.87E-03
		Gd-148	6.04E-05
		Pm-147	3.07E-06
		Sm-151	2.45E-07
		Eu-152	2.29E-04
		Eu-154	2.32E-04
		Eu-155	2.54E-05
		Hf-172	7.33E-06
		Lu-173	1.92E-06
		Lu-174	5.42E-06
		W-185	5.62E-09
		Po-209	9.30E-06
		Po-210	9.30E-06
		Bi-207	4.71E-06
		Bi-210	7.14E-07
		Pb-210	5.62E-05

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
514 (continued)	Waste processing	Ra-226	9.73E-06
		Th-228	1.47E-05
		Th-229	6.04E-05
		Th-230	5.81E-05
		Th-232	7.87E-05
		U-232	6.08E-05
		U-233	7.01E-03
		U-234	3.60E-03
		U-235	2.18E-04
		U-236	8.92E-07
		U-237	1.82E-09
		U-238	8.29E-03
		Pu-236	3.67E-08
		Pu-238	4.72E-04
		Pu-239	6.74E-03
		Pu-240	1.68E-03
		Pu-241	1.49E-04
		Pu-242	1.60E-04
		Pu-244	1.09E-06
		Np-237	6.77E-05
		Np-239	2.41E-06
		Am-241	4.23E-03
		Am-243	8.33E-05
		Cm-244	6.76E-05
		Cf-249	9.30E-06
514 Tank Farm	Unknown	H-3	5.45E-01
		Be-7	3.33E-07
		C-14	2.59E-03
		Na-22	5.74E-05
		P-32	1.01E-01
		P-33	1.86E-04
		S-35	6.39E-03
		K-40	5.16E-05
		Sc-46	1.30E-06
		Cr-51	4.79E-07
		Fe-55	6.65E-07
		Fe-59	1.53E-06
		Mn-54	2.88E-06
		Co-56	2.01E-05
		Co-57	6.68E-03
		Co-58	2.14E-05
		Co-60	4.51E-05
		Zn-65	9.48E-07
		Ni-59	1.33E-06
		Ni-63	4.09E-06
		Y-88	2.12E-05
		Y-91	5.66E-06
		Sr-90	2.05E-01
		Zr-95	1.40E-03
		Nb-95	3.08E-05
		Mo-99	7.99E-06

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
514 Tank Farm (continued)	Unknown	Tc-99	2.76E-05
		Ru-103	2.54E-06
		Ru-106	4.35E-06
		Cd-109	8.97E-08
		Sn-113	4.66E-09
		Ag-110m	8.65E-07
		I-125	5.59E-05
		I-131	1.06E-05
		Sb-120m	7.32E-07
		Sb-124	6.25E-06
		Sb-125	9.93E-06
		Te-132	2.00E-07
		Ba-133	4.99E-05
		Ba-140	9.99E-07
		Cs-134	1.42E-05
		Cs-136	9.31E-07
		Cs-137	1.08E-01
		La-140	2.00E-08
		Ce-139	3.99E-09
		Ce-141	2.93E-05
		Ce-144	1.87E-04
		Nd-147	4.46E-06
		Pm-147	2.40E-06
		Gd-148	1.28E-05
		Sm-151	7.99E-08
		Eu-152	1.82E-04
		Eu-154	1.19E-04
		Eu-155	1.48E-05
		Eu-156	1.13E-07
		Tb-160	1.20E-08
		Hf-172	1.93E-06
		Hf-181	2.66E-06
		Lu-173	1.60E-06
		Lu-174	3.33E-07
		W-185	4.79E-09
		Au-195	4.26E-06
		Hg-203	3.06E-20
		Bi-207	1.04E-06
		Bi-210	5.06E-07
		Po-209	4.50E-06
		Po-210	5.06E-07
		Pb-210	1.36E-05
		Ra-226	1.55E-04
		Pa-233	1.60E-09
		Th-228	6.02E-03
		Th-229	1.28E-05
		Th-230	1.27E-05
		Th-232	1.14E-04
		U-232	1.31E-05
		U-233	4.44E-05
		U-234	1.63E-03

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
514 Tank Farm (continued)	Unknown	U-235	7.36E-04
		U-236	3.33E-07
		U-237	3.46E-04
		U-238	1.02E-02
		Pu-236	3.46E-08
		Pu-238	1.01E-04
		Pu-239	1.50E-03
		Pu-240	1.48E-04
		Pu-241	3.86E-05
		Pu-242	4.37E-05
		Pu-244	7.16E-08
		Np-237	2.31E-05
		Np-239	1.01E-04
		Am-241	1.02E-03
		Am-243	1.97E-05
		Cm-244	4.12E-05
		Cf-249	4.63E-06
612	Waste storage and repackaging	H-3	9.79E-01
		Be-7	1.22E-06
		C-14	7.04E-04
		Na-22	1.87E-04
		P-32	1.29E-02
		P-33	1.05E-08
		S-35	6.98E-06
		Cl-36	3.66E-11
		K-40	2.03E-05
		Sc-46	1.25E-06
		Cr-51	1.34E-06
		Fe-55	1.23E-06
		Mn-54	4.20E-06
		Co-56	7.33E-08
		Co-57	5.01E-06
		Co-58	1.34E-06
		Co-60	1.07E-03
		Ni-63	4.89E-08
		Zn-65	1.22E-08
		Se-75	2.88E-11
		Y-88	1.45E-07
		Y-91	1.05E-11
		Sr-85	8.72E-13
		Sr-89	3.93E-12
		Sr-90	6.02E-04
		Nb-94	5.50E-05
		Nb-95	1.34E-06
		Zr-95	1.28E-07
		Mo-99	1.55E-09
		Tc-99	1.60E-05
		Rh-102	2.01E-10
		Rh-103m	4.10E-13
		Ru-106	2.58E-08
		Cd-109	4.02E-05

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
612 (continued)	Waste storage and repackaging	Cd-115	5.25E-09
		Ag-110m	2.18E-14
		I-125	2.45E-05
		I-131	3.15E-09
		Sb-124	4.45E-14
		Sb-125	4.99E-06
		Sm-151	5.10E-11
		Ba-133	1.76E-05
		Ba-140	7.64E-09
		Cs-134	1.58E-05
		Cs-137	5.11E-04
		Ce-139	2.44E-13
		Ce-141	2.85E-08
		Ce-144	1.52E-04
		Nd-147	1.59E-09
		Pm-147	5.95E-10
		Pm-151	4.73E-10
		Sm-151	8.72E-13
		Gd-146	1.05E-05
		Gd-148	1.57E-05
		Eu-149	2.79E-12
		Eu-152	6.28E-05
		Eu-154	6.10E-05
		Eu-155	5.54E-06
		Tb-160	1.74E-13
		Hf-172	2.44E-06
		Lu-173	1.22E-06
		Lu-174	1.22E-06
		Ta-182	1.40E-11
		W-185	3.66E-09
		Ir-192	8.72E-14
		Au-195	5.41E-12
		Pt-195m	9.59E-10
		Hg-203	8.72E-17
		Bi-207	1.17E-06
		Bi-210	2.44E-07
		Po-209	3.30E-06
		Po-210	2.44E-07
		Pb-210	1.68E-05
		Ra-223	4.73E-12
		Ra-226	1.72E-06
		Ra-228	2.94E-09
		Th-228	1.45E-06
		Th-229	1.57E-05
		Th-230	1.56E-05
		Th-232	1.48E-05
		Th-234	1.13E-12
		U-232	1.58E-05
		U-233	2.08E-05
		U-234	4.93E-04
		U-235	1.85E-04

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
612 (continued)	Waste storage and repackaging	U-237	3.39E-08
		U-238	2.01E-03
		U-239	9.60E-12
		Pu-238	1.15E-04
		Pu-239	3.89E-04
		Pu-240	3.54E-05
		Pu-241	1.28E-04
		Pu-242	2.60E-05
		Pu-244	3.56E-06
		Np-237	1.69E-05
		Np-239	6.09E-07
		Am-241	3.74E-04
		Am-242	6.81E-09
		Am-242m	1.57E-09
		Am-243	4.11E-04
		Am-244	4.45E-06
		Cm-244	1.05E-04
		Cf-249	3.39E-06
		Cf-250	2.24E-12
612 Yard	Unknown	H-3	1.00E+00
		C-14	2.11E-06
		P-32	1.61E-07
		S-35	3.40E-08
		K-40	4.48E-11
		Cr-51	4.84E-11
		Mn-54	6.11E-13
		Co-57	6.38E-13
		Co-60	1.19E-10
		Ni-59	1.01E-12
		Ni-63	2.92E-11
		Se-75	2.45E-08
		Sr-90	9.84E-12
		Nb-95	3.19E-13
		Tc-99	2.44E-12
		Sb-125	4.09E-12
		Cs-134	1.02E-11
		Cs-137	3.60E-10
		Ce-144	4.25E-10
		Pm-147	1.70E-12
		Sm-151	5.26E-13
		Eu-152	5.45E-10
		Eu-154	2.89E-10
		Eu-155	6.11E-12
		Bi-207	1.81E-13
		Bi-214	4.52E-14
		Ra-226	3.65E-11
		Ra-228	1.22E-11
		Th-228	5.50E-10
		Th-230	1.17E-14
		Th-232	5.69E-10
		Th-234	1.17E-11

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
612 Yard (continued)	Unknown	U-233	5.40E-10
		U-234	1.87E-08
		U-235	2.47E-09
		U-238	1.87E-07
		Pu-238	2.43E-09
		Pu-239	5.46E-10
		Pu-240	2.12E-11
		Pu-241	4.43E-09
		Pu-242	1.02E-08
		Np-239	4.80E-11
		Am-241	1.45E-09
		Am-242	1.38E-12
		Am-243	4.22E-10
		Cm-243	4.89E-11
		Cm-244	3.35E-13
		Cm-245	3.03E-11
625	Waste operations	H-3	2.14E-01
		C-14	2.34E-04
		P-32	3.61E-10
		K-40	8.78E-05
		Mn-54	1.36E-06
		Co-57	5.74E-08
		Co-60	9.89E-06
		Sr-90	2.34E-06
		Y-88	1.70E-10
		Zr-95	2.55E-06
		Ru-106	7.01E-07
		Cd-109	4.40E-08
		Sb-125	9.78E-07
		Ba-133	1.01E-07
		Cs-134	1.11E-06
		Cs-137	1.05E-05
		Ce-141	4.46E-09
		Ce-144	1.57E-06
		Eu-152	4.47E-06
		Eu-154	2.55E-06
		Eu-155	2.55E-07
		Bi-214	2.98E-08
		Pb-212	3.83E-08
		Pb-214	4.89E-09
		Pa-231	4.89E-08
		Ra-226	1.51E-05
		Ra-228	6.59E-04
		Th-228	1.55E-06
		Th-230	5.53E-10
		Th-232	3.62E-06
		Th-234	4.89E-04
		U-233	4.46E-08
		U-234	2.76E-01
		U-235	4.89E-01
		U-238	2.32E-03

ATTACHMENT A
ENVIRONMENTAL RELEASES BY BUILDING, 1993 to 2001 (continued)

(Current) building no.	Building activity	Radionuclide	Activity fraction
625 (continued)	Waste operations	U-239	2.55E-07
		Pu-238	7.68E-05
		Pu-239	3.40E-03
		Pu-240	3.83E-06
		Pu-241	1.30E-02
		Pu-242	4.46E-08
		Np-237	2.34E-07
		Am-241	2.87E-04
		Am-243	5.10E-06
		Cm-243	9.78E-09
		Cm-244	2.34E-09
		Cm-245	1.38E-09
2561	Unknown	U-234	5.00E-01
		U-238	5.00E-01
		U-234	8.01E-02
		U-235	1.12E-02
		U-238	8.66E-01
Site 300-801	Unknown	H-3	1
Site 300 Pit 7	Unknown	H-3	1
Site 300-810A	Unknown	U-234	8.38E-02
		U-235	1.16E-02
		U-238	9.05E-01
Site 300-810B	Unknown	U-234	8.47E-02
		U-235	1.13E-02
		U-238	9.04E-01
Site 300-850	Unknown	H-3	9.6E-01
		U-234	3.2E-03
		U-235	4.4E-04
		U-238	3.4E-02
Site 300-851	Unknown	H-3	9.57E-01
		U-234	9.57E-04
		U-235	1.34E-04
		U-238	1.02E-02
Site 300 Well 8 Spring	Unknown	H-3	1

a. This table was derived from data reported for calendar years 1992 through 2001 as part of the National Emission Standards for Hazardous Air Pollutants (NESHAPs) (40 CFR Part 61, Subpart H) reporting process (Harrach et al. 1994; Gallegos et al. 1996; Gallegos and Biermann 1997; Gallegos et al. 1998; Biermann et al. 1999; Gallegos et al. 2000, 2001; Harrach et al. 2002, 2003). As part of its required reporting under NESHAPs, LLNL must monitor emissions by radionuclide and total activity at all building release points. This table lists only buildings for which NESHAPs data were reported; this is not a complete list of all buildings. For a listing of major buildings and activities, see ORAUT (2005b). To develop the table, the isotopic listing of each building in the relevant NESHAPs reports was captured in a building-specific spreadsheet. The annual emissions were totaled for each isotope and then for all isotopes in the listing. Activity fractions were obtained by dividing the Activity total by the “all isotopes” total (ORAUT 2005c). The Activity fraction for each radionuclide in the table is that fraction of the total activity released to the environment (i.e., the sum of the fractions for each building is equal to “1" or 100% of the mixture). However, those radionuclides with Activity fractions equal to “zero”, those with half-lives too short to contribute significant dose (i.e., isotopes of nitrogen and oxygen), and those that contribute submersion dose only were subsequently deleted from the listing.

b. LLNL (1987b).

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA

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INTERNAL DOSIMETRY COWORKER DATA (continued)

B.1 OVERVIEW

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

Bioassay results for LLNL were obtained as a copy of the site's MAPPER database converted to a spreadsheet with a cover letter explaining characteristics of the database and pointing out possibilities of data recording errors that might exist in the database (Mansfield 2006a). As sent to the ORAU Team, the MAPPER database had approximately 34,900 records. It contained no in vivo counting results and had only a few of the many tritium urinalysis results obtained throughout the operation of the Laboratory. The in vivo counting results were not available electronically and, therefore, were not analyzed as a coworker database. (In vivo results for individual energy employees are included in the files sent to NIOSH from LLNL.)

The principal radionuclides with potential for intakes at LLNL were tritium, isotopes of plutonium in a weapons-grade mixture, isotopes of uranium in mixtures typical of DU, NU, or slightly enriched uranium (to 5%), various TRU radionuclides, activation products, and fission products (ORAUT 2005b). The MAPPER database had small numbers of analyses for other radionuclides, such as ^{241}Am, curium, and TRU materials, but the numbers were too small for statistical analysis as an individual coworker dataset. LLNL generally used the gross alpha analysis to screen for intakes of TRU radionuclides with more specific follow-up analyses for the radionuclides of concern.

The urinalysis data chosen for this coworker study because of general applicability and number of measurements were for plutonium, uranium, gross alpha, gross beta, and an analysis labeled MFP. Two previous bioassay laboratory managers confirmed that the gross beta results and the MFP results were from the same procedure and should be treated as the same (ORAUT 2006b; Mansfield 2006b).

The statistical analyses of the bioassay data for each radionuclide were performed in accordance with the internal dosimetry coworker technical information bulletin (ORAUT 2005a) and the procedure, ORAUT-PROC-0095, *Generating Summary Statistics for Coworker Bioassay Data* (ORAUT 2006b). The resultant values were input to the Integrated Modules for Bioassay Analysis (IMBA) computer program and a fit to the data for each of the four analyzed groups at the 50th- and 84th-percentile values was performed to obtain intake rates for assignment of dose distributions. The 50th-percentile intakes were converted to 95th-percentile intakes based on the following methodology:

$$95\text{th-percentile intake} = 50\text{th-percentile intake} \times GSD^{1.645} \quad \text{(B-1)}$$

The tables listing the 50th- and 95th-percentile intakes are the values that the dose reconstructor should use when assigning coworker intakes. The 50th-percentile intakes are assigned as lognormal distributions with the associated GSD. The 95th-percentile intakes are assigned as constant distributions.

Coworker intakes for aged fission products and research-related radionuclides from 1994 to 1995, plutonium from 1991 to 1996, uranium from 1988 to 1996, and americium from 1971 to 1996 can be extended to later years. Building 251 transitioned from a Category II Nonreactor Nuclear Facility in 2002 to a radiological facility in 2005 when equipment and inventory were removed (Gray et al. 2006). During this transition of Building 251, there were no major contamination incidents and no radiation overexposures (in fact, doses were far lower than predicted). Individual and collective doses were

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

maintained as low as is reasonably achievable (ALARA). Collective annual WB doses were at least 3 times lower than ALARA goals and more than 10 times lower than conservative dose projections. Individual annual external WB doses were less than 150 mrem (Mitchell et al. 2006). Based on this information about Building 251, extending coworker intakes to later years is applicable.

Attachment C provides guidance on assigning the 50th- or 95th-percentile intakes based on job category and level of contact with the radioactive material.

Tables B-1 through B-4 list bioassay sample data statistics that were used to determine the coworker intakes.

Table B-1. Americium coworker sample statistic data.

Year	Number of samples	Number of employees
1957	89	78
1958	125	84
1959	133	83
1960	88	62
1961	116	79
1962	121	61
1963	98	59
1964	134	83
1965	139	78
1966–67	83	54
1968	89	58
1969	106	68
1970	122	85
1971–72	78	35
1973–75	51	35
1976–78	112	75
1979–81	127	90
1982	117	64
1983	127	64
1984	140	65
1985	157	92
1986	210	110
1987	160	96
1988	141	81
1989	165	79
1990	173	91
1991	205	104
1992	220	119
1993	319	144
1994	325	138
1995	300	130
1996	173	99

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

Table B-2. Beta coworker sample statistic data.

Year	Number of samples	Number of employees
1957–63	50	41
1964–65	75	60
1966–73	43	42
1974–75	112	57
1976	70	35
1977	96	46
1978–79	144	75
1980–81	148	91
1982	120	69
1983	152	79
1984	200	103
1985	199	112
1986	265	137
1987	227	143
1988	225	123
1989	218	103
1990	250	125
1991	298	136
1992	287	152
1993	407	179
1994	357	153
1995	298	126
1996	171	98

Table B-3. Plutonium coworker sample statistic data.

Year	Number of samples	Number of employees
1957	90	79
1958	125	84
1959	133	83
1960	88	62
1961	116	79
1962	121	61
1963	126	73
1964	140	87
1965	154	82
1966	102	57
1967	84	57
1968	112	66
1969	133	84
1970	133	87
1971	182	111
1972	175	95
1973	106	75
1974	113	83
1975	137	85
1976	135	94

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

Year	Number of samples	Number of employees
1977	169	107
1978	140	102
1979	119	76
1980	118	80
1981	194	141
1982	314	185
1983	340	180
1984	402	205
1985	384	214
1986	464	253
1987	450	268
1988	492	285
1989	529	300
1990	399	233
1991	377	212
1992	443	260
1993	522	261
1994	493	244
1995	447	223
1996	252	168

Table B-4. Uranium coworker sample statistic data.

Year	Number of samples	Number of employees
1958	74	67
1959–61	87	70
1962–64	96	44
1965	107	88
1966	113	90
1967	51	31
1968	141	87
1969	95	92
1970	118	101
1971	35	33
1972	272	172
1973	172	138
1974	139	120
1975 S1	146	134
1975 S2	159	128
1976 S1	139	136
1976 S2	95	92
1977 S1	142	133
1977 S2	151	122
1978 S1	155	148
1978 S2	131	120
1979 S1	213	149
1979 S2	204	163
1980 S1	199	108
1980 S2	141	109

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INTERNAL DOSIMETRY COWORKER DATA (continued)

Year	Number of samples	Number of employees
1981 S1	157	126
1981 S2	168	131
1982 S1	245	163
1982 S2	271	194
1983 S1	278	195
1983 S2	317	219
1984 S1	286	191
1984 S2	230	165
1985 S1	219	159
1985 S2	213	153
1986 S1	238	178
1986 S2	257	194
1987 Q1	220	153
1987 Q2	254	171
1987 Q3	221	139
1987 Q4	224	151
1988 Q1	273	150
1988 Q2	256	154
1988 Q3	236	129
1988 Q4	175	119
1989 Q1	175	126
1989 Q2	195	147
1989 Q3	188	131
1989 Q4	230	146
1990 Q1	211	142
1990 Q2	192	121
1990 Q3	210	120
1990 Q4	238	141
1991 Q1	298	187
1991 Q2	286	153
1991 Q3	284	143
1991 Q4	214	134
1992 Q1	313	204
1992 Q2	326	197
1992 Q3	279	168
1992 Q4	278	174
1993 Q1	332	202
1993 Q2	304	188
1993 Q3	324	201
1993 Q4	316	200
1994 Q1	247	183
1994 Q2	152	122
1994 Q3	148	120
1994 Q4	120	98
1995 Q1	189	153
1995 Q2	114	98
1995 Q3	151	110
1995 Q4	194	146
1996 Q1	197	122
1996 Q2	354	206
1996 Q3	182	134

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

The MAPPER database was used for the determination of the coworker study datasets, rather than the data from individual claims. The cover letter from LLNL transmitting the MAPPER database identified 11 potential issues with the data (Mansfield 2006a). These issues are addressed below.

- Employee Identification. While it is helpful to have information linking all of an individual's samples for a given radionuclide for reviewing the data, investigating outliers, and ensuring that there is no bias due to a few individuals, errors in a fraction of the identifiers will not affect the coworker statistics.

- Sample Dates. Sample dates are used only in grouping results into datasets. The shortest statistical analysis interval is 3 months, so the day of the month that a sample was collected is not necessary information for the coworker study.

- Analyte. Obvious typographical errors in the analyte were corrected. A list of sample numbers associated with likely erroneous results was provided in the cover letter transmitting the MAPPER database, and those involving ^{238}Pu were not included in the ^{239}Pu analysis. Tritium coworker data were not evaluated.

- Results Less Than Zero. The cover letter notes that these obvious problems were corrected before sending the dataset to the ORAU Team.

- Result Error. Result error has no impact on a coworker analysis; all results are used as reported.

- Units. As noted by LLNL, the errors were largely typographical errors and the correct values are apparent.

- Media. There are only eight samples with a media code R or S; all were assumed to be urine samples. Neither the sample with media code B nor the two samples with a result of 1747.8, were included the analyses, both because the same sample number was attributed to two individuals and because the result appears to be far outside the range of other urine values, indicating a very unusual circumstance or error.

- Priority. Priority of a sample is irrelevant to the coworker analysis.

- Exposure Area. Because there are limited samples available for the site, no analysis by work area was performed. This field was therefore not used in the study.

- Result Status. All comments were reviewed and samples indicated to be invalid (e.g., spiked, cross contaminated, etc.) were excluded from the analysis.

- Assumptions about Sample Volume. Discussions were held with LLNL personnel, including the former and present bioassay laboratory managers, about this issue. A resolution favorable to claimants was adopted; see the discussion in Section B.3.2 for details.

B.2 LIMITATIONS

NIOSH has determined, with concurrence from the Secretary of HHS, that MFP doses at LLNL cannot be reconstructed between 1950 and 1973, inclusive. For this reason, a class of LLNL employees has been added to the SEC. Therefore, only a limited assessment of internal dose can be performed for

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

periods before 1974 for workers who were monitored for some type of radiation exposure that is covered under the EEOICPA (Leavitt 2008). If monitoring data are available for workers who are included in the SEC, dose is to be assigned as appropriate based on available monitoring data. However, such dose reconstructions are still considered partial dose reconstructions.

NIOSH has further determined that internal doses from ^{233}U at LLNL cannot be reconstructed with sufficient accuracy between 1974 and 1989, inclusive. For this reason, a class of LLNL employees has been added to the SEC (Burwell 2016). If monitoring data are available for workers who are included in the SEC, dose is to be assigned as appropriate based on available monitoring data. However, such dose reconstructions are still considered partial dose reconstructions.

B.3 DISCUSSION OF THE DATASETS

B.3.1 Uranium

The MAPPER database contained uranium urinalysis data from 1958 to 1996. Nearly all of the results were reported as ^{238}U in micrograms per liter. The LLNL cover letter indicates that the results actually represent total uranium (Mansfield 2006a). According to Section 5.2, most uranium exposure was to DU or NU. The AVLIS project converted kilogram quantities of NU to slightly enriched uranium, up to 5%, as a demonstration plant. The uranium for the AVLIS project came from the Feed Materials Production Center. The AVLIS project operated in Buildings 175, 177, and the 482 complex from 1973 through 1999. HEU that came from Y-12 was used throughout the site's history in only a few selected buildings. HEU was used in Building 231 and the Oralloy Shop in Building 321C. There was another project involving HEU in the 1990s, involving the United States and Russian HEU Purchase Agreement. The program monitors the Russian process of converting weapons-usable HEU into low-enriched uranium. A Transparency Monitoring Office in Russia was established in 1996 by DOE and is staffed in part by Livermore workers. When identified in MAPPER, results from workers on these projects were excluded from the uranium dataset because of the higher specific activity of uranium on that project (31 results). Uranium results were left in units of micrograms per liter for the statistical analysis and curve fitting.

The uranium dataset has no negative (i.e., <0) results. MDAs for the uranium analysis methods were not provided in the LLNL cover letter. Table 5-8 lists the MDA as 5 µg/L through 1973, 1 µg/L for 1974 through 1978, 0.3 µg/L for 1979 through 1989, and 0.05 µg/L for 1990 through 1999. Most of the results less than 1 µg/L were recorded as zero before 1974. In 1974, the database shows a distinctly different pattern, with an overall decrease in the general magnitude of the results, a lower reporting level at around 0.01 µg/L, and sufficient results between 0.01 to 1 µg/L to support a reasonably normal or lognormal pattern implying that a change to a more sensitive analysis technique took place. The zero results were included in the ranking of the data but not the curve fitting. In several years before 1974, there were a few results recorded with values between 1 and 0 µg/L. Because there were just a few of these and not in all years, they seemed to be anomalous recordings relative to the 1-µg/L reporting level. Because they might have been misrecorded data, such as misplaced decimals or incorrect unit conversion, and because they were clearly anomalous when included in the lognormal plots of the results, these results were excluded from the final annual datasets. There were 37 of these results throughout the period from 1962 through 1973, in comparison with 1,200 results for the coworker study during this period.

The uranium dataset had entries from recounts on the same sample and entries showing multiple same-day samples from a given worker. The recounted samples were identified as having identical information, including sample number, but different results and different uncertainty. The multiple

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

same-day samples were indicated by having the same worker identification and sample dates, but the sample numbers were different, often by one digit. Because two counts on a single sample or two samples from a given worker on the same day produces a bias in comparison with other workers' samples that were given only a single count, the multicounted samples were replaced by the average of the multiple counts.

Duplicate entries that were obvious and samples that were marked by LLNL as baselines or quality control samples were excluded.

Table B-5 lists the time intervals for separate statistical analyses. Intervals were chosen to obtain approximately 100 results or more per interval. However, 1967 had only 51 results and 1971 had only 34 results. (The median and 84th-percentile values for 1971 were also much lower than surrounding years, perhaps indicating that there was little work with uranium at the site in 1971.) The statistical parameters for each interval (e.g., 50th and 84th percentiles) were assigned to the midpoint of the interval.

Table B-5. Time intervals for statistical analysis of uranium urine samples.

Calendar year	Analysis interval
1958	Year
1959−1961	Treated as a single interval
1962–1964	Treated as a single interval
1965–1974	Year
1975–1986	6 months
1987−1996	3 months

B.3.2 Gross Alpha, Plutonium, and Americium/Curium

In one analytical procedure used at LLNL, plutonium, americium, neptunium, curium, actinium, and thorium were extracted from the urine matrix and counted by a gross alpha technique. This is referred to as a "gross alpha" analysis because any measured activity in the urine sample was an indication that one or more of these radionuclides was present. In MAPPER these are called "ALPHA" results.

In another procedure the activity was separated from its urine matrix with anion exchange resin. Plutonium, americium, and curium were retained on the resin, with the plutonium subsequently eluted off. This plutonium fraction of the eluate is referred to as "PU239" in MAPPER and in this report. Pulse height analysis was used for plutonium analysis starting in 1967 or 1968 (the first ^{238}Pu results show up in 1968) (ORAUT 2006b), so as of 1968 a PU239 result is interpreted to mean $^{239+240}$Pu.

In an interview (ORAUT 2006b) with the [position redacted], it was pointed out that the eluate from the anion exchange column that did not contain plutonium was often collected and analyzed by a gross alpha counting technique. This effluent would contain americium and curium and the results were also reported as ALPHA in MAPPER. Note that the two different ALPHA analyses in MAPPER potentially report different radionuclides, so to avoid confusion this analysis is referred to here as "AM/CM," while the results from the procedure described in the first paragraph are referred to as "GA." A pair of PU239 and AM/CM analyses is referred to as a "split" analysis. There was no reliable way to determine from the ALPHA results in MAPPER which procedure was used, although it is likely that all splits used the anion exchange procedure (ORAUT 2006b) and this will be assumed here.

In summary, four types of analytical results are evaluated here:

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

- GA – a nonspecific analysis that can potentially report any combination of plutonium, americium, neptunium, curium, actinium, and thorium.

- PU239 – an analysis specific for ^{239}Pu. Might or might not have an associated AM/CM result.

- AM/CM – a nonspecific analysis that can potentially report any combination of alpha-emitting isotopes of americium and curium. Always has an associated PU239 result.

- AM241 – an analysis that is specific for ^{241}Am.

And, to be clear:

ALPHA – an analyte type in MAPPER that could be either GA or AM/CM.

All PU239 and AM/CM urinalysis results were recorded in dpm which, according to the LLNL cover letter, should be interpreted as dpm/24 hours with the exception of samples that were split for AM/CM and PU239 analyses (Mansfield 2006a) (see additional discussion on splits below). Most GA urinalysis results were recorded in dpm, which should also be interpreted as dpm/24 hours. Some GA sample results in 1957 and 1958 had units of cpm; because the appropriate conversion factor between cpm and dpm was not known, the cpm results were excluded from this analysis.

There were no negative (less than zero) PU239 or AM/CM results in the dataset, but numerous results were recorded as zero and some small positive results were below the reported MDA for the time. For example, in 1990, the recorded PU239 results descend rather orderly down to 0.001 dpm, then there are 3 results at 1.11×10^{-7}, 9.3×10^{-8}, and 7.7×10^{-9} dpm. All of these types of extremely small results were indicated in the database as not detected (code 99). However, when included as recorded they distorted the curve fitting and so were treated as zero for curve fitting purposes. (Another way to look at this problem is that it appears that most of the time results below 0.001 dpm were recorded as zero, but the policy was not rigorous and some of these very small results were entered. It is also possible these samples were analyzed by a method different from all the others and so were recorded differently.) Therefore, all PU239 and AM/CM results less than 0.001 dpm were censored in the analysis (i.e., the rank of the sample was represented in the probability plot but the numerical value was not in fitting the lognormal distribution to data for each time interval). There were no negative gross alpha results, except in 1996. A lower recording level of about 0.01 dpm appears to have been used fairly consistently throughout from 1957 to 1995 with a few results recorded in the 0.001- to 0.009-dpm range intermittently throughout the years. As with PU239, all ALPHA results equal to or less than 0.001 dpm were censored in the analysis.

Starting in 1976, some samples were split and given two analyses, one recorded as PU239 and one recorded as ALPHA (called AM/CM here). According to the LLNL cover letter (Mansfield 2006a), the recorded results were for the split sample, not the total sample, and to obtain a 24-hour result the split-sample results should be doubled. However, in a conference call interview with the former [position redacted], present [position redacted], and other LLNL personnel, the former [position redacted] said that during [pronoun redacted] time in the [building redacted] plutonium/alpha split samples were reported as “per total sample volume” (ORAUT 2006b). The present [position redacted] confirmed that the interpretation in the cover letter (that splits should be considered as representing half the volume) was correct for recent times (Mansfield 2006a; ORAUT 2006b). No one was sure of the exact time when the change in policy occurred. There was agreement that the change occurred in the mid- to late 1980s. The former [position redacted] was present in the [building redacted] until 1986 or 1987. As a consequence, the data were analyzed twice: once with split-

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sample results being doubled starting in 1987 and the other with split-sample results being doubled starting in 1988. Review of the plutonium data was inconclusive, but continuity of the geometric means for the gross alpha results was better for the 1988 assumption. Therefore, a PU239 and an ALPHA result in MAPPER on the same day are considered to be "splits." Splits have different sample numbers, and the ALPHA (AM/CM) portion of the split does not contain PU239. Therefore, all PU239 and ALPHA (AM/CM) results from splits from 1988 to 1996 were multiplied by a factor of 2.

In later years, it appeared that many samples were recounted and some workers had multiple same-day samples. Multiple results on the same day are "recounts" if they have the same sample number. The final result for a recount is the mean of the multiple results, which is assigned to one sample, discarding the others. Multiple results on the same day are partial samples if they have different sample numbers. These results are assumed to be expressed in dpm per partial sample, so the final result for this sample is the sum of the results, which is assigned to one sample, discarding the others.

Incident samples are problematic for performing group statistics and applying group median values to unmonitored workers because of the large number of samples that often were obtained for a single individual after a known acute intake. LLNL sent a list of known large acute intakes or intakes that occurred to LLNL employees while working at other sites; this list was used to remove sample results from known acute intakes of plutonium. There were 19 samples excluded from the ALPHA results from a non-DOE facility. There were 25 samples excluded from the PU239 results from other DOE sites. There were 18 samples excluded from the PU239 results from wounds. Other sample results removed from the dataset before analysis included duplicate entries and those marked as invalid for some reason.

For statistical analysis the data were divided into two categories:

- All PU239 and GA results are considered to be ^{239}Pu.
- All AM/CM, AM-241, and GA results are considered to be americium/curium.

Note that the GA results play a dual role, being considered to measure ^{239}Pu on one hand and americium/curium on the other. In general, the time interval for separate statistical analyses was the calendar year with the goal of having about 100 or more results per analysis period. There were sufficient ^{239}Pu results to permit analyses for each year from 1957 to 1996, but a number of years had to be combined for the americium/curium: 1966 and 1967; 1971 and 1972; 1973, 1974, and 1975; 1976, 1977, and 1978; and 1979, 1980, and 1981. The results of the statistical analysis (the 50th- and 84th-percentile excretion rates) were assigned to the midpoint of the analysis interval, which is July 1 for an annual interval.

B.3.3 Gross Beta and Mixed Fission Products

There is only one procedure in the 1979 bioassay procedures manual for gross beta analysis, "Gross Beta Determination of Fission Products in Urine," and, according to the former bioassay laboratory, that was the only procedure used (Miller 1979; ORAUT 2006b). As stated above, the gross beta and MFP results in MAPPER were treated as a single dataset. The gross beta procedure consisted of wet-ashing to white salts, dissolution in nitric acid, coprecipitation with ammonium hydroxide, plating on stainless-steel planchets, and gross beta counting.

MAPPER had gross beta or MFP data starting in 1957. In general, the results were recorded as dpm or dpm/L, but some results were recorded as cpm, cpm/L, µCi, or µCi/L. The cpm and cpm/L results were excluded from the statistical analysis to avoid having to estimate the cpm-to-dpm conversion

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factor. It was also not certain that the unit was entered correctly; that is, the magnitude of the result might have been correct, but the unit should have been entered as dpm. The µCi and µCi/L samples were also excluded because the magnitudes of the results were not obviously consistent with the µCi unit and error in the recording of the units was considered possible. Results reported as dpm/L were converted to dpm/24-hour sample by multiplying by 1.4 L/d reference excretion (ICRP 1975).

The only negative numbers in the database were recorded in 1996. Negative numbers and zeros were included in the ranking of the results but were not included in the fitting of the lognormal curve.

Starting in 1984, there were situations in which two samples were listed for the same worker in the same day. Some of these had the same sample numbers but different results. These were assumed to be recounts on the same sample. To reduce bias from multiple counts on a single sample, the average of the same-day results were used. Also starting in 1984, there were situations in which two or more samples were listed for the same worker in the same day but had different sample numbers. It is not known if these were actually different voidings from the worker or different aliquots of the same sample. To reduce bias, these were reduced to a single result. If the results were reported as dpm/L, the average of the two results was used before normalization to 24-hour samples; if the results were reported as dpm, the two results were added to get the total dpm/d.

Duplicate entries that were obvious and samples that were marked by LLNL as baselines or quality control samples were excluded. No samples were excluded due to known large acute intakes. There was a single high outlier in the dataset in 1985, which appeared to be an error (5,718,000 dpm/L without a follow-up sample or recount and marked as “negative” in the MAPPER field that shows the dosimetrist’s disposition of the result). This result was excluded.

The initial statistical analyses of the gross beta results resulted in poor fits and large GSDs for many of the years. This situation appeared to be due in part because there was a scattering of results between 1 to 0 dpm/d that were not consistent with a lognormal or even a normal distribution. These results were few in number and were scattered over several decades (e.g., a few between 0.1 to 1 dpm and a few between 0.01 to 0.1 dpm). An analysis of the 99-code break points (i.e., the bioassay result below which almost all samples were labeled as nondetections) for the gross beta analyses implied that the decision level for the analysis was in the tens of dpm/d before 1982 and hundreds of dpm/L for 1982 through 1996. Because these random, small results were well below the detection or decision level and appeared to be outliers relative to the rest of the distribution, all results less than 1 dpm/d were treated as if they were zero. As a consequence, they were included in the ranking but were not part of the fitting of the lognormal curve.

The entire dataset for 1996 was suspect. The dataset had a definite low bias, with negative numbers running into the thousands and with 46 results <0, 101 results recorded exactly as zero, and only 4 results >0. The distribution for this year was the mirror image of a lognormal distribution. Statistical parameters were not generated for this year.

Table B-6 lists the time intervals for separate statistical analyses. Intervals were chosen to obtain approximately 100 results or more per interval (although in some years consideration was given to how many results were >0). Before 1974, there were too few samples to perform meaningful statistics so the intakes were determined only for 1974 through 1995. The statistical parameters for each interval (e.g., 50th and 84th percentiles) were assigned to the midpoint of the interval.

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Table B-6. Time intervals for statistical analysis of gross beta urine samples.

Calendar year	Analysis interval
1957−1973	Intakes not determined (168 samples total for the 17-yr period)
1974–1975	Treated as a single interval
1976	Year (70 samples)
1977	Year
1978−1979	Treated as a single interval
1980–1981	Treated as a single interval
1982–1995	Year
1996	Not analyzed

The procedure did not state exactly which elements would have been quantitatively captured on the planchet. This question was posed to a Hanford [position redacted]; [pronoun redacted] reply was (ORAUT 2006d):

> *The LLNL procedure will recover and measure nearly everything except K-40, Cs-137, tritium, carbon, sulfur, and halogens. It will recover and measure strontium, barium, radium, most of the transition metals, all the rare earths, and all the actinides. Some of the transition metals that form strong ammine complexes probably aren't completely recovered by this method; they include Co, Ni, Cu, Zn, and Ag. Nickel is probably not recovered at all. Iron should be recovered and measured completely. Ruthenium is probably mostly recovered by this method.*

B.3.4 Analysis

For each of the radionuclide groups above, a lognormal distribution for the data in the intervals specified in Tables B-1 and B-2 was assumed (ORAUT 2006c). The 50th- and 84th-percentile values were calculated using the method described in ORAUT (2012). Table B-7 shows the statistical analysis results for uranium.

Table B-7. Uranium urine bioassay data for 50th and 84th percentiles (µg/L and mg/d) along with the period and effective bioassay date, 1958 through 1996.

Period	Effective bioassay date	50th[a] (µg/L)	84th[b] (µg/L)	50th[a] (mg/day)	84th[b] (mg/day)
1958	6/30/1958	6.89E+00	1.39E+01	9.64E-03	1.95E-02
1959–61	6/30/1960	6.13E+00	1.48E+01	8.58E-03	2.07E-02
1962–64	6/30/1963	5.77E+00	1.35E+01	8.08E-03	1.89E-02
1965	6/30/1965	7.61E+00	1.54E+01	1.07E-02	2.16E-02
1966	6/30/1966	1.62E+00	6.24E+00	2.26E-03	8.74E-03
1967	6/30/1967	6.91E+00	1.40E+01	9.67E-03	1.96E-02
1968	6/30/1968	5.42E+00	9.34E+00	7.58E-03	1.31E-02
1969	6/30/1969	3.68E+00	7.73E+00	5.15E-03	1.08E-02
1970	6/30/1970	1.91E+00	5.41E+00	2.68E-03	7.58E-03
1971	6/30/1971	9.16E-01	3.89E+00	1.28E-03	5.45E-03
1972	6/30/1972	8.32E+00	2.03E+01	1.16E-02	2.84E-02
1973	6/30/1973	4.31E+00	1.33E+01	6.04E-03	1.87E-02
1974	6/30/1974	4.57E-02	2.61E-01	6.41E-05	3.66E-04
1975 S1[c]	3/31/1975	6.25E-02	1.87E-01	8.75E-05	2.62E-04
1975 S2[c]	9/30/1975	7.56E-02	2.95E-01	1.06E-04	4.13E-04

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Period	Effective bioassay date	50th[a] (µg/L)	84th[b] (µg/L)	50th[a] (mg/day)	84th[b] (mg/day)
1976 S1[c]	3/31/1976	3.08E-02	2.09E-01	4.32E-05	2.93E-04
1976 S2[c]	9/30/1976	2.75E-02	1.79E-01	3.86E-05	2.50E-04
1977 S1[c]	3/31/1977	2.11E-02	1.03E-01	2.95E-05	1.44E-04
1977 S2[c]	9/30/1977	2.42E-02	1.90E-01	3.39E-05	2.66E-04
1978 S1[c]	3/31/1978	3.27E-02	1.65E-01	4.58E-05	2.32E-04
1978 S2[c]	9/30/1978	3.32E-02	1.72E-01	4.64E-05	2.40E-04
1979 S1[c]	3/31/1979	2.14E-02	1.07E-01	2.99E-05	1.50E-04
1979 S2[c]	9/30/1979	3.71E-02	1.74E-01	5.19E-05	2.44E-04
1980 S1[c]	3/31/1980	2.48E-02	2.75E-01	3.47E-05	3.85E-04
1980 S2[c]	9/30/1980	8.37E-03	6.44E-02	1.17E-05	9.02E-05
1981 S1[c]	3/31/1981	9.34E-03	8.28E-02	1.31E-05	1.16E-04
1981 S2[c]	9/30/1981	8.42E-02	3.66E-01	1.18E-04	5.12E-04
1982 S1[c]	3/31/1982	3.96E-02	1.98E-01	5.55E-05	2.77E-04
1982 S2[c]	9/30/1982	8.18E-02	3.61E-01	1.15E-04	5.06E-04
1983 S1[c]	3/31/1983	2.84E-02	2.23E-01	3.98E-05	3.12E-04
1983 S2[c]	9/30/1983	3.69E-02	2.04E-01	5.17E-05	2.85E-04
1984 S1[c]	3/31/1984	2.78E-02	1.55E-01	3.90E-05	2.17E-04
1984 S2[c]	9/30/1984	4.76E-02	1.59E-01	6.66E-05	2.22E-04
1985 S1[c]	3/31/1985	3.33E-02	2.18E-01	4.66E-05	3.05E-04
1985 S2[c]	9/30/1985	2.86E-02	1.10E-01	4.00E-05	1.54E-04
1986 S1[c]	3/31/1986	2.01E-02	1.37E-01	2.82E-05	1.92E-04
1986 S2[c]	9/30/1986	5.12E-03	5.04E-02	7.16E-06	7.05E-05
1987 Q1[c]	2/15/1987	6.01E-03	5.42E-02	8.41E-06	7.59E-05
1987 Q2[c]	5/15/1987	1.12E-02	6.62E-02	1.56E-05	9.27E-05
1987 Q3[c]	8/15/1987	2.04E-02	1.05E-01	2.85E-05	1.47E-04
1987 Q4[c]	11/15/1987	1.90E-02	1.23E-01	2.66E-05	1.72E-04
1988 Q1[c]	2/15/1988	1.38E-02	1.15E-01	1.93E-05	1.61E-04
1988 Q2[c]	5/15/1988	6.78E-02	1.58E-01	9.49E-05	2.22E-04
1988 Q3[c]	8/15/1988	6.88E-02	1.66E-01	9.64E-05	2.32E-04
1988 Q4[c]	11/15/1988	1.04E-01	2.75E-01	1.45E-04	3.84E-04
1989 Q1[c]	2/15/1989	7.95E-02	1.88E-01	1.11E-04	2.63E-04
1989 Q2[c]	5/15/1989	1.04E-01	2.58E-01	1.46E-04	3.61E-04
1989 Q3[c]	8/15/1989	7.69E-02	2.16E-01	1.08E-04	3.02E-04
1989 Q4[c]	11/15/1989	1.04E-01	2.83E-01	1.46E-04	3.97E-04
1990 Q1[c]	2/15/1990	5.56E-02	1.42E-01	7.78E-05	1.99E-04
1990 Q2[c]	5/15/1990	4.67E-02	1.19E-01	6.54E-05	1.67E-04
1990 Q3[c]	8/15/1990	4.64E-02	1.02E-01	6.49E-05	1.42E-04
1990 Q4[c]	11/15/1990	5.33E-02	8.95E-02	7.46E-05	1.25E-04
1991 Q1[c]	2/15/1991	7.23E-02	1.35E-01	1.01E-04	1.89E-04
1991 Q2[c]	5/15/1991	8.44E-02	1.59E-01	1.18E-04	2.22E-04
1991 Q3[c]	8/15/1991	6.28E-02	1.11E-01	8.79E-05	1.55E-04
1991 Q4[c]	11/15/1991	3.92E-02	7.68E-02	5.48E-05	1.08E-04
1992 Q1[c]	2/15/1992	3.84E-02	7.15E-02	5.37E-05	1.00E-04
1992 Q2[c]	5/15/1992	3.67E-02	6.04E-02	5.13E-05	8.45E-05
1992 Q3[c]	8/15/1992	4.19E-02	7.38E-02	5.86E-05	1.03E-04
1992 Q4[c]	11/15/1992	3.11E-02	4.56E-02	4.35E-05	6.38E-05
1993 Q1[c]	2/15/1993	3.58E-02	5.45E-02	5.01E-05	7.63E-05
1993 Q2[c]	5/15/1993	4.38E-02	6.67E-02	6.14E-05	9.34E-05
1993 Q3[c]	8/15/1993	4.34E-02	6.85E-02	6.07E-05	9.58E-05
1993 Q4[c]	11/15/1993	5.26E-02	7.10E-02	7.36E-05	9.94E-05
1994 Q1[c]	2/15/1994	4.76E-02	6.45E-02	6.67E-05	9.03E-05

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Period	Effective bioassay date	50th[a] (µg/L)	84th[b] (µg/L)	50th[a] (mg/day)	84th[b] (mg/day)
1994 Q2[c]	5/15/1994	4.32E-02	6.54E-02	6.05E-05	9.16E-05
1994 Q3[c]	8/15/1994	5.07E-02	7.46E-02	7.09E-05	1.04E-04
1994 Q4[c]	11/15/1994	5.34E-02	7.86E-02	7.47E-05	1.10E-04
1995 Q1[c]	2/15/1995	6.47E-02	9.70E-02	9.06E-05	1.36E-04
1995 Q2[c]	5/15/1995	5.13E-02	7.84E-02	7.18E-05	1.10E-04
1995 Q3[c]	8/15/1995	4.63E-02	6.61E-02	6.48E-05	9.26E-05
1995 Q4[c]	11/15/1995	4.71E-02	7.43E-02	6.59E-05	1.04E-04
1996 Q1[c]	2/15/1996	5.24E-02	9.03E-02	7.34E-05	1.26E-04
1996 Q2[c]	5/15/1996	4.86E-02	8.39E-02	6.81E-05	1.18E-04
1996 Q3[c]	8/15/1996	5.53E-02	8.90E-02	7.75E-05	1.25E-04

a. 50th-percentile value of the fitted line.
b. 84th-percentile value of the fitted line.
c. S1 means first 6 months of the year, S2 means second 6 months of the year, Q1 means months 1 to 3 of the year, Q2 means months 4 to 6, Q3 means months 7 to 9, and Q4 means months 10 to 12.

B.4 INTAKE MODELING

B.4.1 Assumptions

All urinalysis results were assumed to be representative of a full-day (24-hour) urinary excretion or were normalized to be representative of a full day before creating the lognormal plots with the exception of the uranium data, which were normalized to daily excretion before intake modeling [7]. Each result that was used in the intake calculation was assumed to have a normal distribution and a uniform absolute error of 1 was applied to all results, which weighted all results equally [8]. A chronic exposure pattern was assumed unless the data clearly showed a sharp, short-term increase in excretion. Intakes were assumed to be from inhalation using a default breathing rate of 1.2 m^3/hr and a 5-µm AMAD particle size distribution [9].

When calculating doses to individuals from bioassay data, a GSD of 3 has been used to account for biological variation and uncertainty in the models. It was considered inappropriate to assign a value less than 3 for the coworker data. Therefore, a GSD of 3 was assigned for any intake period in which the calculated GSD was <3 [10].

B.4.2 Bioassay Fitting and Intakes

IMBA was used to fit the bioassay results to a series of chronic inhalations. The intake assumptions were based on observed patterns in the bioassay data. Intervals with constant chronic intake rates were chosen by selecting periods during which the bioassay results were of similar magnitudes. A new chronic intake period was started where the data indicated a significant sustained change in the results. The effective bioassay dates that were used in IMBA to calculate the intake rates were the midpoints of the sampling periods [11]. The 50th- and 84th-percentile excretion values were fit as independent datasets producing separate 50th- and 84th-percentile intakes (ORAUT 2005a). The GSDs were calculated as the ratio of the 84th- to the 50th-percentile intakes for each intake interval (ORAUT 2005a).

This TBD does not specify inhalation absorption types for the various radionuclides or groups of radionuclides; therefore, the bioassay results were entered into IMBA with assumed lung absorption types chosen to be consistent with International Commission on Radiological Protection (ICRP) Publication 68 (ICRP 1995). The plots at the end of this attachment show the resultant 50th-

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percentile intakes. The bioassay data that were used in the fits are shown as solid blue dots (●) (dark spots when printed) and data that are not used in the fits are shown as red dots (●) (light dots when printed).

Some materials have very long radiological half-lives and/or are retained in the body for long periods. In such cases, the excretion results for different chronic intake periods are not independent. For example, an intake in the 1950s could contribute to urinary excretion in the 1980s and later. To avoid potential underestimation of intakes for people who worked at LLNL for relatively short periods, each chronic intake of long-lived or long-retained material was independently fit using only the bioassay results from the single intake period. This fitting method will result in a best estimate of dose if the person worked in only one period and a potential overestimate if an individual worked in multiple periods.

B.4.2.1 Uranium

The 50th- and 84th-percentile uranium urinary excretion concentrations in micrograms per liter were converted to milligrams per day assuming a daily urine volume of 1.4 L/d (ICRP 1975). These data are shown in Table B-8. The intakes were determined from the milligrams per day excretion values. For ease of use by dose reconstructors, the intakes were also converted to activity using the specific activity of NU (682 pCi/mg), which is favorable to claimants in comparison with the specific activity of DU. While workers at LLNL might not have been chronically exposed to NU, chronic intakes will approximate a series of acute intakes with unknown intake dates. Plots of the fits of models to urinary excretion are shown in Figures B-1 through B-24 at the end of this attachment.

The intake rates, GSDs, and periods in which they are applicable are given in Table B-8. Table B-9 lists the 50th- and 95th-percentile coworker uranium intakes to be assigned for dose reconstruction.

Table B-8. Uranium mass and activity intake rates for three different periods.[a]

Type	Dates	50th percentile (mg/d)	84th percentile (mg/d)	50th percentile (pCi/d)	84th percentile (pCi/d)	GSD
F	1/1/58–12/31/73	2.51E–02	5.81E–02	1.71E+01	3.96E+01	2.32[a]
F	1/1/74–12/31/87	1.65E–04	8.52E–04	1.13E–01	5.81E–01	5.16
F	1/1/88–12/31/96	2.87E–04	3.73E–04	1.96E–01	2.54E–01	1.30[a]
M	1/1/58–12/31/73	1.03E–01	2.39E–01	7.04E+01	1.65E+02	2.32[a]
M	1/1/74–12/31/87	6.76E–04	3.49E–03	4.61E–01	2.38E+00	5.16
M	1/1/88–12/31/96	1.20E–03	1.582E–03	8.16E–01	1.08E+00	1.32[a]
S	1/1/58–12/31/73	1.24E+00	2.93E+00	8.49E+02	1.99E+03	2.35[a]
S	1/1/74–12/31/87	8.55E–03	4.19E–02	5.83E+00	2.86E+01	4.90
S	1/1/88–12/31/96	1.75E–02	3.01E–02	1.19E+01	2.05E+01	1.72[a]

a. Use the default GSD of 3.0 instead of the calculated GSD.

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Table B-9. 50th- and 95th-percentile uranium mass and activity intake rates for three different periods.[a]

Type	Dates	50th percentile (mg/d)	50th percentile (pCi/d)	GSD	95th percentile (mg/day)	95th percentile (pCi/d)
F	1/1/58–12/31/73	2.51E–02	1.71E+01	3.00	1.53E–01	1.04E+02
F	1/1/74–12/31/87	1.65E–04	1.13E–01	5.16	2.45E–03	1.68E+00
F	1/1/88–12/31/96	2.87E–04	1.96E–01	3.00	1.75E–03	1.19E+00
M	1/1/58–12/31/73	1.03E–01	7.04E+01	3.00	6.28E–01	4.29E+02
M	1/1/74–12/31/87	6.76E–04	4.61E–01	5.16	1.01E–02	6.86E+00
M	1/1/88–12/31/96	1.20E–03	8.16E–01	3.00	7.31E–03	4.97E+00
S	1/1/58–12/31/73	1.24E+00	8.49E+02	3.00	7.56E+00	5.17E+03
S	1/1/74–12/31/87	8.55E–03	5.83E+00	4.90	1.17E–01	7.96E+01
S	1/1/88–12/31/96	1.75E–02	1.19E+01	3.00	1.07E–01	7.25E+01

a. Use the intakes in pCi/d assuming all the activity is from ^{234}U, which is favorable to claimants.

There were no bioassay results for uranium before 1958 in the MAPPER database and, according to the laboratory manager circa 1958, few or none might have been obtained, so workers exposed to uranium during 1952 through 1957 who did not have subsequent uranium bioassay would be considered unmonitored. Therefore, unmonitored uranium doses will not be assessed for the period before 1958, which is included in the designated SEC period.

B.4.2.2 Plutonium and Americium

Table B-10 provides the 50th- and 84th-percentile values for ^{239}Pu urinary excretion per day. Plots of the fits of models to urinary excretion are shown in Figures B-25 through B-47. The 50th- and 84th-percentile intakes based on urinary excretion modeled as ^{239}Pu are summarized in Tables B-11 and B-12. Either absorption type M or S might have been present in the LLNL workplace, so the dose reconstructor should use whichever is more favorable to the claimant. All GSDs are larger than 3, so they can be used directly from the table. The plutonium intakes in Tables B-11 and B-12 can be considered pure ^{239}Pu; that is, other isotopes in a plutonium mixture should be based on isotopic ratios from Table 5-2. Tables B-13 and B-14 list the 50th- and 95th-percentile coworker plutonium intakes to be assigned for dose reconstruction.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

Table B-10. 50th- and 84th-percentiles for ^{239}Pu[a] urine bioassay data for 1957 through 1996.

Year	Effective bioassay date	50th-percentile excretion rate (dpm/d)	84th-percentile excretion rate (dpm/d)
1957	7/1/1957	4.470E–02	2.829E–01
1958	7/1/1958	9.435E–02	1.002E+00
1959	7/1/1959	1.362E–01	1.361E+00
1960	7/1/1960	7.974E–02	3.566E–01
1961	7/1/1961	3.559E–02	4.136E–01
1962	7/1/1962	4.280E–02	1.113E+00
1963	7/1/1963	2.858E–02	2.956E–01
1964	7/1/1964	3.106E–02	1.133E–01
1965	7/1/1965	1.571E–02	1.547E–01
1966	7/1/1966	1.007E–02	1.547E–01
1967	7/1/1967	6.051E–03	3.878E–02
1968	7/1/1968	4.899E–03	2.202E–02
1969	7/1/1969	1.462E–03	1.526E–02
1970	7/1/1970	2.788E–04	6.192E–03
1971	7/1/1971	2.427E–03	2.898E–02
1972	7/1/1972	3.008E–03	2.499E–02
1973	7/1/1973	4.306E–03	3.266E–02
1974	7/1/1974	2.651E–03	2.339E–02
1975	7/1/1975	1.395E–02	9.313E–02
1976	7/1/1976	3.893E–03	5.441E–02
1977	7/1/1977	3.085E–03	2.961E–02
1978	7/1/1978	1.541E–03	3.706E–02
1979	7/1/1979	7.989E–03	4.731E–02
1980	7/1/1980	1.759E–02	1.376E–01
1981	7/1/1981	5.392E–03	3.582E–02
1982	7/1/1982	4.136E–04	7.478E–03
1983	7/1/1983	1.650E–03	1.078E–02
1984	7/1/1984	8.048E–04	9.488E–03
1985	7/1/1985	6.502E–04	7.737E–03
1986	7/1/1986	9.787E–04	1.257E–02
1987	7/1/1987	2.037E–03	9.097E–03
1988	7/1/1988	1.207E–03	9.843E–03
1989	7/1/1989	3.082E–03	1.816E–02
1990	7/1/1990	1.092E–03	1.479E–02
1991	7/1/1991	2.890E–04	3.935E–03
1992	7/1/1992	1.308E–04	3.465E–03
1993	7/1/1993	2.425E–04	4.860E–03
1994	7/1/1994	2.670E–04	5.789E–03
1995	7/1/1995	2.991E–04	5.973E–03
1996	7/1/1996	1.357E–04	4.863E–03

a. The data for 1970 were excluded from the analysis because they are anomalously low.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

Table B-11. 50th- and 84th-percentile type M ^{239}Pu intake rates from coworker plutonium urine bioassay data for 1957 through 1996.

From	To	50th percentile (dpm/day)	84th percentile (dpm/day)	GSD
1/1/1957	12/31/1960	22.97	123.02	5.36
1/1/1961	12/31/1963	8.09	123.02	15.20
1/1/1964	12/31/1964	8.09	26.19	3.24
1/1/1965	12/31/1967	1.88	26.19	13.95
1/1/1968	12/31/1968	1.88	6.32	3.37
1/1/1969	12/31/1981	0.82	6.32	7.69
1/1/1982	12/31/1990	0.23	1.88	8.11
1/1/1991	12/31/1996	0.04	0.98	22.59

Table B-12. 50th- and 84th-percentile type S ^{239}Pu intake rates from coworker plutonium urine bioassay data for 1957 through 1996.

From	To	50th percentile (dpm/day)	84th percentile (dpm/day)	GSD
1/1/1957	12/31/1960	580.54	2,280.00	3.93
1/1/1961	12/31/1963	126.67	2,280.00	18.00
1/1/1964	12/31/1965	126.67	1,203.20	9.50
1/1/1966	12/31/1968	47.40	163.17	3.44
1/1/1969	12/31/1969	13.41	163.17	12.16
1/1/1970	12/31/1981	13.41	120.09	8.95
1/1/1982	12/31/1990	4.24	33.96	8.01
1/1/1991	12/31/1996	0.88	20.58	23.38

Table B-13. 50th- and 95th-percentile type M ^{239}Pu intake rates from coworker plutonium urine bioassay data for 1957 through 1996.

From	To	50th percentile (dpm/d)	84th percentile (dpm/d)	GSD	95th percentile (dpm/d)
1/1/1957	12/31/1960	22.97	123.02	5.36	363.61
1/1/1961	12/31/1963	8.09	123.02	15.2	711.35
1/1/1964	12/31/1964	8.09	26.19	3.24	55.95
1/1/1965	12/31/1967	1.88	26.19	13.95	143.54
1/1/1968	12/31/1968	1.88	6.32	3.37	13.87
1/1/1969	12/31/1981	0.82	6.32	7.69	23.51
1/1/1982	12/31/1990	0.23	1.88	8.11	7.20
1/1/1991	12/31/1996	0.04	0.98	22.59	6.75

For assessing type SS solubility, refer to ORAUT-OTIB-0049, *Estimating Doses for Plutonium Strongly Retained in the Lung* (ORAUT 2010).

Table B-15 provides the 50th- and 84th-percentile values for ^{241}Am urinary excretion per day. Plots of the fits of models to urinary excretion are shown in Figures B-48 through B-54. Intakes were modeled assuming all of the americium/curium results were ^{241}Am. The biokinetic model for curium is basically the same as that for americium and produces doses comparable to those from ^{241}Am for longer-lived isotopes of curium (ICRP 1995). Therefore, using ^{241}Am as the modeled radionuclide is favorable to

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

Table B-14. 50th- and 95th-percentile type S ^{239}Pu intake rates from coworker plutonium urine bioassay data for 1957 through 1996.

From	To	50th percentile (dpm/d)	GSD	95th percentile (dpm/d)
1/1/1957	12/31/1960	580.54	3.93	5,515.79
1/1/1961	12/31/1963	126.67	18	14,709.47
1/1/1964	12/31/1965	126.67	9.5	5,140.78
1/1/1966	12/31/1968	47.4	3.44	361.76
1/1/1969	12/31/1969	13.41	12.16	816.85
1/1/1970	12/31/1981	13.41	8.95	493.38
1/1/1982	12/31/1990	4.24	8.01	129.97
1/1/1991	12/31/1996	0.88	23.38	157.12

claimants. The 50th- and 84th-percentile intakes based on urinary excretion modeled as ^{241}Am are summarized in Table B-16. Table B-17 lists the 50th- and 95th-percentile coworker americium intakes to be assigned for dose reconstruction. The coworker americium intakes should be assigned for cases in which there was potential for internal exposure from americium or curium. Separated americium was used in Building 251 as early as 1955 for tracer studies. Curium was used primarily in Building 251 around the same time frame as americium, and in Building 332. Americium coworker intakes should only be assigned in the case for work in Building 251. This can be determined based on the worker's job description, work location, claimant telephone interview, and DOE dosimetry records information. In the absence of specific information, the plutonium coworker intakes, as part of a weapons-grade mixture, should be assigned.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

Table B-15. 50th- and 84th-percentiles for ^{241}Am urine bioassay data for 1957 through 1996.

Year	Effective bioassay date	50th-percentile excretion rate (dpm/d)	84th-percentile excretion rate (dpm/d)
1957	7/1/1957	4.313E-02	2.785E-01
1958	7/1/1958	9.435E-02	1.002E+00
1959	7/1/1959	1.362E-01	1.361E+00
1960	7/1/1960	7.974E-02	3.566E-01
1961	7/1/1961	3.559E-02	4.136E-01
1962	7/1/1962	4.280E-02	1.113E+00
1963	7/1/1963	4.462E-02	4.147E-01
1964	7/1/1964	3.329E-02	1.106E-01
1965	7/1/1965	2.384E-02	1.910E-01
1966–67	12/31/1966	9.702E-03	1.660E-01
1968	7/1/1968	5.216E-03	2.327E-02
1969	7/1/1969	1.179E-03	1.462E-02
1970	7/1/1970	8.473E-05	2.928E-03
1971–72	12/31/1971	2.371E-03	2.889E-02
1973–75	7/1/1974	1.395E-02	1.283E-01
1976–78	7/1/1977	3.564E-02	7.259E-01
1979–81	7/1/1980	2.059E-02	1.807E-01
1982	7/1/1982	4.221E-02	2.041E-01
1983	7/1/1983	1.466E-02	9.062E-02
1984	7/1/1984	3.748E-02	2.872E-01
1985	7/1/1985	2.131E-02	1.345E-01
1986	7/1/1986	5.588E-02	2.278E-01
1987	7/1/1987	7.659E-02	5.604E-01
1988	7/1/1988	4.988E-02	3.161E-01
1989	7/1/1989	2.334E-02	2.421E-01
1990	7/1/1990	3.713E-02	2.369E-01
1991	7/1/1991	1.872E-02	1.024E-01
1992	7/1/1992	5.301E-02	2.260E-01
1993	7/1/1993	1.601E-02	1.010E-01
1994	7/1/1994	2.848E-02	2.101E-01
1995	7/1/1995	3.952E-02	1.900E-01
1996	7/1/1996	7.815E-02	3.557E-01

Table B-16. 50th- and 84th-percentile type M ^{241}Am intake rates from coworker plutonium urine bioassay data for 1957 through 1996.

From	To	50th percentile (dpm/d)	84th percentile (dpm/d)	GSD
1/1/1957	12/31/1960	10.1	43.8	4.36
1/1/1961	12/31/1965	3.48	43.8	12.6
1/1/1966	12/31/1970	0.49	5.72	11.8
1/1/1971	12/31/1996	1.93	12.7	6.54

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

Table B-17. 50th- and 95th-percentile type M ^{241}Am intake rates from coworker plutonium urine bioassay data for 1957 through 1996.

From	To	50th percentile (dpm/d)	GSD	95th percentile (dpm/d)
1/1/1957	12/31/1960	10.1	4.36	113.84
1/1/1961	12/31/1965	3.48	12.6	224.74
1/1/1966	12/31/1970	0.49	11.8	28.41
1/1/1971	12/31/1996	1.93	6.54	42.38

There were no bioassay results for plutonium or gross alpha before 1957 in the MAPPER database and, according to the [position redacted] circa 1958, few or none might have been obtained (ORAUT 2006b). Therefore, unmonitored plutonium doses will not be assessed for the period before 1957, which is included in the designated SEC period.

B.4.2.3 Gross Beta/MFP

Interpreting gross beta urinalyses requires:

- At least approximate knowledge of the ratio of the various possible fission/activation products at the time of intake,
- How the mix of those radionuclides will be altered by human biokinetics and be manifested in urine, and
- How the latter mix will contribute to gross beta counts given the impact of the chemistry and counting technique.

Note: See Section 5.2 for source term information.

Reactor Mixtures, 1974 to 1980
As shown in Figures B-55 and B-56, the gross beta urine data in Table B-18 were modeled as 100% type F ^{90}Sr. The resulting intakes should be multiplied by the activity fractions of ^{90}Sr in urine after intakes of 10-day old, 40-day-old, 180-day-old, and 1-year-old reactor fuel in ORAUT-OTIB-0054 (ORAUT 2015, Table 7-1a) for the ATR, N Reactor, and TRIGA reactor types. Table B-19 lists the 50- and 84th-percentile intake rates of type F ^{90}Sr for 1974 through 1980. The 50th- and 95th-percentile ^{90}Sr intake rates (Table B-20) should also be adjusted for the indicator radionuclide activity fractions in Table 7-1a in OTIB-0054, and should be used in conjunction with the associated radionuclides' ratios, excluding ^{131}I, for the ATR, N Reactor, and TRIGA reactor types for estimating intakes for relevant radionuclides based on intake activity relative to ^{90}Sr for 10-day old, 40-day-old, 180-day-old, and 1-year-old reactor fuel from Table 7-3a through 7-3c, and 7-3f through 7-3i in OTIB-0054. As noted previously, the LPTR operated from December of 1957 to March 31, 1980.

To summarize, the following method is used to assign internal dose from gross beta coworker intakes from reactor mixtures:

1. Determine the ^{90}Sr reactor mixture coworker intake rate by multiplying the ^{90}Sr coworker intake rate in Table B-20 by the ^{90}Sr average fraction of beta activity in urine sample for 10-day-old, 40-day-old, 180-day-old, and 1-year-old reactor fuel for the ATR, N Reactor, and TRIGA reactor types in Table 7-1a in OTIB-0054.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

2. Determine the associated radionuclides, excluding ^{131}I, using the ratios for estimating intakes for intakes based on intake activity relative to ^{90}Sr for 10-day-old, 40-day-old, 180-day-old, and 1-year-old reactor fuel for the ATR, N Reactor, and TRIGA reactor types in Tables 7-3a through 7-3c, and Tables 7-3f through 7-3i in OTIB-0054.

3. Assign the highest dose determined for the above decay times.

Table B-18. 50th- and 84th-percentiles for gross beta urine bioassay data for 1974 through 1995.

Year	Effective bioassay date	50th-percentile excretion rate (dpm/d)	84th-percentile excretion rate (dpm/d)
1974–75	12/31/1974	19.22	199.1
1976	07/01/1976	17.74	136.8
1977	07/01/1977	18.13	103.8
1978–79	12/31/1978	1.474	55.76
1980–81	12/31/1980	13.04	165.8
1982	07/01/1982	22.56	236.8
1983	07/01/1983	96.75	636.4
1984	07/01/1984	9.806	99.19
1985	07/01/1985	20.17	145.1
1986	07/01/1986	17.59	203.3
1987	07/01/1987	64.88	395.0
1988	07/01/1988	125.1	777.1
1989	07/01/1989	62.62	544.2
1990	07/01/1990	62.73	455.0
1991	07/01/1991	14.79	145.5
1992	07/01/1992	7.883	76.81
1993	07/01/1993	1.128	19.69
1994	07/01/1994	0.1914	4.956
1995	07/01/1995	0.2789	4.694

Table B-19. 50th- and 84th-percentile intake rates of type F ^{90}Sr for 1974 through 1980.[a]

From	To	50th percentile (dpm/d)	84th percentile (dpm/d)	GSD
1/1/1974	3/31/1980	62.72	605.52	9.65

a. Intake rates are unadjusted for the indicator radionuclide activity fractions in Table 7.1a in OTIB-0054.

The intake rates in Table B-20 are used in conjunction with OTIB-0054 to assign intakes of fission product mixtures from reactors based on the indicator radionuclide activity fractions and associated radionuclides ratios relative to ^{90}Sr for 10-day old reactor fuel for the ATR, N Reactor, and TRIGA reactor types using Table 7.1a, and Tables 7-3a through 7.3c and Tables 7.3f through 7.3i in OTIB-0054.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

Table B-20. 50th- and 95th-percentile type F ^{90}Sr intake rates for 1974 through 1980.

From	To	50th percentile (dpm/d)	GSD	95th percentile (dpm/d)
1/1/1974	3/31/1980	62.72	9.65	2611.88

Weapons Residue, 1974 to 1992

As shown in Figures B-57 and B-58, the gross beta urinary excretion for 1974 through 1992 was modeled as type F ^{103}Ru. Ruthenium-103 intakes were then assigned using 59% of the geometric means and 84th percentiles. Additional intakes of type S ^{141}Ce and type F ^{89}Sr were included at 50% each of the ^{103}Ru intakes. Tungsten-181 is not a beta emitter and would not have been detected in the gross beta urinalysis (Kocher 1981); however, the additional dose to each organ from using ^{103}Ru compensates for the unmeasured intake or dose from ^{181}W. Table B-21 lists the 50th- and 84th-percentile ^{103}Ru intakes. Table B-22 lists the 50th- and 84th-percentile ^{141}Ce, and ^{89}Sr intakes. Table B-23 lists the 50th- and 95th-percentile ^{103}Ru intakes to be assigned for dose reconstruction. Table B-24 lists the 50th- and 95th-percentile ^{141}Ce and ^{89}Sr intakes to be assigned for dose reconstruction.

Table B-21. 50th- and 84th-percentile intake rates of type F ^{103}Ru for 1974 through 1992.

From	To	50th percentile (dpm/d)	84th percentile (dpm/d)	GSD
1/1/1974	12/31/1982	62.72	605.52	9.65
1/1/1983	12/31/1983	411.48	2,706.57	6.58
1/1/1984	12/31/1986	62.72	605.52	9.65
1/1/1987	12/31/1987	269.51	1,975.44	7.33
1/1/1988	12/31/1988	531.74	3,304.41	6.21
1/1/1989	12/31/1990	269.51	1,975.44	7.33
1/1/1991	12/31/1992	62.72	605.52	9.65

Aged Fission Products and Research-Related Radionuclides, After 1992

As shown in Figures B-59 and B-60, the gross beta urinary excretion data for 1993 to 1995 were modeled as type F ^{90}Sr. Intakes of ^{32}P, ^{14}C, ^{137}Cs, and ^{90}Y equal to the ^{90}Sr intakes were then applied. Table B-25 lists the 50th- and 84th-percentile ^{90}Sr, ^{90}Y, ^{32}P, ^{137}Cs, and ^{14}C intakes. Table B-22 lists the 50th- and 84th-percentile ^{141}Ce and ^{89}Sr intakes. Table B-26 lists the 50th- and 95th-percentile ^{90}Sr, ^{90}Y, ^{32}P, ^{137}Cs, and ^{14}C intakes to be assigned for dose reconstruction.

Table B-22. 50th- and 84th-percentile intake rates of type S ^{141}Ce and type F ^{89}Sr for 1974 through 1992.

From	To	50th percentile (dpm/d)	84th percentile (dpm/d)	GSD
1/1/1974	12/31/1982	31.36	302.76	9.65
1/1/1983	12/31/1983	205.74	1,353.28	6.58
1/1/1984	12/31/1986	31.36	302.76	9.65
1/1/1987	12/31/1987	134.76	987.72	7.33
1/1/1988	12/31/1988	265.87	1,652.21	6.21
1/1/1989	12/31/1990	134.76	987.72	7.33
1/1/1991	12/31/1992	31.36	302.76	9.65

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

Table B-23. 50th- and 95th-percentile type F ^{103}Ru intake rates for 1974 through 1992. These intake rates are used to assign intakes of fission and activation product mixtures from weapons test residuals.

From	To	50th percentile (dpm/d)	GSD	95th percentile (dpm/d)
1/1/1974	12/31/1982	62.72	9.65	2,611.88
1/1/1983	12/31/1983	411.48	6.58	9,127.05
1/1/1984	12/31/1986	62.72	9.65	2,611.88
1/1/1987	12/31/1987	269.51	7.33	7,139.55
1/1/1988	12/31/1988	531.74	6.21	10,723.46
1/1/1989	12/31/1990	269.51	7.33	7,139.55
1/1/1991	12/31/1992	62.72	9.65	2,611.88

Table B-24. 50th- and 95th-percentile type S ^{141}Ce and type F ^{89}Sr intake rates for 1974 through 1992. These intake rates are used to assign intakes of fission and activation product mixtures from weapons test residuals. Note that the indicated intake rate is assigned for each radionuclide.

From	To	50th percentile (dpm/d)	GSD	95th percentile (dpm/d)
1/1/1974	12/31/1982	31.36	9.65	1,305.94
1/1/1983	12/31/1983	205.74	6.58	4,563.52
1/1/1984	12/31/1986	31.36	9.65	1,305.94
1/1/1987	12/31/1987	134.76	7.33	3,569.91
1/1/1988	12/31/1988	265.87	6.21	5,361.73
1/1/1989	12/31/1990	134.76	7.33	3,569.91
1/1/1991	12/31/1992	31.36	9.65	1,305.94

Table B-25. 50th- and 84th-percentile intake rates of type F ^{90}Sr, ^{90}Y, ^{32}P, ^{137}Cs, and ^{14}C for after 1992. (Note that the indicated intake rate is assigned for each radionuclide.)[a]

From	To	50th percentile (dpm/d)	84th percentile (dpm/d)	GSD
01/01/1993	12/31/1993	5.11	89.5	17.5
01/01/1994	12/31/1995	0.909	19.1	20.9

a. Refer to Section 5.3.2.3 for solubility types for Y-90, P-32, Cs-137, and C-14.

Table B-26. 50th- and 95th-percentile type F ^{90}Sr, ^{90}Y, ^{32}P, ^{137}Cs, and ^{14}C intake rates for after 1992. (Note that the indicated intake rate is assigned for each radionuclide.)[a]

From	To	50th percentile (dpm/d)	GSD	95th percentile (dpm/d)
01/01/1993	12/31/1993	5.11	17.5	566.52
01/01/1994	12/31/1995	0.909	20.9	134.96

a. Refer to Section 5.3.2.3 for solubility types for Y-90, P-32, Cs-137, and C-14.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

Because there were few bioassays before 1974, guidance on assigning intakes of beta/gamma-emitting radionuclides before 1974 were not developed. Intakes and doses before 1974 are covered under the SEC designation (Leavitt 2008).

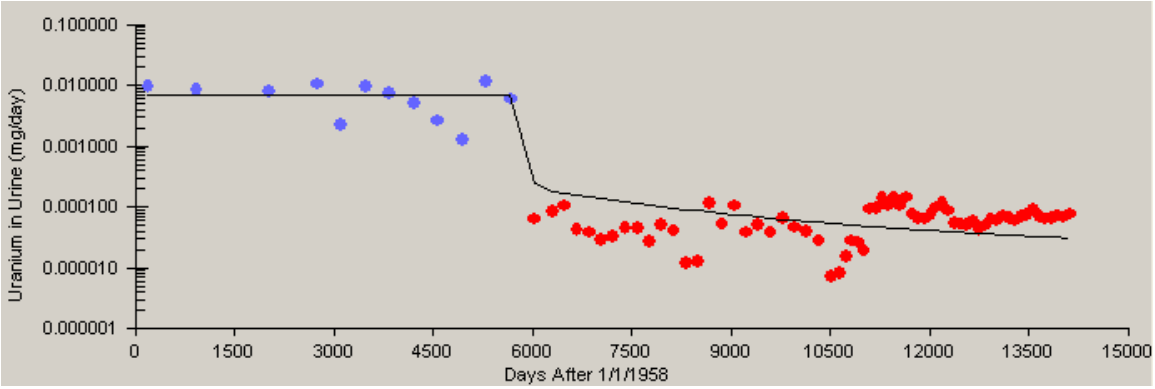

Figure B-1. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type F NU from January 1958 through December 1973 (blue dots).

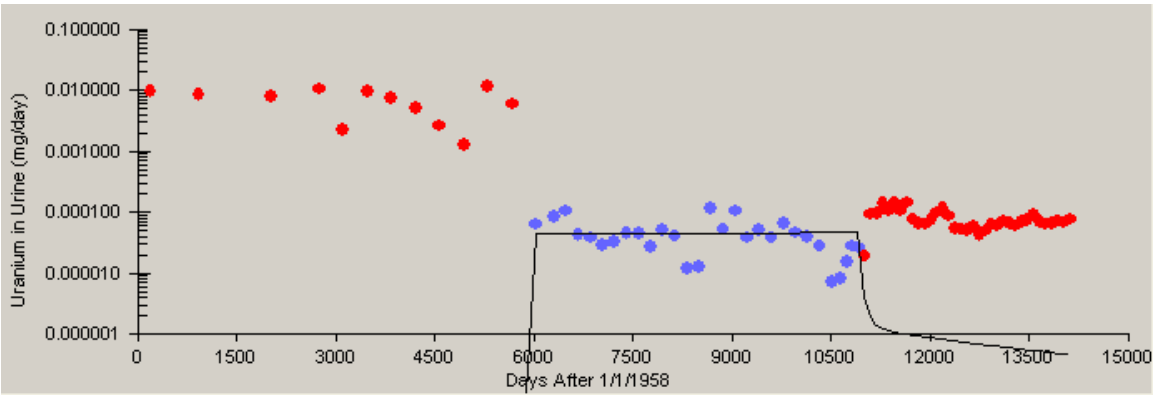

Figure B-2. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type F NU from January 1974 through December 1987 (blue dots).

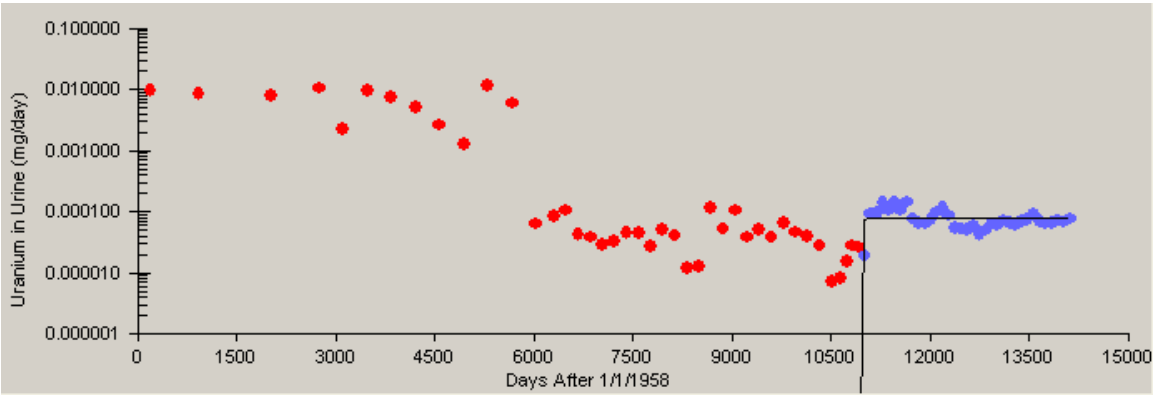

Figure B-3. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type F NU from January 1988 through December 1996 (blue dots).

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

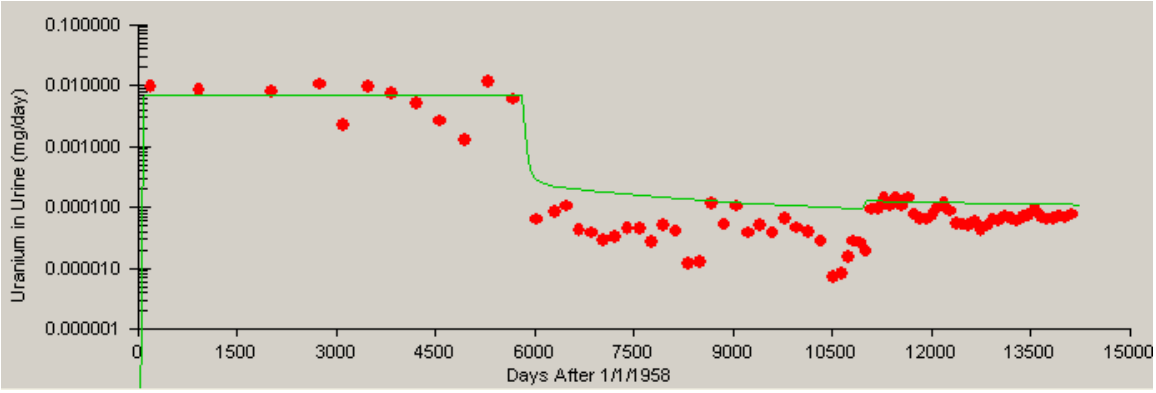

Figure B-4. Predicted and observed 50th-percentile urinary excretion assuming three separate chronic inhalation intakes of type F NU.

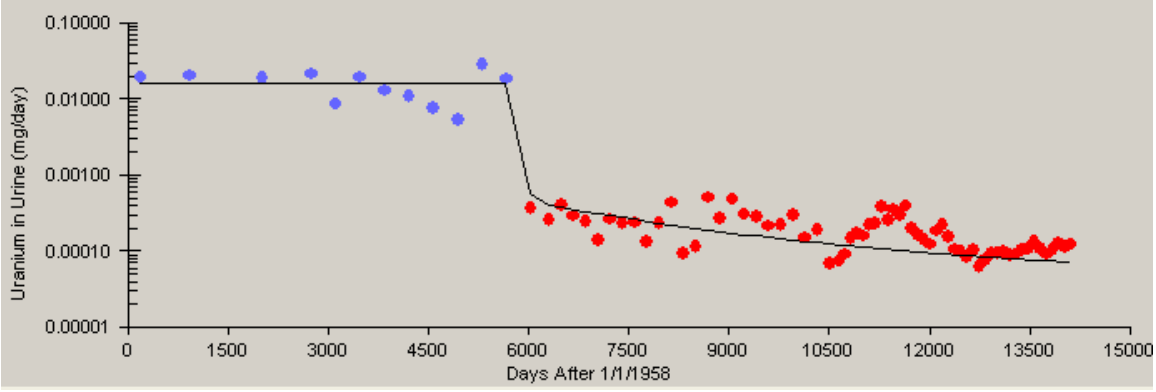

Figure B-5. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type F NU from January 1958 through December 1973 (blue dots).

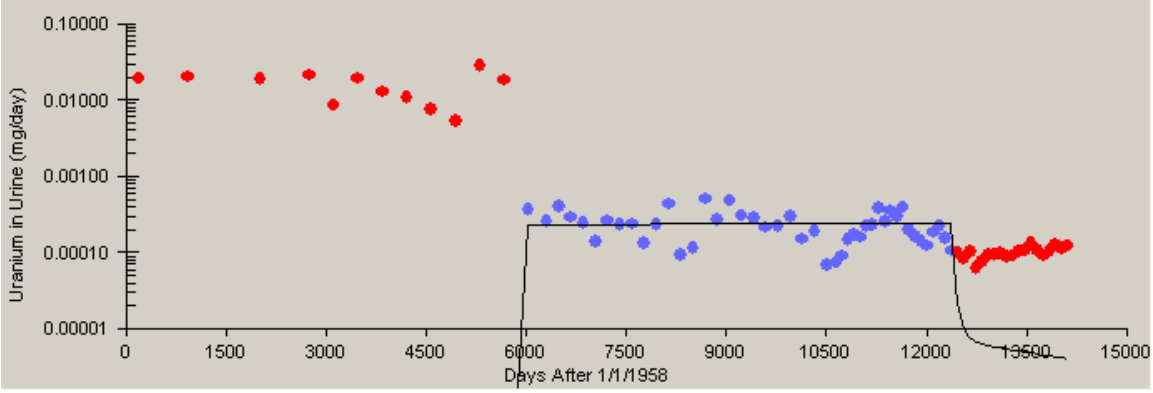

Figure B-6. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type F NU from January 1974 through December 1987 (blue dots).

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

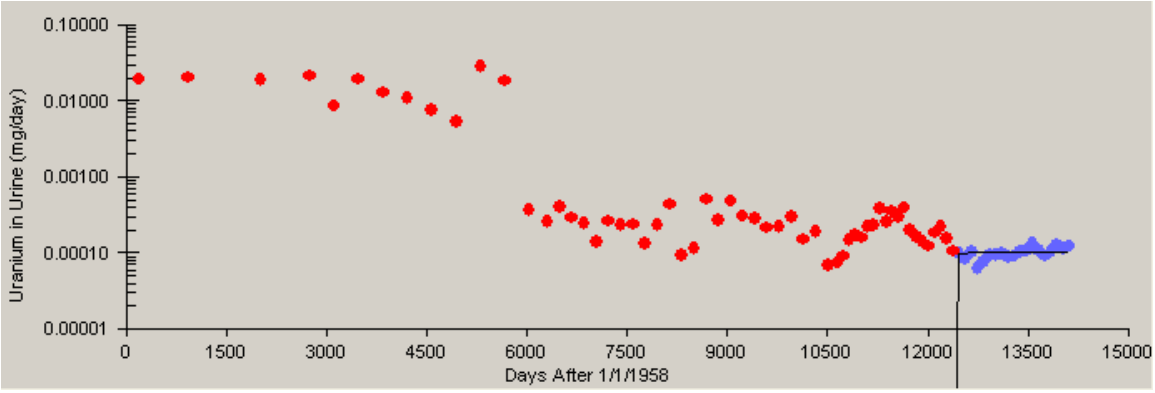

Figure B-7. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type F NU from January 1988 through December 1996 (blue dots).

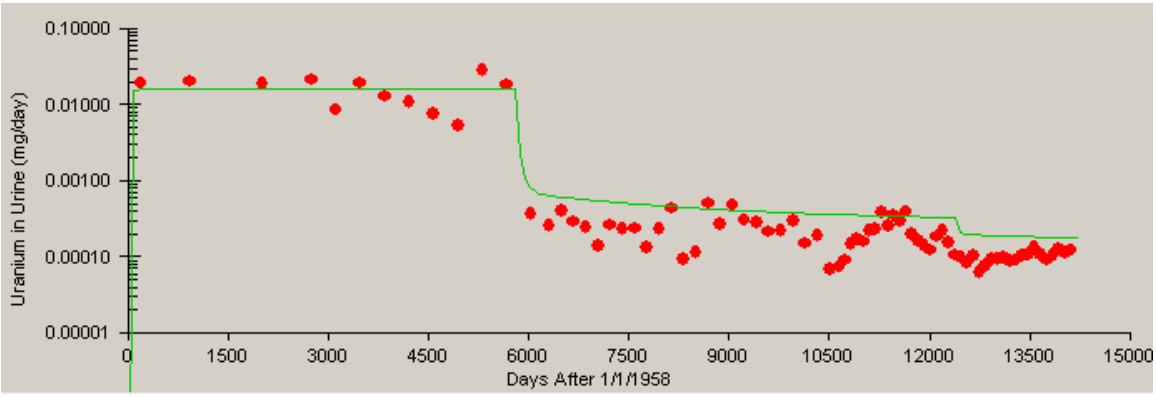

Figure B-8. Predicted and observed 84th-percentile urinary excretion assuming three separate chronic inhalation intakes of type F NU.

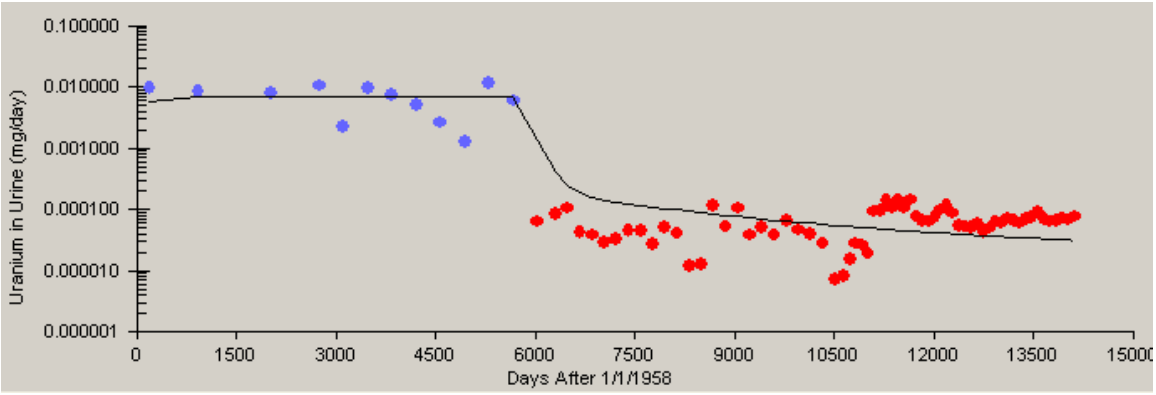

Figure B-9. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M NU from January 1958 through December 1973 (blue dots).

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

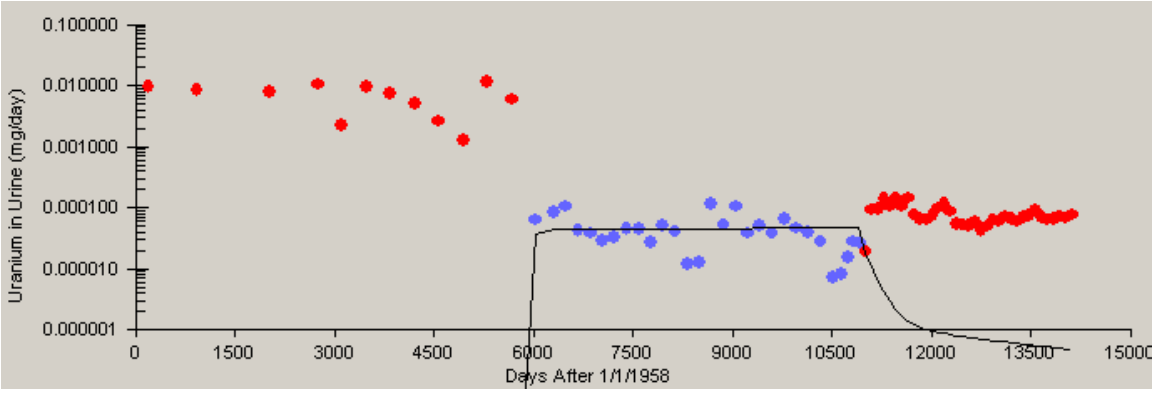

Figure B-10. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M NU from January 1974 through December 1987 (blue dots).

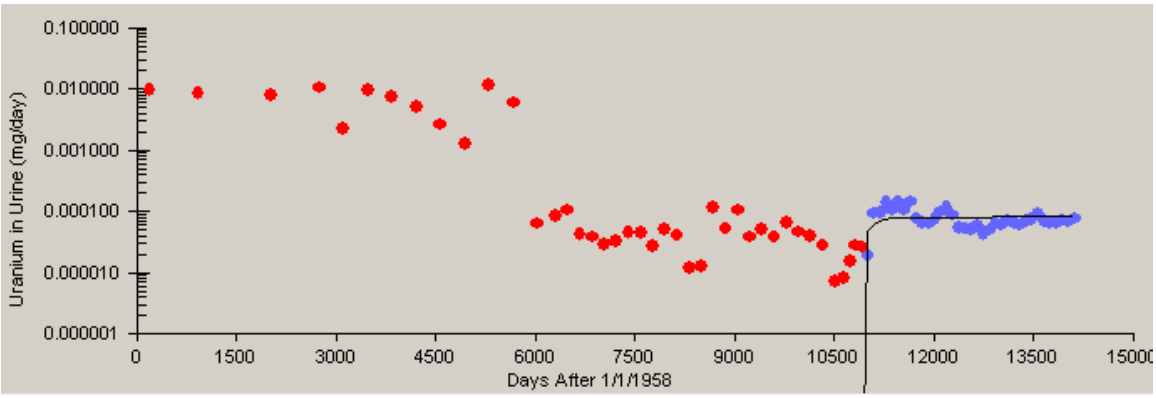

Figure B-11. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M NU from January 1988 through December 1996 (blue dots).

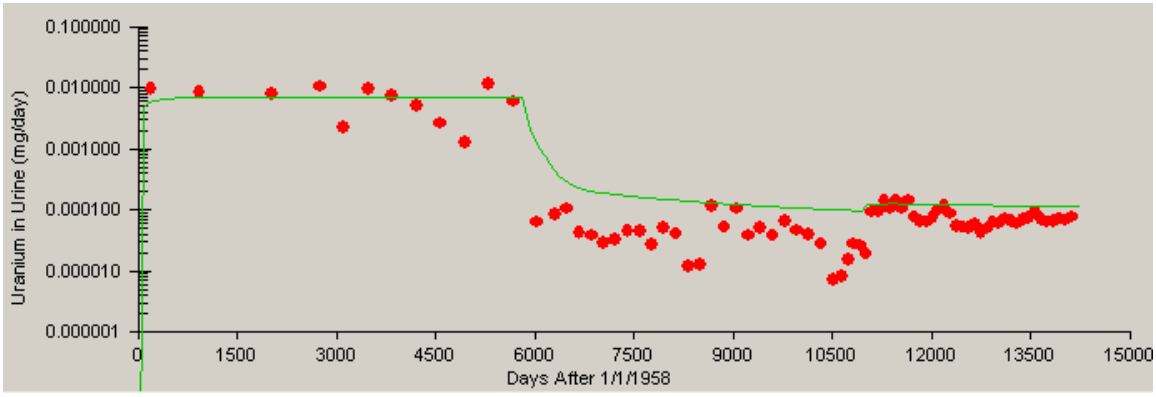

Figure B-12. Predicted and observed 50th-percentile urinary excretion assuming three separate chronic inhalation intakes of type M NU.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

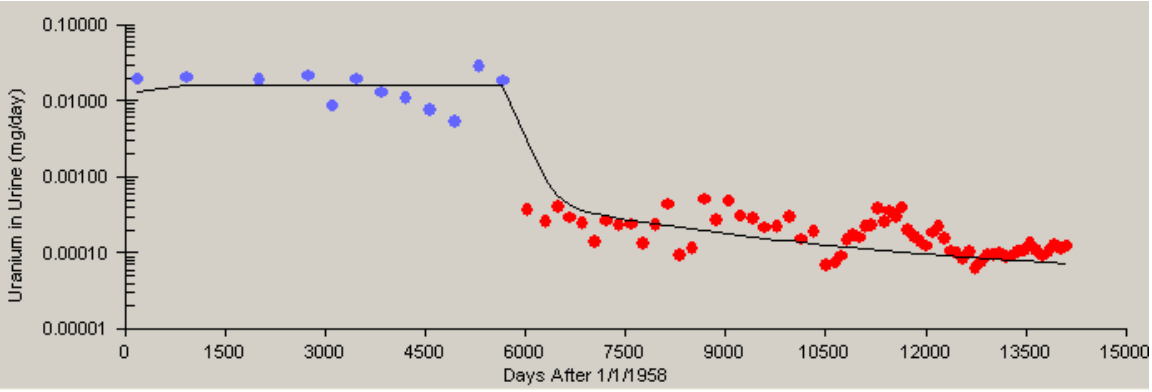

Figure B-13. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M NU from January 1958 through December 1973 (blue dots).

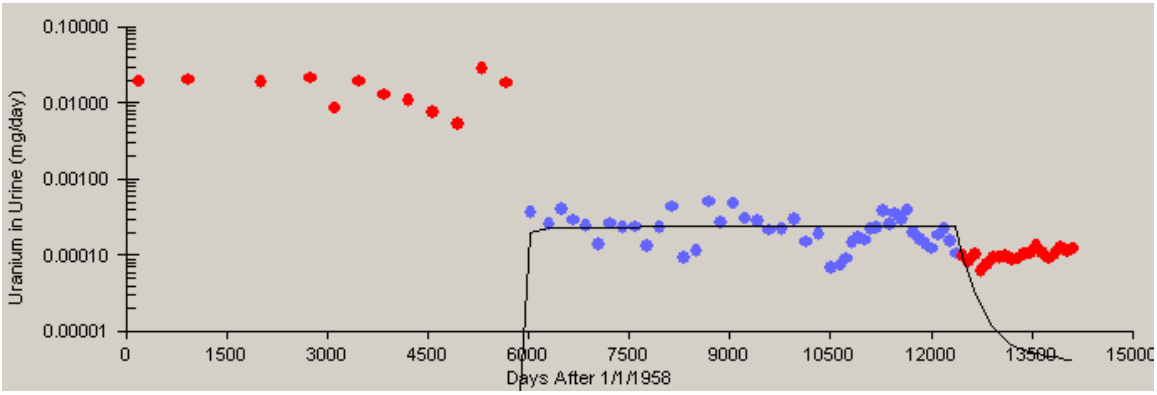

Figure B-14. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M NU from January 1974 through December 1987 (blue dots).

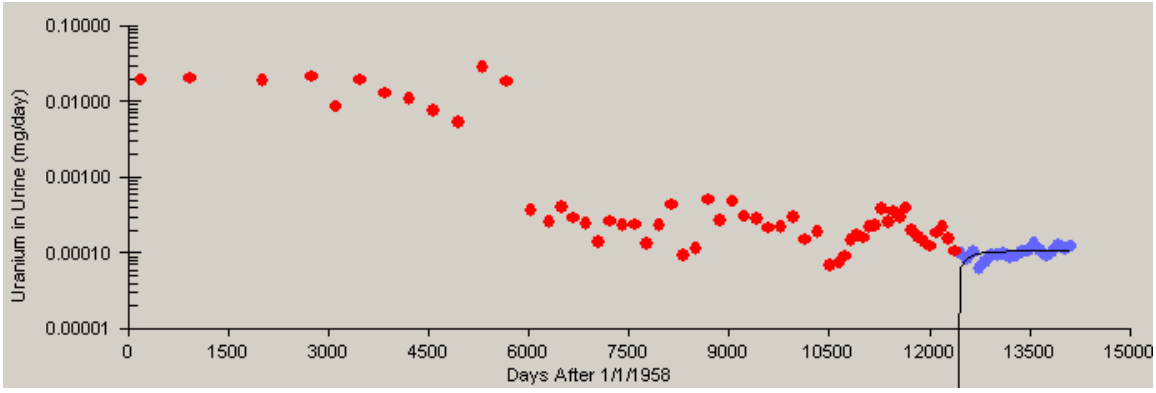

Figure B-15. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M NU from January 1988 through December 1996 (blue dots).

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

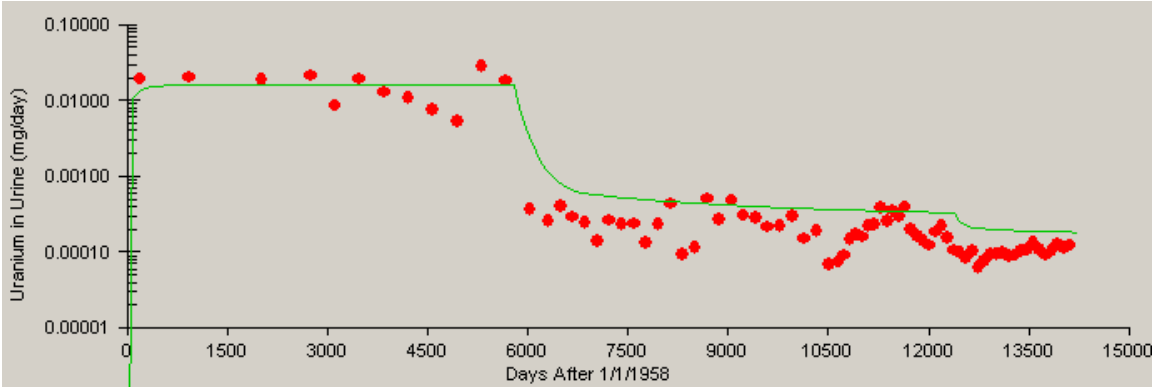

Figure B-16. Predicted and observed 84th-percentile urinary excretion assuming three separate chronic inhalation intakes of type M NU (blue dots).

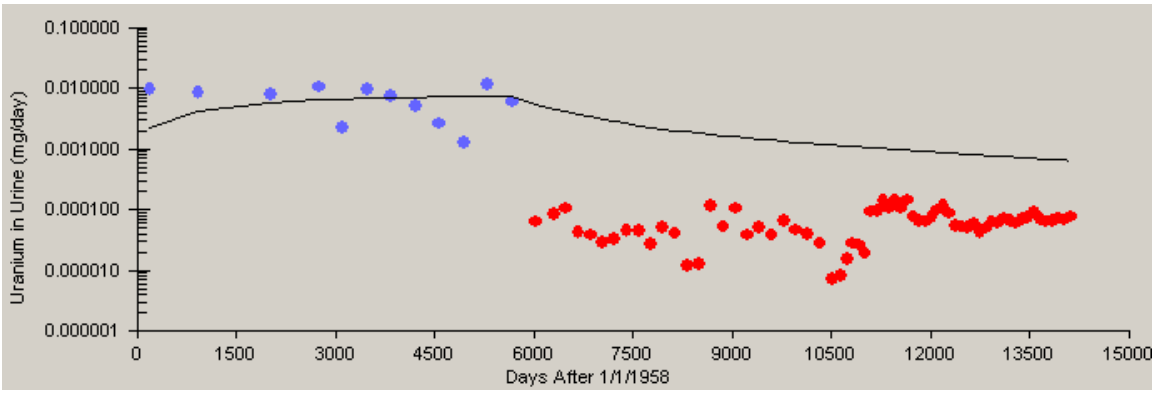

Figure B-17. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type S NU from January 1959 through December 1973 (blue dots).

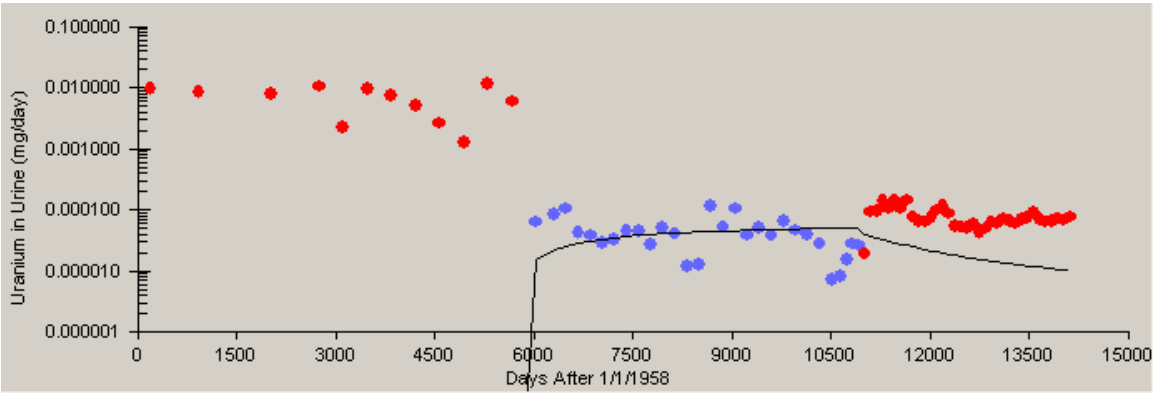

Figure B-18. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type S NU from January 1974 through December 1987 (blue dots).

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

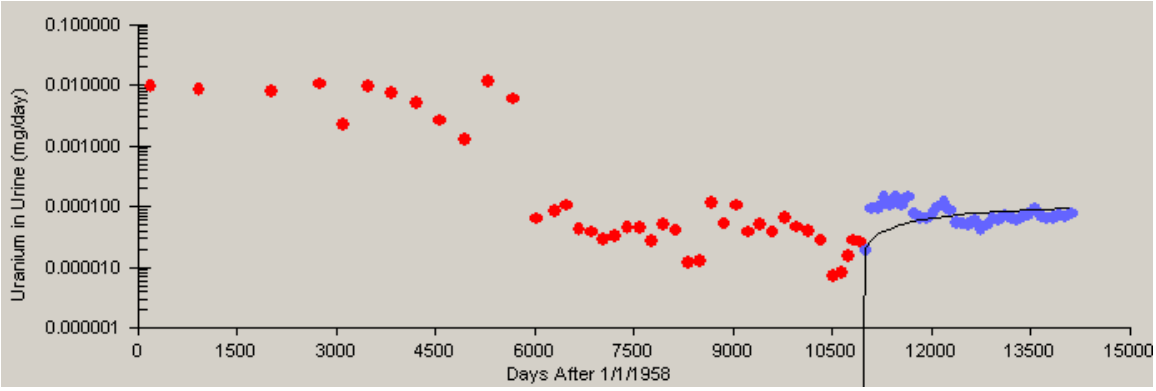

Figure B-19. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type S NU from January 1988 through December 1996 (blue dots).

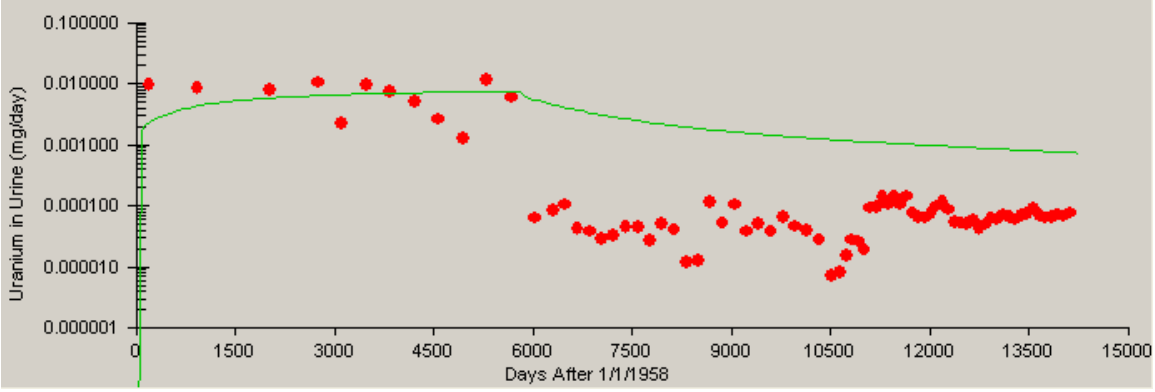

Figure B-20. Predicted and observed 50th-percentile urinary excretion assuming three separate chronic inhalation intakes of type S NU.

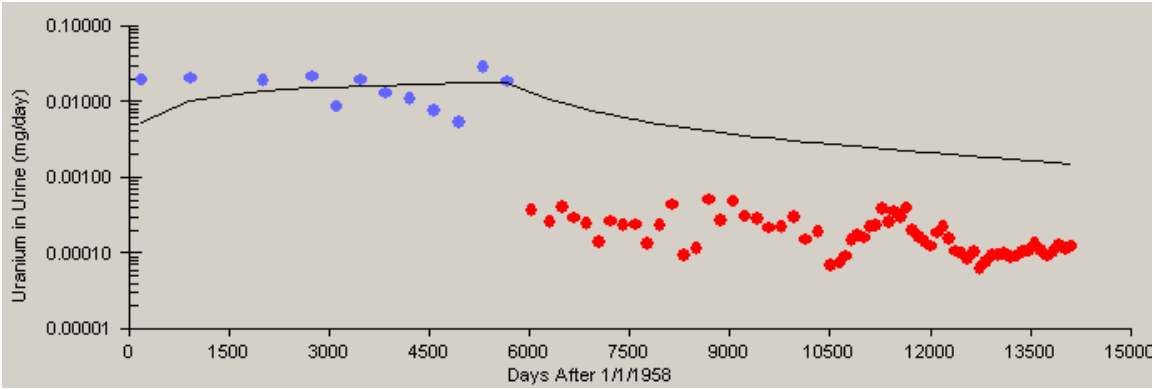

Figure B-21. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type S NU from January 1958 through December 1973 (blue dots).

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

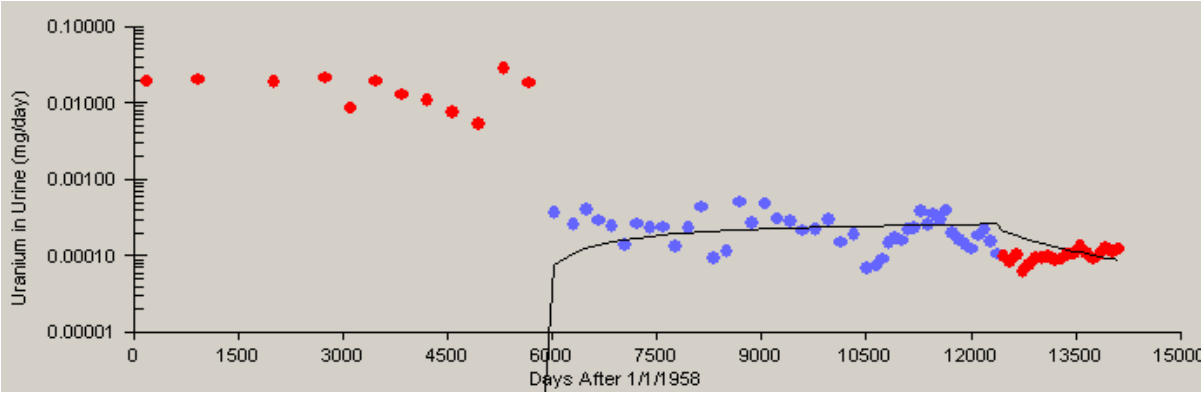

Figure B-22. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type S NU from January 1974 through December 1987 (blue dots).

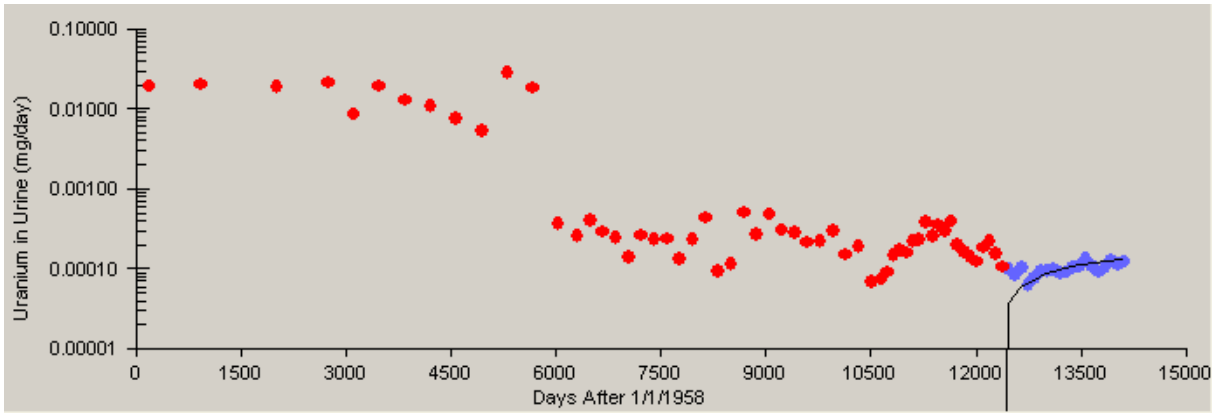

Figure B-23. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type S NU from January 1988 through December 1996 (blue dots).

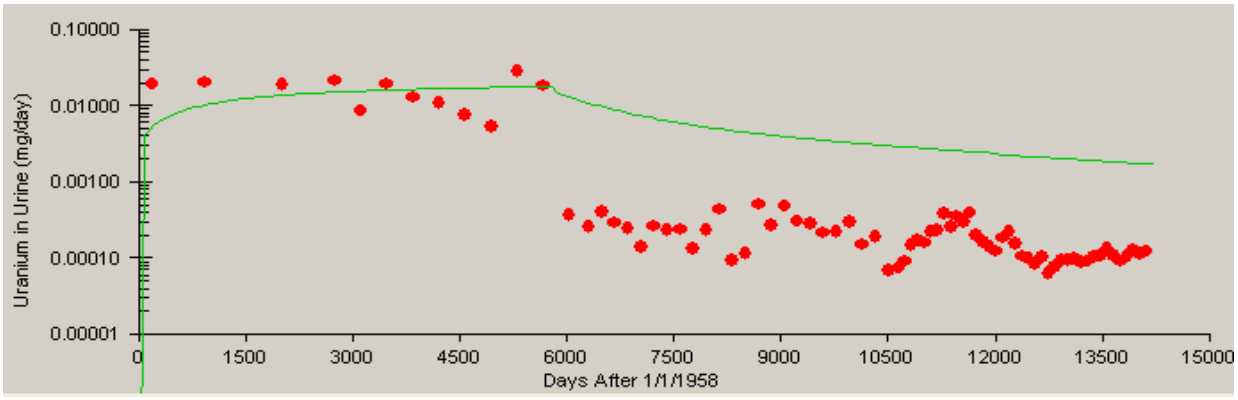

Figure B-24. Predicted and observed 84th-percentile urinary excretion assuming three separate chronic inhalation intakes of type S NU.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

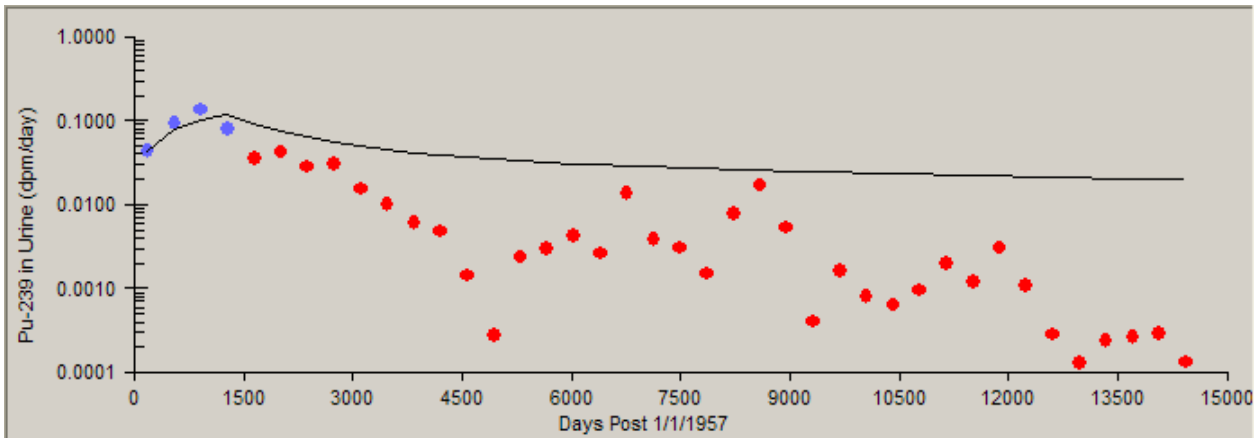

Figure B-25. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1957 through December 1960.

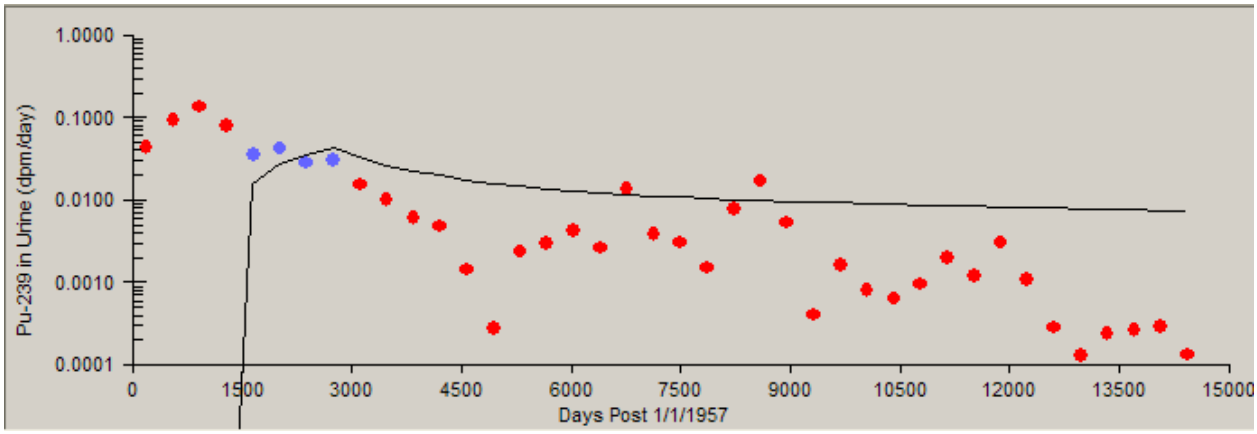

Figure B-26. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1961 through December 1964.

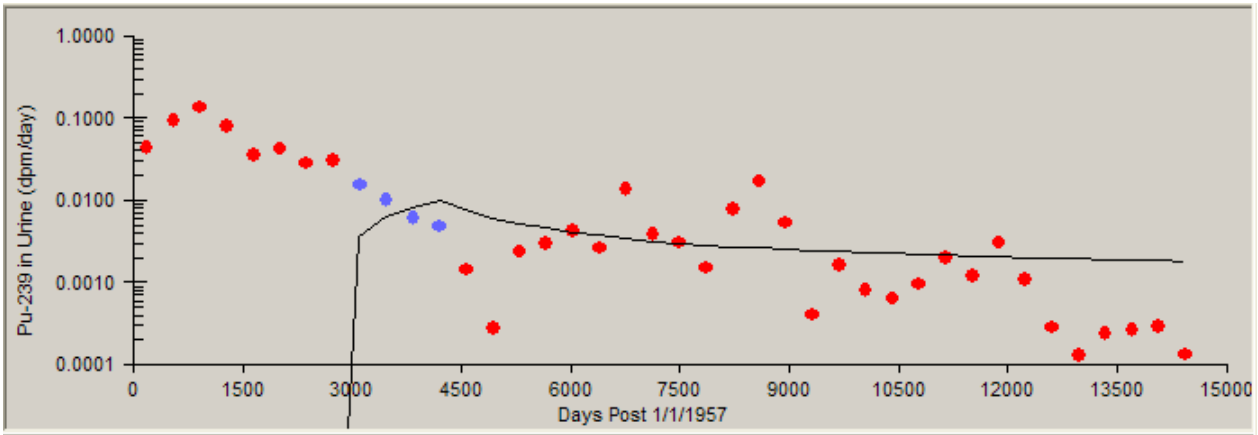

Figure B-27. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1965 through December 1968.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

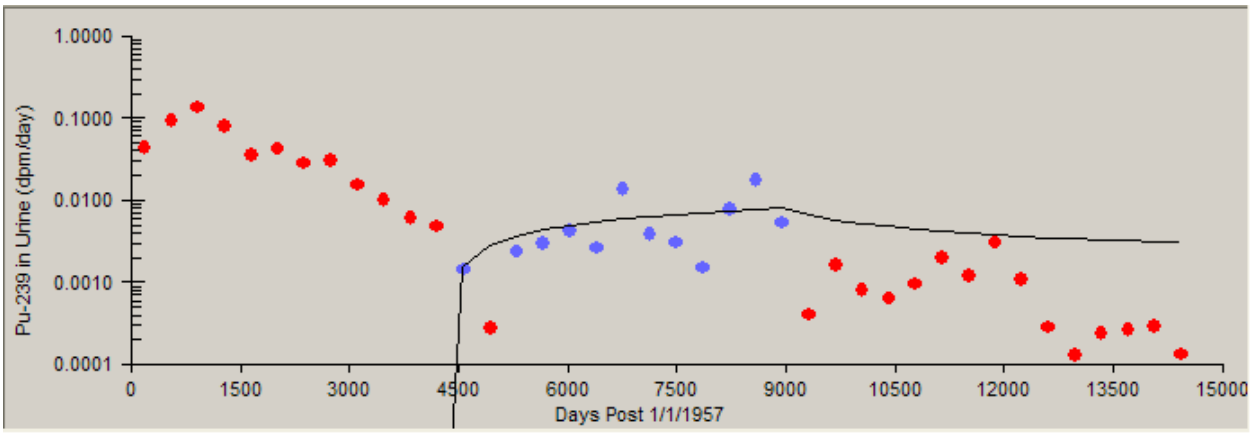

Figure B-28. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1969 through December 1981.

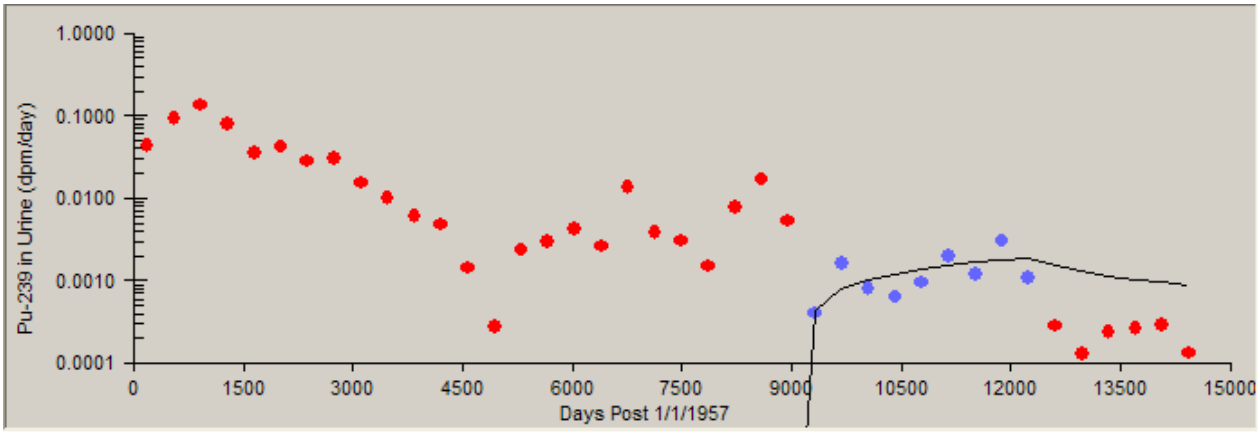

Figure B-29. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1982 through December 1990.

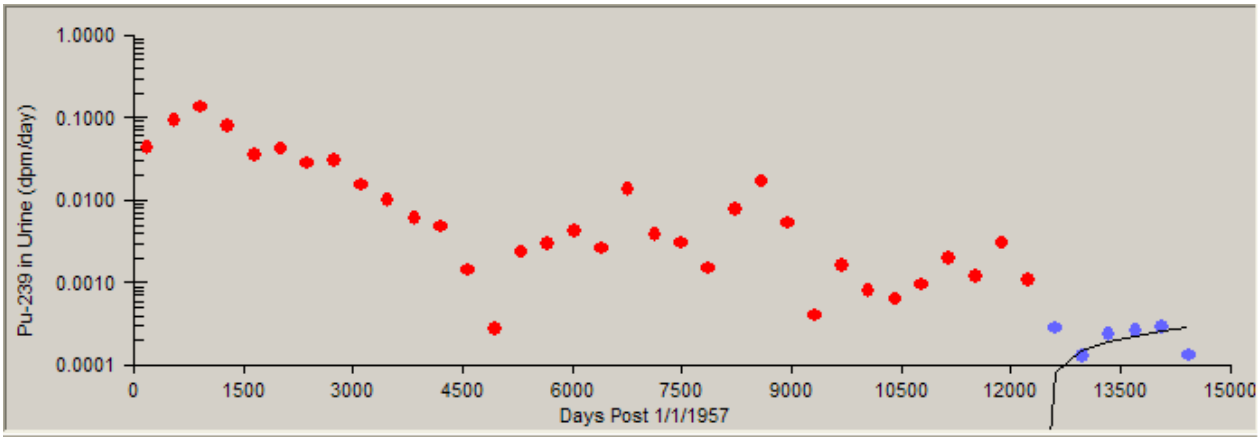

Figure B-30. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1957 through December 1996.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

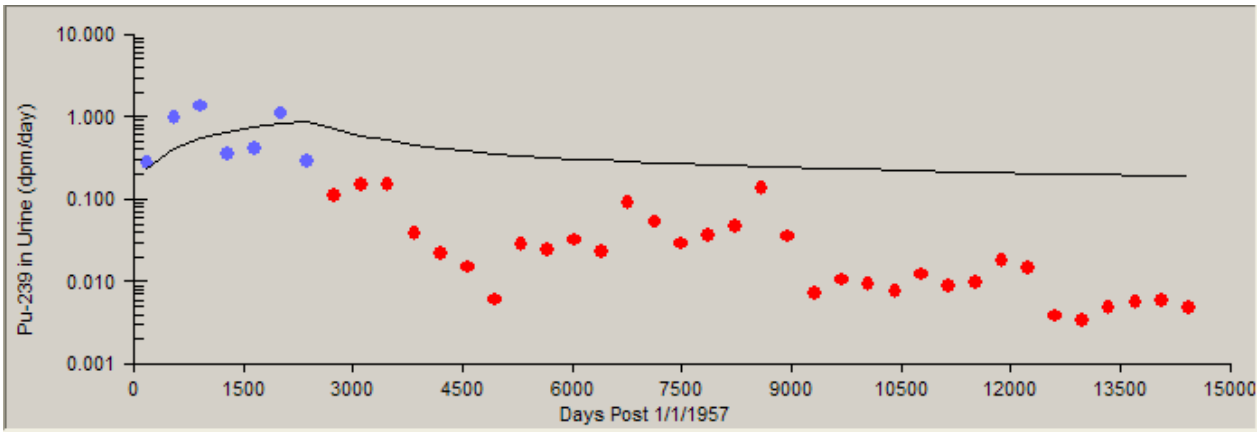

Figure B-31. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1957 through December 1963.

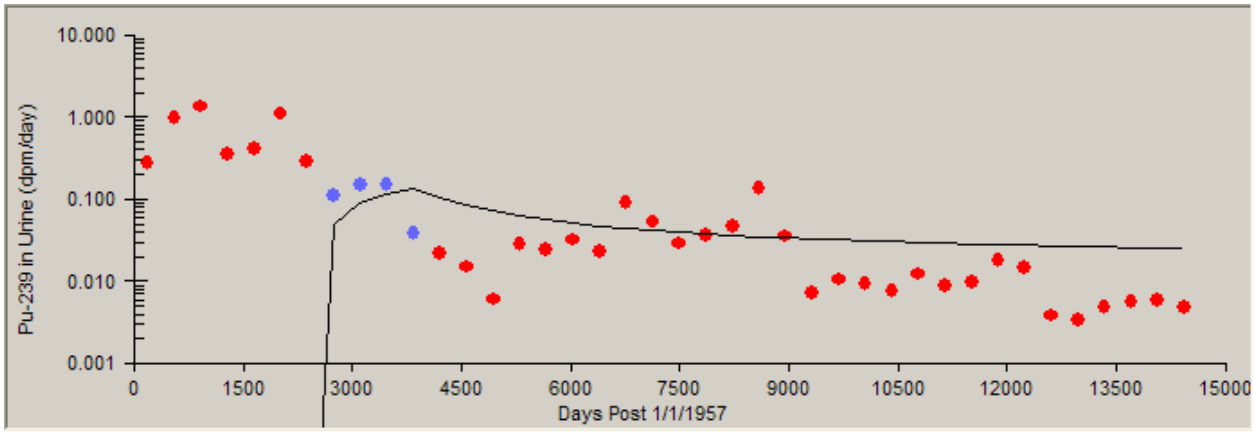

Figure B-32. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1964 through December 1967.

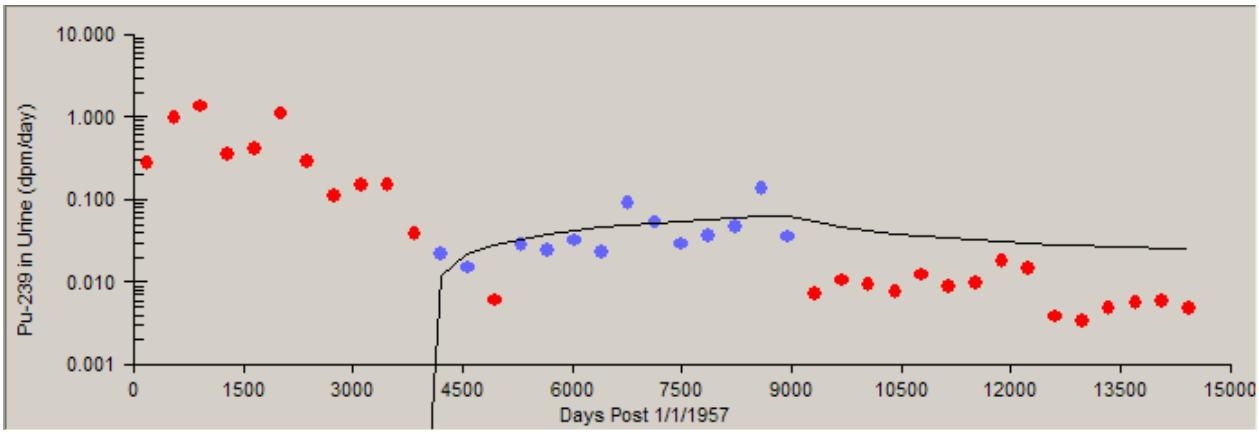

Figure B-33. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1968 through December 1981.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

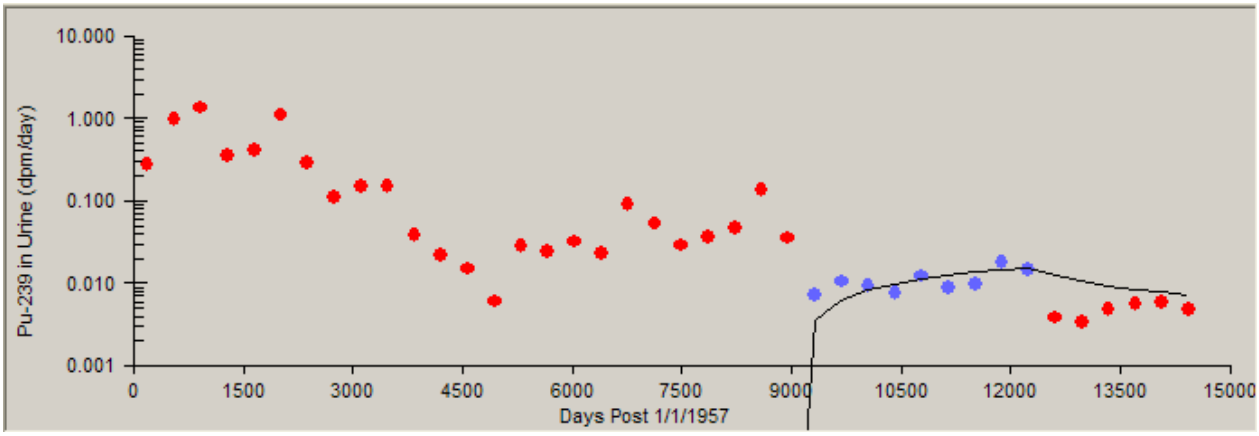

Figure B-34. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1982 through December 1990.

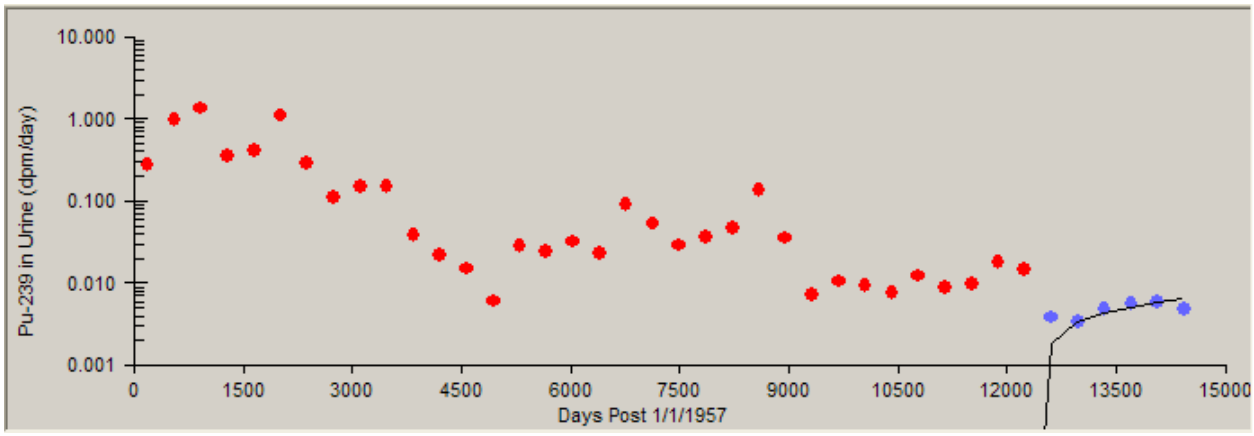

Figure B-35. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M plutonium from January 1991 through December 1996.

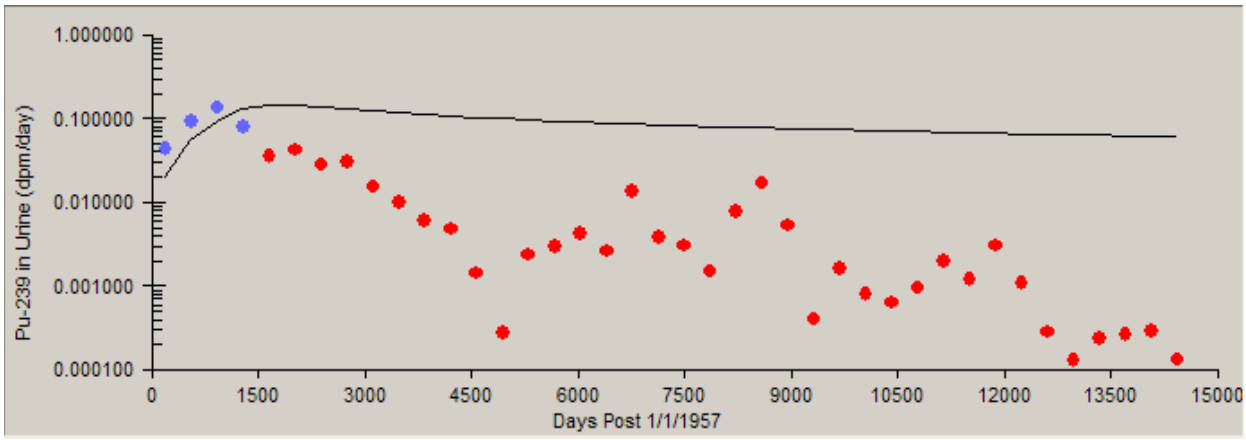

Figure B-36. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1957 through December 1960.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

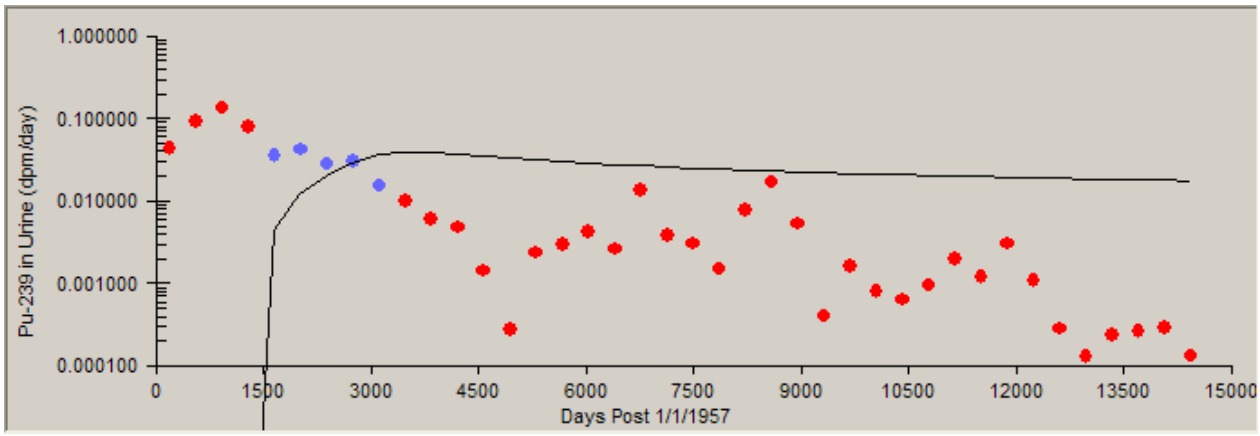

Figure B-37. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1961 through December 1965.

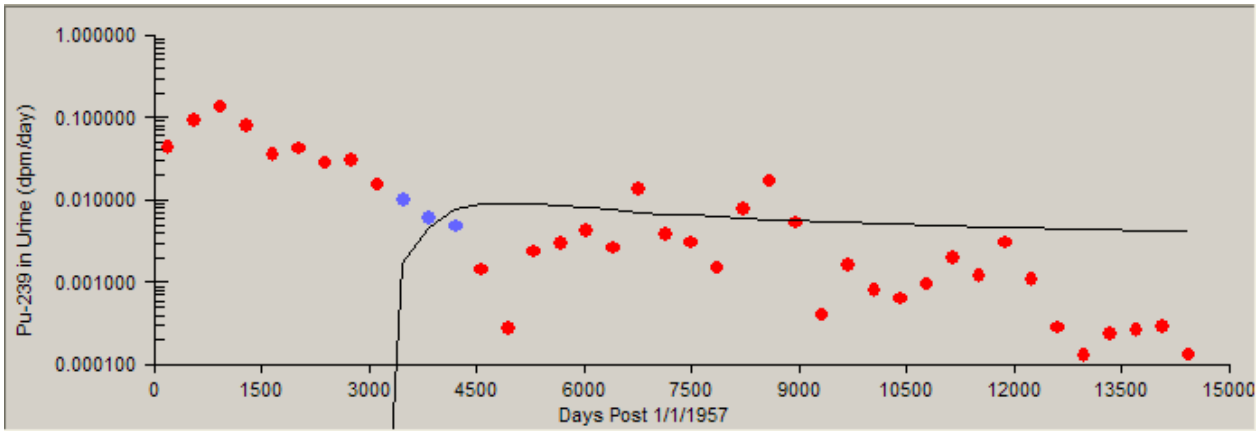

Figure B-38. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1966 through December 1968.

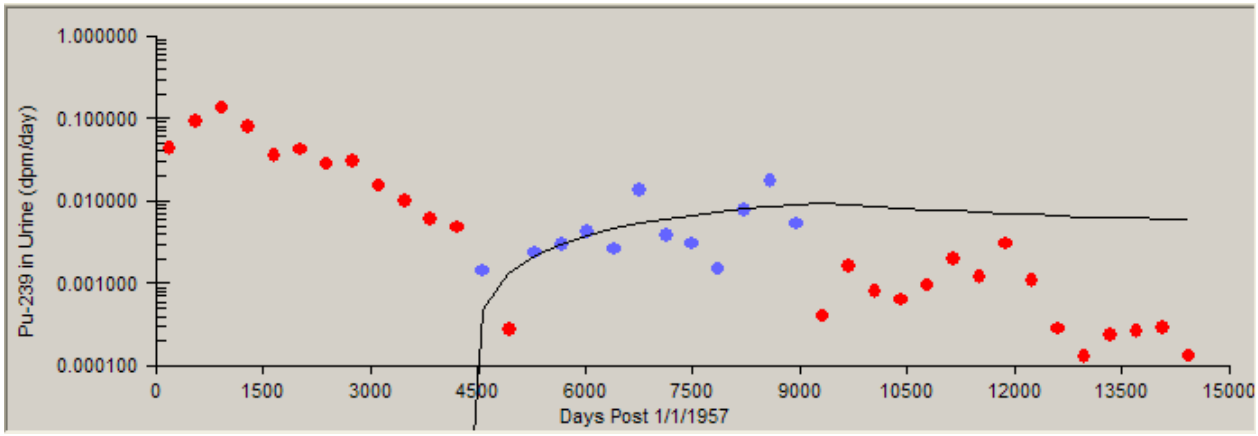

Figure B-39. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1969 through December 1981.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

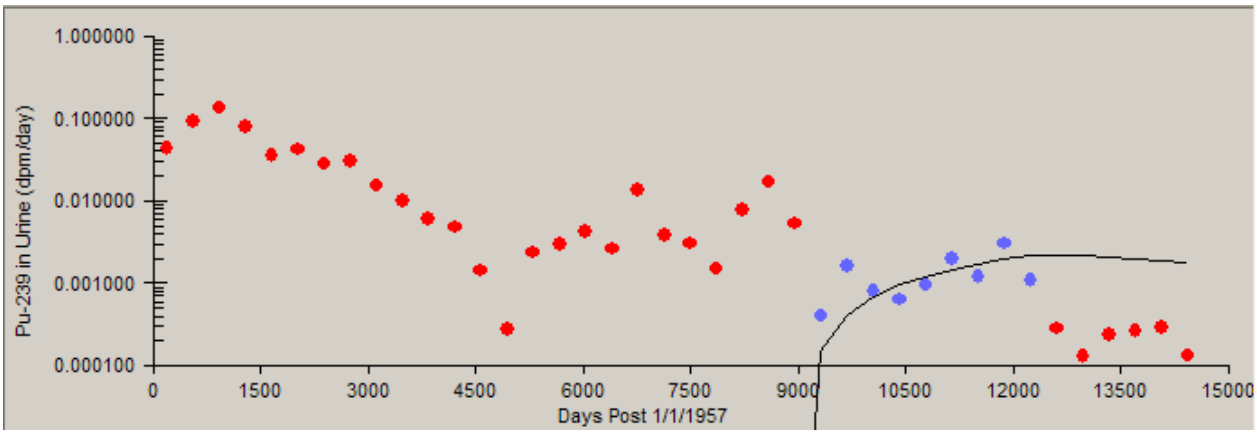

Figure B-40. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1982 through December 1990.

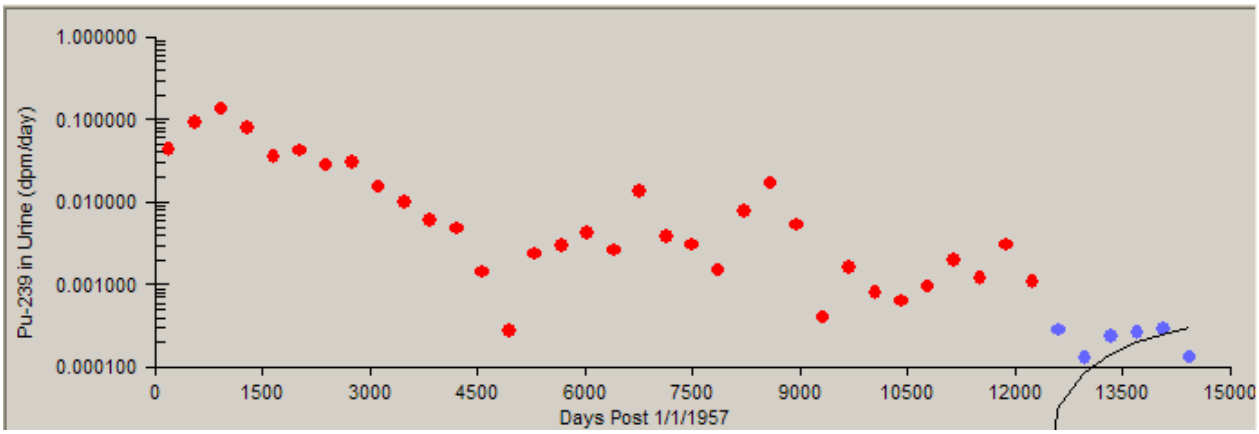

Figure B-41. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1991 through December 1996.

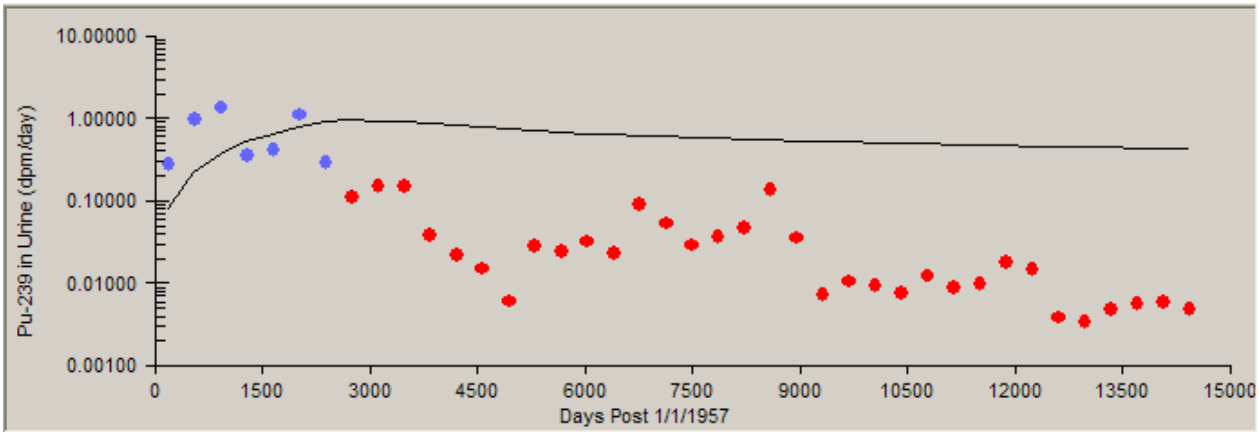

Figure B-42. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1957 through December 1963.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

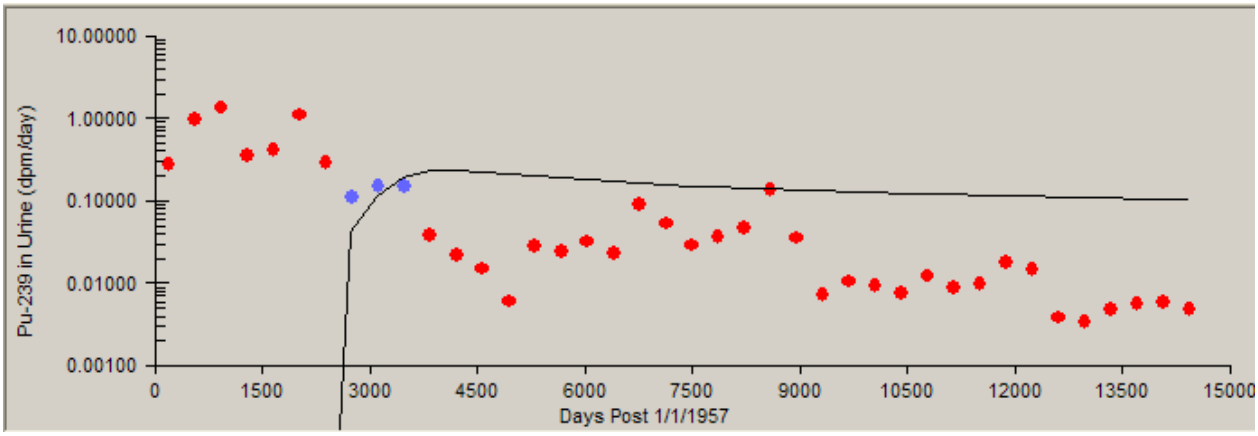

Figure B-43. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1964 through December 1966.

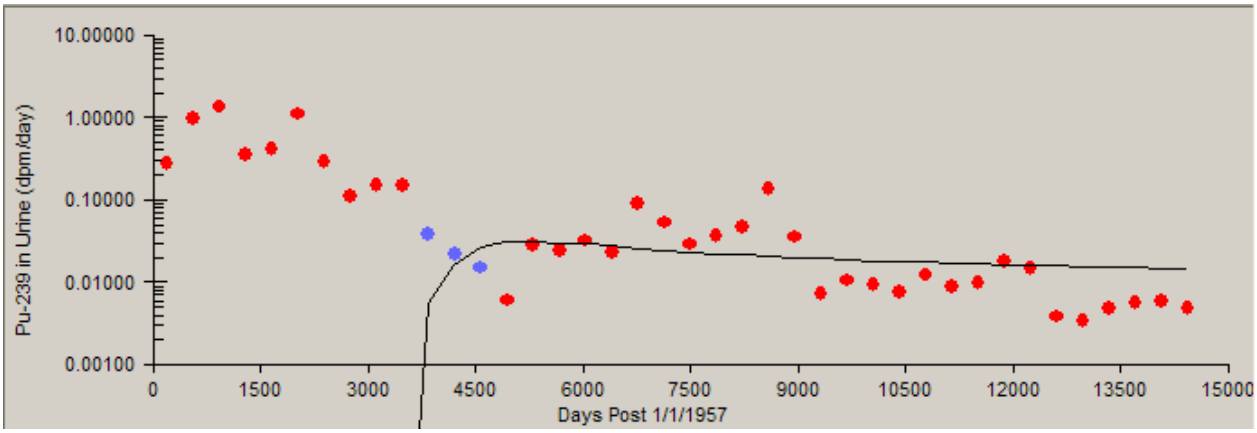

Figure B-44. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1967 through December 1969.

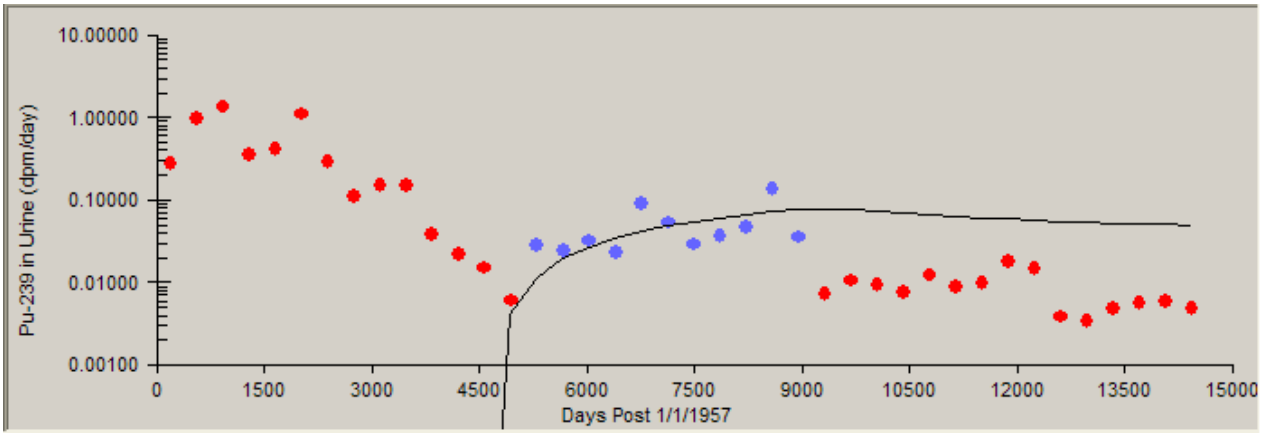

Figure B-45. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1970 through December 1981.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

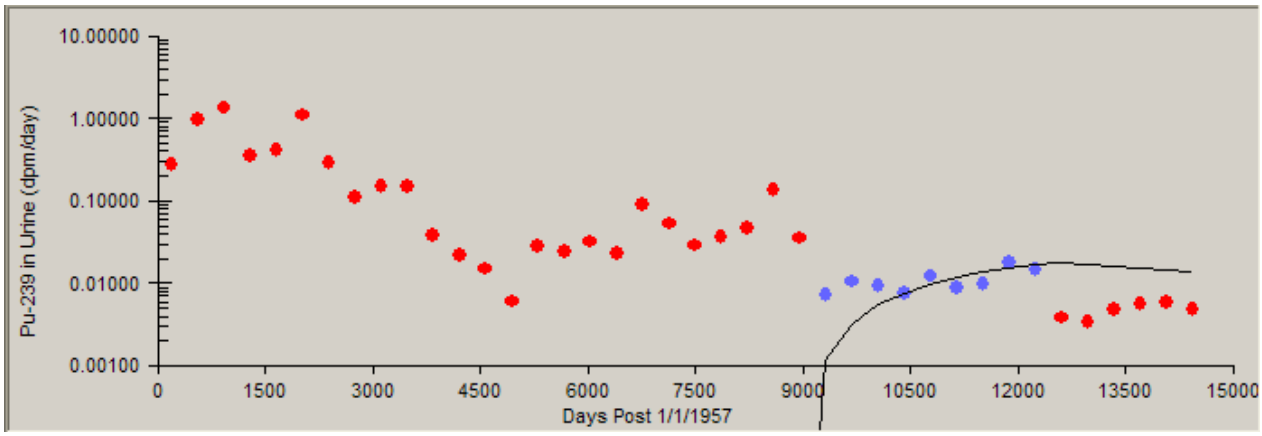

Figure B-46. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1982 through December 1990.

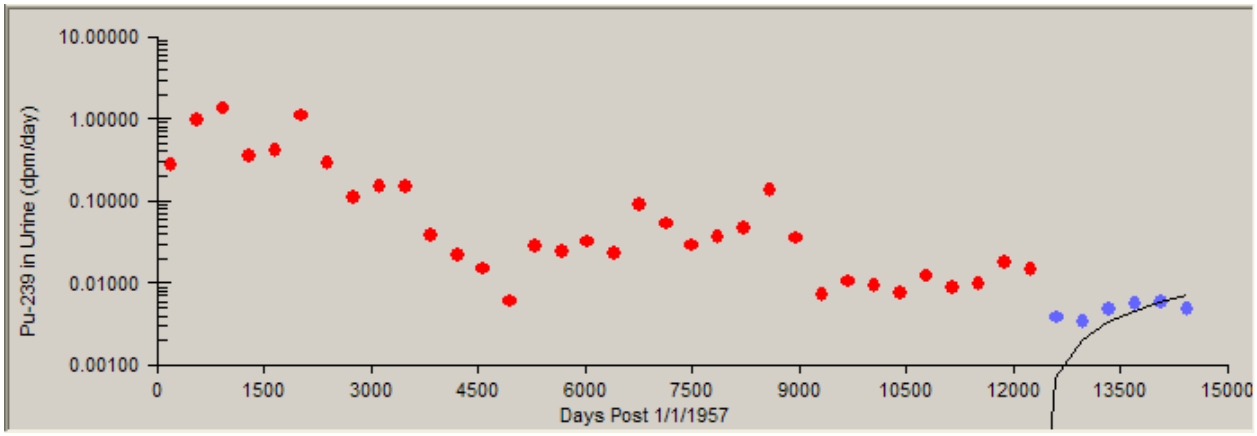

Figure B-47. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type S plutonium from January 1991 through December 1996.

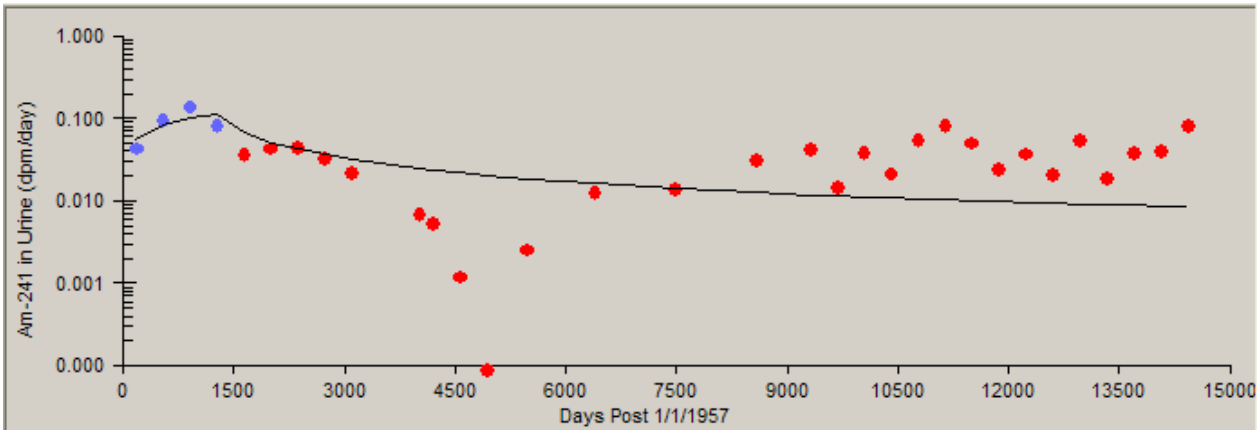

Figure B-48. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M ^{241}Am from January 1957 through December 1960.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

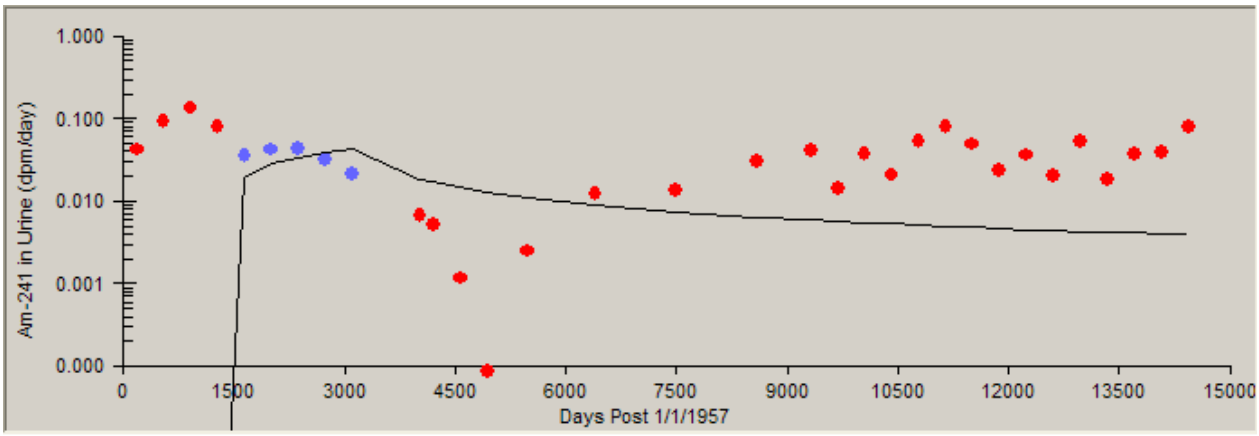

Figure B-49. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M ^{241}Am from January 1961 through December 1965.

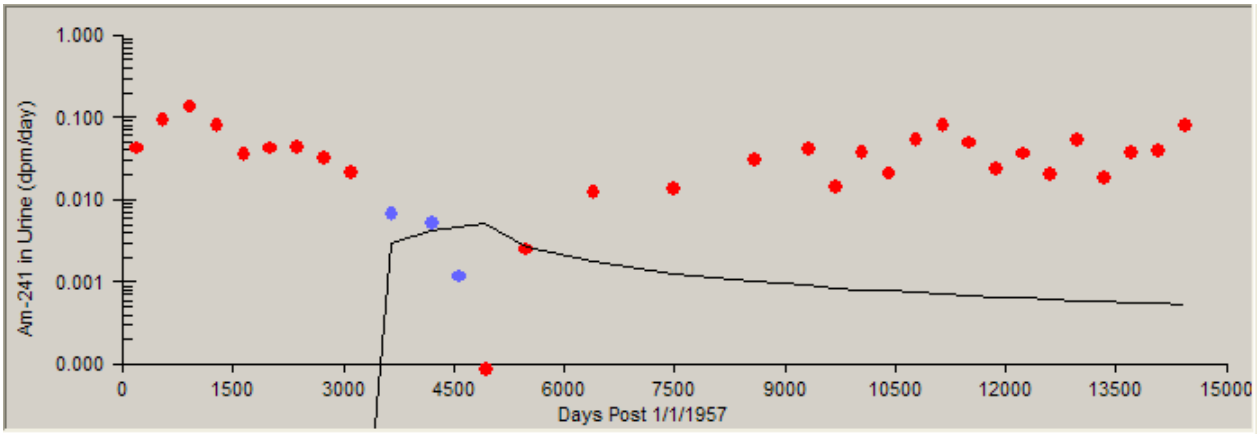

Figure B-50. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M ^{241}Am from January 1966 through December 1970.

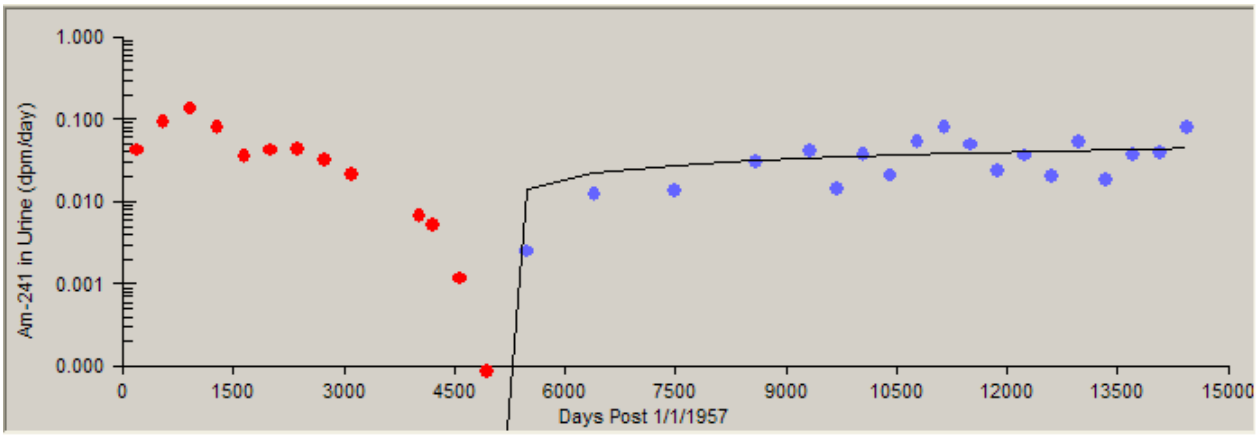

Figure B-51. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type M ^{241}Am from January 1973 through December 1979.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

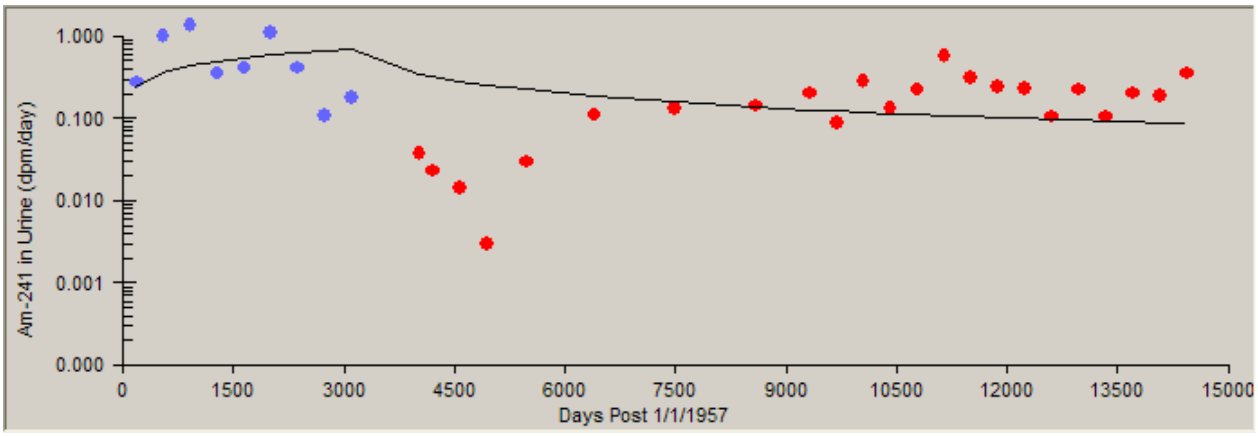

Figure B-52. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M ^{241}Am from January 1957 through December 1965.

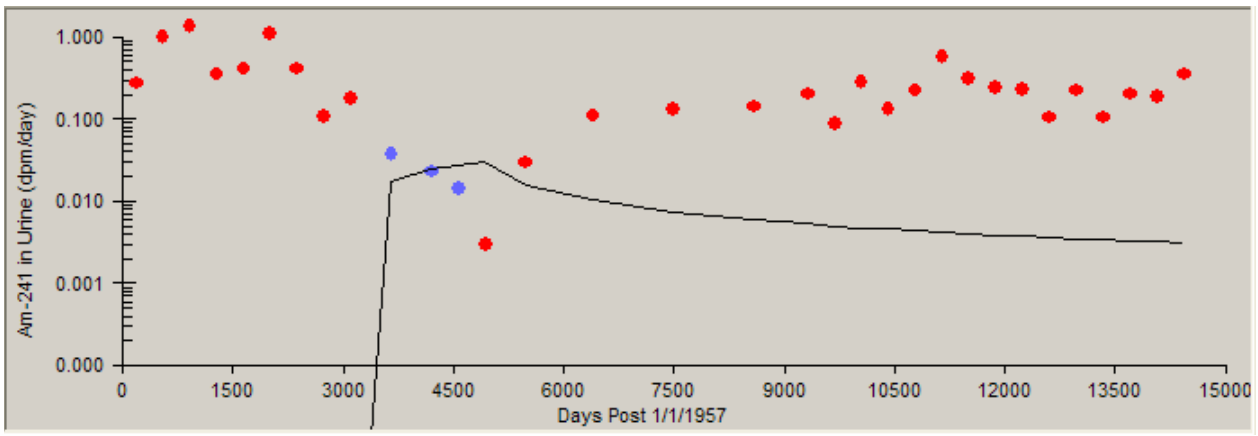

Figure B-53. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M ^{241}Am from January 1966 through December 1970.

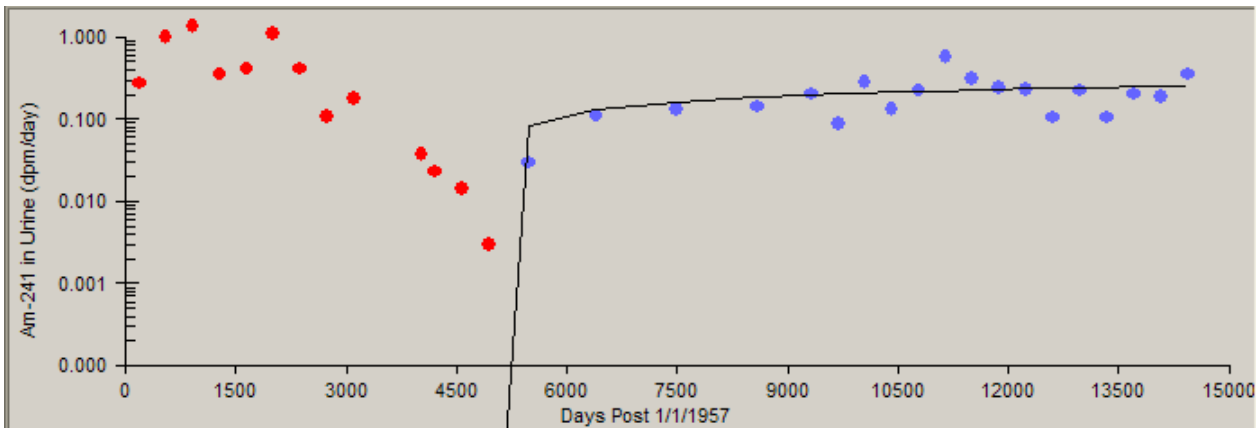

Figure B-54. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type M ^{241}Am from January 1971 through December 1996.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

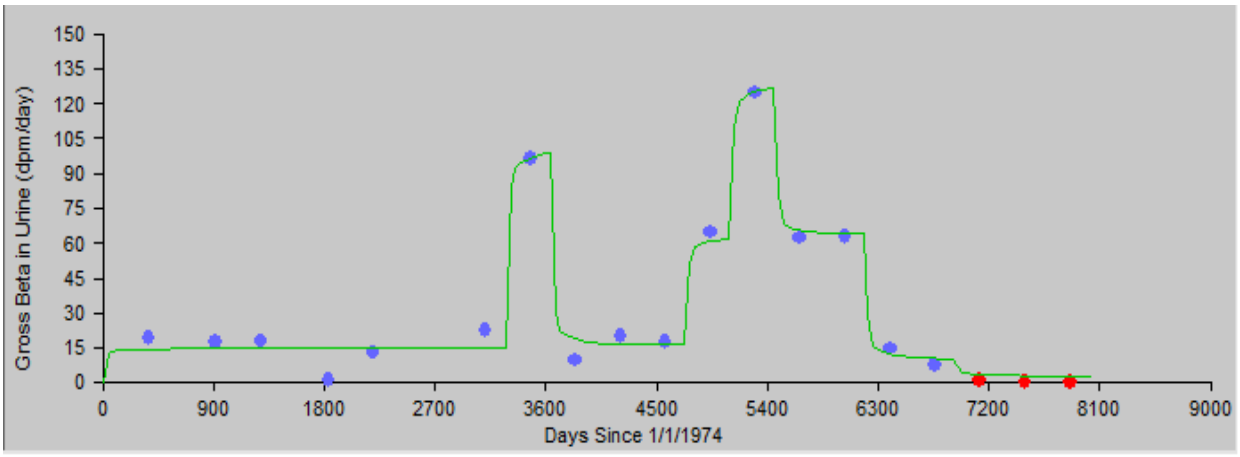

Figure B-55. Predicted and observed 50th-percentile gross beta urinary excretion assuming a chronic inhalation intake of type F ^{90}Sr from January 1974 through December 1992.

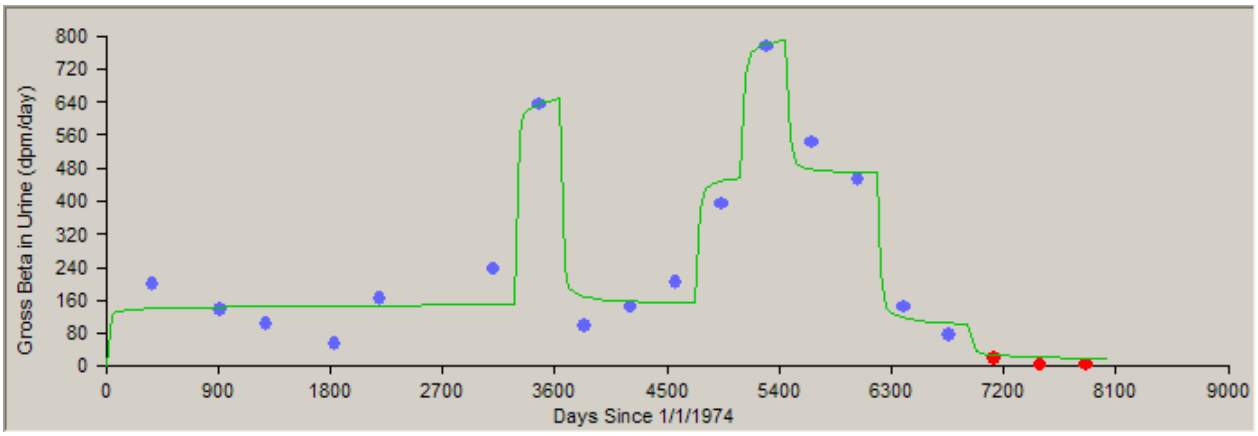

Figure B-56. Predicted and observed 84th-percentile gross beta urinary excretion assuming a chronic inhalation intake of type F ^{90}Sr from January 1974 through December 1992.

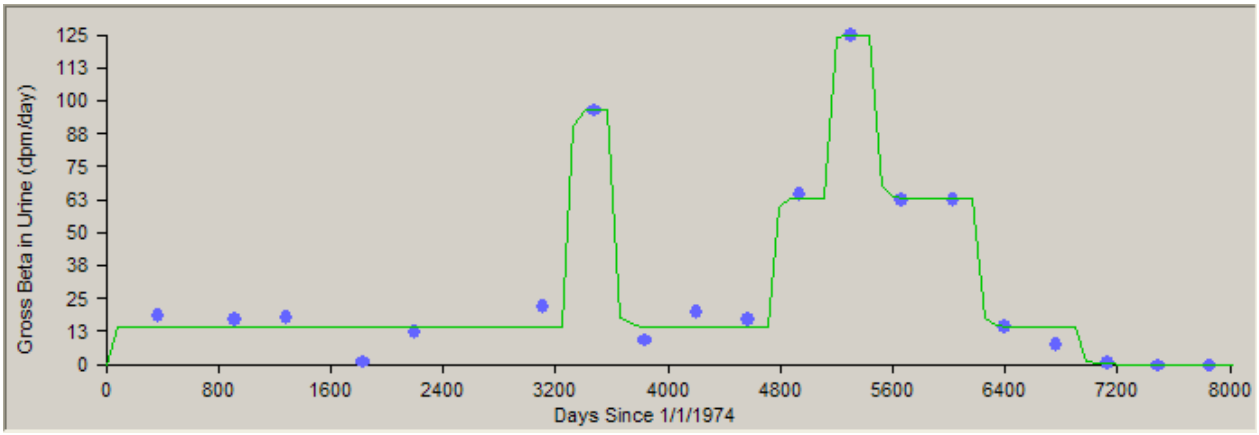

Figure B-57. Predicted and observed 50th-percentile gross beta urinary excretion assuming a chronic inhalation intake of type F ^{103}Ru from January 1974 through December 1992.

ATTACHMENT B
INTERNAL DOSIMETRY COWORKER DATA (continued)

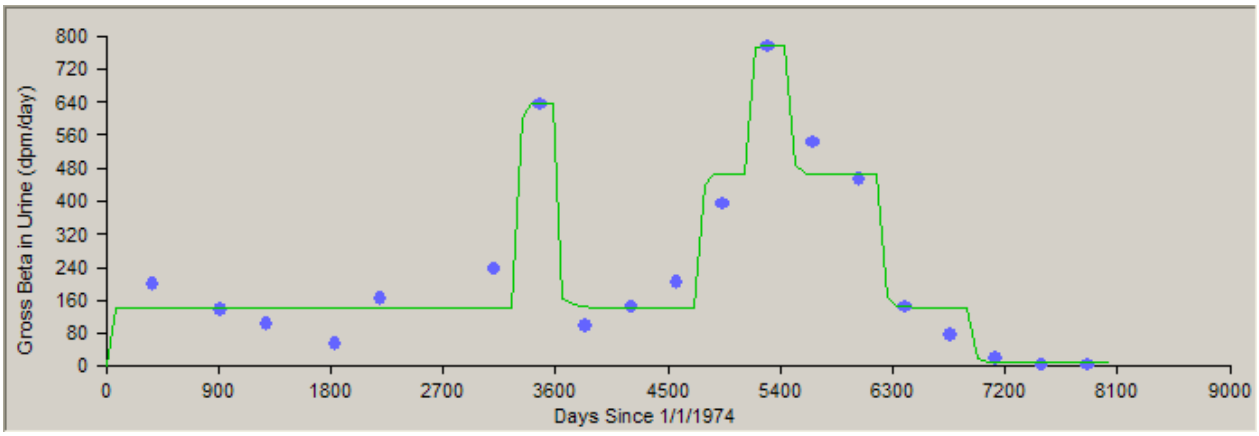

Figure B-58. Predicted and observed 84th-percentile gross beta urinary excretion assuming a chronic inhalation intake of type F ^{103}Ru from January 1974 through December 1992.

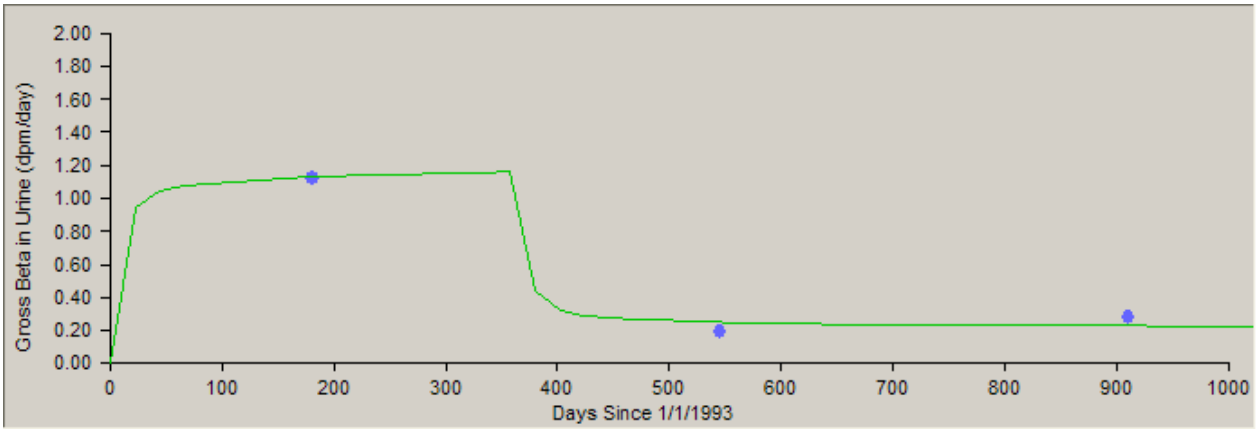

Figure B-59. Predicted and observed 50th-percentile urinary excretion assuming a chronic inhalation intake of type F ^{90}Sr from January 1993 through December 1995.

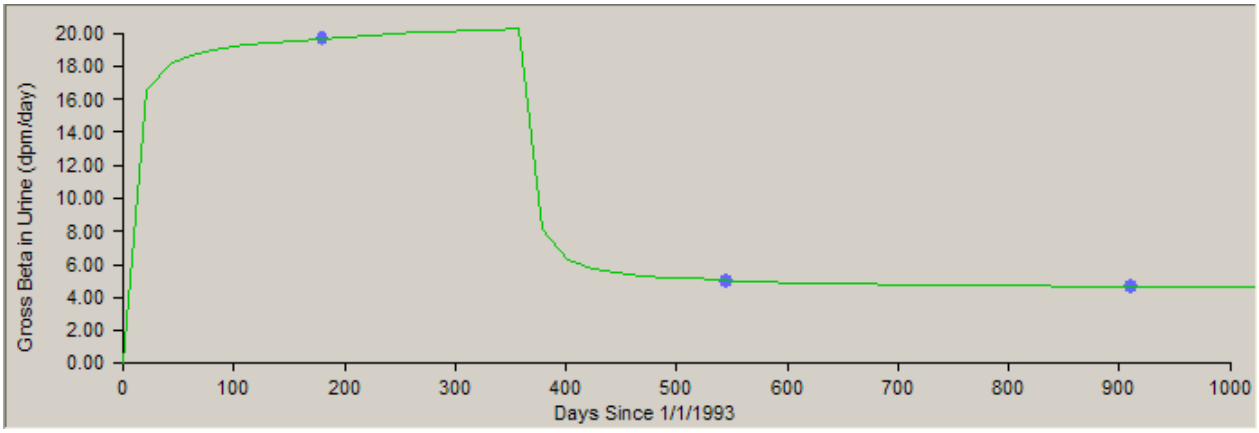

Figure B-60. Predicted and observed 84th-percentile urinary excretion assuming a chronic inhalation intake of type F ^{90}Sr from January 1993 through December 1995.

ATTACHMENT C
SCALING OF COWORKER DOSE INTAKES CONSISTENT WITH TYPE AND DURATION OF EXPOSURES

Assigned Intakes Using Case-Specific Information

As indicated previously in Section 5.3, workers were selected for internal monitoring based on direction by the program supervisor and health physics staff based on the potential of internal exposure. Therefore, the assignment of coworker intakes to personnel without internal monitoring data should be uncommon. However, for those cases where coworker intakes need to be assigned, the following guidance should be used.

The major presumptive exposures at LLNL were made up of the primary radionuclides plutonium, uranium, and tritium. Other radionuclides might have resulted in intakes to personnel but, due to the small amounts of material, the limited operations, and the use of engineering controls such as hot cells, fume hoods, gloveboxes, and dry boxes, potential exposures are relatively unlikely. When exposures were possible, the intakes were likely to be small and highly dependent on work category. Work locations are another important consideration when evaluating for a potential unmonitored intake. Although it is not possible in all cases to precisely locate an LLNL employee's work areas, it is certainly true that the nonprimary radionuclides presented a limited hazard to most LLNL personnel. In this attachment, a method is presented to scale assigned intakes based on job category and likely duration of exposure to limit the margin of overestimation.

Level of contact

For the purpose of dose reconstruction, the amount of exposure should be considered when assigning coworker dose intakes. For personnel with full contact with the material, the 95th percentile of the calculated coworker intakes should be assigned. For others, the 50th percentile is more appropriate, as listed in Table C-1.

ATTACHMENT C
SCALING OF COWORKER DOSE INTAKES CONSISTENT WITH TYPE AND DURATION OF EXPOSURES (continued)

Table C-1. Levels of contact for certain job categories.

Contact level	Description	Example job titles	Percentile of CW intake
Primary contact	Contact with material in unencapsulated and uncontained form. Normal contact with the material is under less controlled environments such as in fume hoods and inside chemical apparatus.	Chemists, researchers, metallurgists	95th
Secondary contact	Normal contact with the material is in containment or process equipment, occasional contact with airborne material during excursions.	Operators, janitors, decommissioning and decontamination workers, health physics personnel routinely assigned	50th
Incidental contact	Contact with contamination and airborne radioactive material only during an excursion or incident or as part of an exposure during a maintenance, repair, waste disposal, decontamination, or decommissioning activity.	Janitors, decommissioning and decontamination workers, maintenance, health physics personnel not routinely assigned construction activities associated with waste disposal or demolition of legacy structures	50th
Contact unlikely, but presumed	Nonradiological job categories, construction activities not associated with waste or decontamination and decommissioning, but evidence supports a potential incidental exposure.	Construction activities	50th
Contact unlikely, not presumed	Nonradiological job categories, and no evidence to suggest a presumptive exposure.	Administrative personnel, clerk, secretary, telephone operator, draftsman, computer specialist[a]	Assume environmental internal dose

a. These are just a few example job titles where internal exposure are considered unlikely. Refer to ORAUT (2004) for other examples of job categories that might have little or no potential for workplace internal exposures.