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Rocky Flats Plant – Occupational Internal Dose

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☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
01/12/2004	00	New document to establish technical basis document for the Rocky Flats Plant - occupational internal dose. First approved issue. Initiated by Robert Meyer.
12/13/2005	00 PC-1	Approved page change revision initiated to incorporate recent direction from NIOSH to include DOL review comments on page 7 in Section 5.1. Change made to Table 5.3.1.4.2-2 on page 17 in Section 5.3. No sections were deleted. Retraining is not required. Initiated by Robert Meyer.
02/01/2007	01	Approved Revision 01 revised to reflect current introductory material. Revised in response to Union comments. Change made to Table 5-9. Constitutes a total rewrite of document. This revision addresses Worker Outreach comments as described in CT-0201, CT-0206, and CT-0207. Revised to incorporate attribution, per ORAU request. The Worker Outreach comments from the June 23, 2004, meeting of the United Steelworkers of America Local 8031 and Rocky Flats Security Officers Local Union 1 are addressed in Section 5.3.2 and 5.4.2 regarding internal organ counts; Sections 5.2.1.2, 5.2.2.2, 5.2.3.2., and 5.2.4.2 regarding particle size and that as many reports as possible have been reviewed in the writing of this section. The Worker Outreach comments from the June 23, 2004, meeting of Colorado State Building and Construction Trades regarding whole-body counting are addressed in Sections 5.3.2 and 5.2.4. Incorporates internal and NIOSH formal review comments. This revision results no change to the assigned dose and no PER is required. Training required: As determined by Task Manager. Initiated by Robert Meyer.
08/17/2007	02	Approved Revision 02 initiated to capture Advisory Board comments associated with the June 2007 Advisory Board meeting. Incorporation of thorium and internal coworker from ORAUT-OTIB-0038 and OCAS-TIB-014. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Task Manager. Initiated by Mutty M. Sharfi.
09/30/2014	03	Revision initiated to incorporate Advisory Board comments, the approval of SEC-00192, and new dose reconstruction approaches in assessing tritium, ^{233}U, and recycled uranium. Incorporates additional clarifications on current dose reconstruction guidance. 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.
09/01/2020	04	Revision initiated to capture Advisory Board comments and items associated with SEC-00192. Includes update to the co-exposure study in Attachment D to capture the data from OCAS-TIB-0014. Added the potential and intake rates for type SS plutonium. 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.

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ACRONYMS AND ABBREVIATIONS

ABRWH	Advisory Board on Radiation and Worker Health
AEB	Alvin E. Blackwell Bioassay Laboratory
AEC	U.S. Atomic Energy Commission
AMAD	activity median aerodynamic diameter
AWE	atomic weapons employer
Bq	becquerel
CAM	continuous air monitor
CEDR	Comprehensive Epidemiology Data Resource
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
CML	Critical Mass Laboratory
cpm	counts per minute
CWT	chest wall thickness
d	day
DCF	dose conversion factors
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
DTPA	diethylene triamine pentaacetic acid
DU	depleted uranium
EDTA	ethylene diamine tetraacetic acid
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EMCBC	Environmental Management Consolidated Business Center
EPA	U.S. Environmental Protection Agency
ERDA	U.S. Energy Research and Development Administration
EU	enriched uranium
F	fast (absorption type)
FBI	Federal Bureau of Investigation
FOIA	Freedom of Information Act
FY	fiscal year
g	gram
GSD	geometric standard deviation
HEU	highly enriched uranium
hr	hour
HSDS	Health Sciences Data System
HT	elemental tritium (tritiated gas)
HTO	tritiated water vapor
ICRP	International Commission on Radiological Protection
IDOT	Internal Dosimetry Tool
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
INL	Idaho National Laboratory

IREP	Interactive RadioEpidemiological Program
keV	kilovolts-electron, 1,000 electron volts
kg	kilogram
L	liter
L X-ray	low-energy X-ray
LANL	Los Alamos National Laboratory
LLNL	Lawrence Livermore National Laboratory
M	medium (absorption type)
MDA	minimum detectable amount
MDL	minimum detection level
MFAP	mixed fission and activation product
MgTh	magnesium-thorium (alloy)
min	minute
mL	milliliter
MLT	minutes live time
mm	millimeter
MSS	multispectral scanner survey
MPLB	maximum permissible lung burden
mR	millirad
mrem	millirem
mW	milliwatt
MW	megawatt
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
NMMSS	Nuclear Materials Management and Safeguards System
NOCTS	NIOSH-Division of Compensation Analysis and Support Claims Tracking System
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
Oy	bare HEU metal
pCi	picocurie
PER	Program Evaluation Report
PGT	Princeton Gamma Tech
PHA	pulse height analysis
ppm	parts per million
PuNp	plutonium-neptunium (alloy)
QA	quality assurance
QC	quality control
RCG	radioactivity concentration guide
RCRA	Resource Conservation and Recovery Act of 1976
RFAO	Rocky Flats Area Office
RFP	Rocky Flats Plant
ROI	region of interest
RWMC	Radioactive Waste Management Complex
s	second

S	slow (absorption type)
SAAM	selective alpha air monitor
SEC	Special Exposure Cohort
SNL	Sandia National Laboratories
SRDB Ref ID	Site Research Database Reference Identification (number)
SRS	Savannah River Site
SS	super S (absorption type)
SSR	safe, secure railcar
SST	safe, secure transport
Sv	sievert
TBD	technical basis document
TBP	tributyl phosphate
TLD	thermoluminescent dosimeter
TOPO	trioctylphosphene oxide
TSA	Technical Safety Appraisal
TTA	thenoyltrifluoroacetone
U.S.C.	United States Code
UNH	uranyl nitrate
W	watt
WG	weapons grade
wk	week
WISC	Waste Stream Identification and Characterization
yr	year
ZPPR	Zero Power Plutonium (later Physics) Reactor
µCi	microcurie
µg	microgram
µm	micrometer
γ	gamma particle
§	section or sections

5.1 INTRODUCTION

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide 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 Rocky Flats Plant (RFP) internal dosimetry data for dose reconstruction and includes guidance for the appropriate use of that information.

5.1.2 Scope

Workers at RFP had the potential to receive intakes of plutonium, americium, enriched uranium (EU), depleted uranium (DU), and tritium, as well as miscellaneous other radionuclides (Daugherty et al. 2001). Section 5.2 describes the available source term information including isotopic composition, solubility, and particle size. Site-specific internal dosimetry information for other radionuclides such as thorium, curium, and neptunium is rare or not available.

Sections 5.3 and 5.4 discuss these two datasets in detail including the history, sensitivity, and pertinent nuances of the methods and data.

The internal exposure record for a worker consists of records of the bioassay data and reports of involvement in incidents, accidents, or special situations. Section 5.6 describes samples of these records and reports with explanations of the aspects important to dosimetry.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 5.7.

Attachments A and B provide detailed information about minimum detectable amounts (MDAs) for urinalysis and in vivo lung counts, respectively. Attachment C provides examples of bioassay data records and reports, and Attachment D details internal co-exposure dosimetry data. Attachment E evaluates the potential for internal dose from ^{237}Np. Attachment F addresses worker statements about the presence of MgTh alloy at RFP, and Attachment G examines tritium issues. Attachment H provides additional information on the Critical Mass Laboratory (CML), and Attachment I provides an analysis of the potential effects on dose reconstruction due to a Federal Bureau of Investigation (FBI) raid and investigation of RFP activities in 1989 (see Section 5.1.4).

5.1.3 Special Exposure Cohort

The Secretary of the U.S. Department of Health and Human Services has designated the following class of employees from RFP as an addition to the SEC (Sebelius 2013, p. 3):

> *All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Rocky Flats Plant in Golden, Colorado, from April 1, 1952, through December 31, 1983, 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 included in the Special Exposure Cohort.*

NIOSH has determined that doses to unmonitored RFP workers from neptunium, thorium, and ^{233}U (and its associated ^{232}U and ^{228}Th progeny) cannot be reconstructed from April 1, 1952, through December 31, 1983, inclusive (NIOSH 2013a).

The class includes all workers during the SEC period. Because of the identified dose reconstruction infeasibility, all dose reconstructions for monitored workers during the 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 that data. However, such dose reconstructions are still considered

partial dose reconstructions because of the determination that exposure to neptunium, thorium, and ^{233}U (and its associated ^{232}U and ^{228}Th progeny) during the SEC period cannot be bounded.

5.1.4 Federal Bureau of Investigation Raid in 1989

On October 24 and 25, 1989, a former RFP worker made allegations in a set of interviews conducted by the U.S. Environmental Protection Agency (EPA) National Enforcement Investigation Center's Office of Criminal Investigations and the FBI. These interviews resulted from a phone call by the interviewee to the FBI Rocky Flats Hotline on June 16, 1989, alleging safety violations and manipulation of laboratory samples at RFP.

Attachment I provides a detailed assessment of these allegations. The following is a summary of the conclusions of this assessment:

- No scientific basis for concluding that the issues raised about environmental samples would compromise radiological count results, nor does the reviewed information corroborate a link between the environmental and occupational radiological programs.

- There were no situations identified where falsification or invalidation of data would impact the ability to perform dose reconstruction under EEOICPA.

- The charges against Rockwell were specific to environmental Resource Conservation and Recovery Act of 1976 (RCRA) and Clean Water Act laws and the impact to the environment; the charges did not specifically call out a data falsification, data validity issues, or a data quality violation.

- One individual provided information about involvement in shredding documents. While the documents being destroyed could have been some kind of field surveys, there is no indication that those surveys have an impact on the ability to bound or reconstruct dose for the class, as long as the personnel monitoring data exist. These records do exist in the associated personnel files in the NIOSH-Division of Compensation Analysis and Support Claims Tracking System (NOCTS); thus, those files were not destroyed.

- An issue of "penciling-in" information on radiological field survey records was raised. The primary source of radiological information for individual dose reconstruction is the individual thermoluminescent dosimeter (TLD) results and bioassay information. These are performed in a laboratory and not documented in the field.

- Concerns about bioassay sample analysis results (false positives and [statistical] variations); bioassay sample handling and processing; personnel contamination and contamination incidents; and issues about tritium bubblers, neptunium, MgTh alloy, and the CML. The dose reconstruction process accounts for the potential for missed doses and incorporates methods that are favorable to the claimant. All of the issues concerning tritium bubblers, neptunium, MgTh Alloy, and the CML are addressed in other sections of this TBD.

- A contention was made that there was an additional August 1989 aerial multispectral scanner survey (MSS) performed at RFP in addition to the one performed in June and July of 1989, and that the flyover data indicate the presence of the isotopes ^{137}Cs and ^{90}Sr, which is used to imply that an unreported criticality occurred. Neither the August 1989 flyover survey nor evidence supporting a criticality event could be located. Based on interviews, document reviews, and files provided, no evidence or information was identified that disputes the ability to bound RFP worker dose under EEOICPA.

- A review of the RFP Technical Safety Appraisal (TSA) concluded that there was no identified impact on radiological personnel monitoring data that are used to support bounding/reconstructing dose for the RFP worker class.

- A review of an allegation of record falsification involving mislabeling waste for shipment off site concluded that none of this information impacts the ability to reconstruct radiation dose with sufficient accuracy under the EEOICPA Program for the RFP worker class.

- A review of excerpts from the notebook/logbook of an RFP Manager who made notes on the radiological program did not support a data falsification issue that would impact the ability to reconstruct dose for the RFP worker class.

- A review of the quantity of available personnel radiological monitoring data available at the time of this assessment. It was concluded that there exists a sufficient quantity of individual monitoring data to support the assessment of RFP personnel doses.

Based on this information, the conclusion was reached that there was no impact to the ability to perform individual dose reconstructions with sufficient accuracy under EEOICPA due to the FBI raid or issues about data falsification or invalidation.

5.2 SOURCE TERM

In 1993, the Secretary of Energy formally announced the end of nuclear production at RFP. Remediation was completed at the RFP in late 2005. Co-exposure intakes should be assigned, when applicable, up through 2005. Only environmental intakes should be assigned after 2005.

5.2.1 Plutonium

5.2.1.1 Isotopic Composition

Three aspects of the isotopic composition of plutonium are important to internal dose reconstruction:

- The percent by weight of ^{241}Pu, which is needed to calculate the ingrowth of ^{241}Am for the lung count data;

- The fraction of the activity for each alpha-emitting plutonium isotope, which is needed to account for the dose from unmeasured isotopes; and

- The ratio of the activity of ^{241}Pu to the alpha activity of the other plutonium isotopes, which is needed to calculate the intake of ^{241}Pu from intakes from bioassay data for ^{239}Pu and ^{240}Pu.

Table 5-1 lists the weight percent and fraction of alpha activity for each isotope of weapons-grade (WG) plutonium that was present at RFP throughout most of its 1952-to-1989 production history.

The Zero Power Plutonium (later Physics) Reactor (ZPPR) special project in the mid-1960s involved reactor-grade plutonium. The ratio of the activity of ^{241}Pu to the alpha activity of the other plutonium isotopes is 32.

Table 5-2 lists the weight percent and alpha activity fraction for each isotope. Reports of accidents or incidents that involved ZPPR plutonium generally note “ZPPR” or “ZPPR material,” especially on the lung count reports (RFP 1976–1996, pp. 99, 225).

Table 5-1. Weight percent and fraction of alpha activity for WG plutonium.[a]

Isotope	Weight percent	Fraction of alpha activity[b]
Pu-238	0.01	0.023
Pu-239	93.79	0.8
Pu-240	5.8	0.18
Pu-241	0.36[b]	Not applicable
Pu-242	0.03	Negligible

a. Source: *Final Environmental Impact Statement, Rocky Flats Plant Site* (DOE 1980, Volume 1, Table 2.7.2-2, p. 236). Values are the average for RFP plutonium from July 1976 to July 1, 1978. This isotopic composition is also typical of plutonium metal processed at RFP to 1990 (James 1990).
b. The percent by weight of ^{241}Pu for 1959 to 1977 was 0.49, with a range of 0.35 to 0.65 (KHC 2002, p. 120).

Table 5-2. Weight percent and fraction of alpha activity for ZPPR plutonium.[a]

Isotope	Weight percent	Fraction of alpha activity
Pu-239	87.6	0.7
Pu-240	10.0	0.3
Pu-241	2.4	Not applicable

a. These ZPPR values are based on extracted data in a working file from an undocumented source.

Dose reconstructions should account for the activity of ^{241}Am in the plutonium mixture. The concentration of the ^{241}Am is variable depending on the time since the plutonium was purified and whether the mixture involved waste or byproduct (separated ^{241}Am) from the purification of aged plutonium. Starting in 1969, parts per million of ^{241}Am (ppm ^{241}Am) were measured for the plutonium mixture in significant possible inhalation incidents and were generally recorded on lung count reports for involved workers. A nominal amount, 100 or 1,000 ppm by mass, of ^{241}Am should be assumed if no other data are available. Note that the practice at RFP was to measure the ppm ^{241}Am in a representative sample of material that was involved in a possible inhalation incident. If a representative sample was not obtained or the origin of the intake was not known, a default value of 1,000 ppm ^{241}Am was used and was assigned to the date of the intake or to the date of the first positive lung count if the date of the intake was not known. The fact that RFP arbitrarily assumed 1,000 ppm should not be the basis for determining the plutonium mixture.

If the plutonium intake for WG plutonium is assessed for ^{239,240}Pu, the activity of ^{241}Am in the intake mixture is calculated by:

$$^{241}\text{Am activity} = {}^{239,240}\text{Pu activity} \times [48.2 \times \text{ppm } {}^{241}\text{Am} \div (1 \times 10^6 - \text{ppm } {}^{241}\text{Am})].$$

For ZPPR plutonium:

$$^{241}\text{Am activity} = {}^{239,240}\text{Pu activity} \times [44.6 \times \text{ppm } {}^{241}\text{Am} \div (1 \times 10^6 - \text{ppm } {}^{241}\text{Am})].$$

The multiplier for WG plutonium is the inverse of Equation B-17 in Attachment B. This multiplier is modified to apply to ZPPR plutonium based on the ratio of the weighted specific activities of the ^{239}Pu and ^{240}Pu for WG and ZPPR plutonium, 0.071 and 0.0767, respectively. The ratio of 0.926 times 48.2 results in the value of 44.6 in the ZPPR multiplying factor.

In December 1989, DOE suspended plutonium processing. Therefore, RFP would not have received any new plutonium shipments after this date. However, the americium in the residual plutonium

mixture would have continued to age over time. To account for this aged material, after 1989, only the 6% 10-year-aged plutonium mixture ratio should be considered, based on guidance in the ORAUT-TKBS-0006-5, *Hanford Site – Occupational Internal Dose* (ORAUT 2015d). The Hanford data is being used as a surrogate because it was the main source of RFP's plutonium material. This results in a $^{239+240}$Pu:^{238}Pu:^{241}Pu:^{241}Am of 1.00:0.111:7.150:0.146 and a ^{241}Pu:^{241}Am activity ratio of 49.12:1 (4,912%).

5.2.1.2 Plutonium Solubility and Particle Size

The plutonium fire on October 15, 1965, in Buildings 776 and 777, is a special case. The plutonium, which was strongly retained in the lungs of exposed workers with relatively low transfer to the urine, exhibited highly insoluble (type SS) characteristics (ORAUT 2020).

Plutonium in chemical processing operations can be either soluble (type M), insoluble (type S), highly insoluble (type SS), or a mixture of solubilities. Dose reconstructors should select the material type that is most favorable to the claimant (ORAUT 2018). Lung count data in conjunction with urine data can help to determine absorption type.

In general, particle size and distributions are not available for work areas or incidents at RFP. Therefore, dose reconstructions should use the default value of 5-μm activity median aerodynamic diameter (AMAD) (NIOSH 2002).

One exception is the plutonium fire on October 15, 1965, in Buildings 776 and 777 (Dow 1965), for which Mann and Kirchner (1967) measured a mass median diameter of 0.3 μm (1-μm AMAD) with a geometric deviation of 1.83. Therefore, for individuals potentially involved with a plutonium fire, the more favorable particle size of 1-μm or 5-μm AMAD should be assumed.

The 1-μm particle adjustment for RFP plutonium fires should only be applied for energy employees who were involved with a known intake from a plutonium fire (or any time dose reconstructors deem use of a 1-μm AMAD particle size appropriate) (NIOSH 2002). This can be from involvement with the plutonium fire itself, including being in the building or area and exposed to smoke or airborne activity from the fire as well as involvement in cleanup activities immediately after the fire.

The application of the 1-μm particle size adjustment only applies to individuals who were involved in a fire (i.e., operators and firefighters) and the individuals who performed the immediate cleanup of the incident. Once that is accomplished, it is assumed that the particle size reverts back to the default 5-μm AMAD. When applicable, the adjustment factor is applied only to the dose associated with the intake that is directly from the fire and cleanup. The 1-μm particle size adjustment typically applies for a short period (i.e., days, weeks, or a few months).

The use of the 1-μm particle adjustment for RFP plutonium fires is specific to the intake being assessed. If an earlier or later intake is assessed that is not associated with a plutonium fire, the 1-μm particle adjustment factor does not apply.

5.2.2 Americium

5.2.2.1 Isotopic Composition

For the NIOSH Dose Reconstruction Project, the measured americium is ^{241}Am. The source of the americium is only from the decay of ^{241}Pu. No other americium isotopes are involved.

5.2.2.2 Americium Solubility and Particle Size

Americium was present in two forms at RFP, as a purified byproduct of plutonium recovery and as atoms that are formed by the nuclear transformation of ^{241}Pu and embedded in the matrix of the plutonium particle. As a purified byproduct, International Commission on Radiological Protection (ICRP) Publication 68 specifies americium inhalation absorption as type M (ICRP 1995, p. 85). For embedded atoms in the matrix of an inhaled plutonium particle, dose reconstructors should use the solubility classification for the plutonium particle in Section 5.2.1.2 (ICRP 1994).

Dose reconstructors should use the default 5-µm AMAD particle size (NIOSH 2002) except for fire incidents, in which a 1-µm AMAD should be assumed for consistency with Section 5.2.1.2 above.

5.2.3 Uranium

5.2.3.1 Enriched Uranium

5.2.3.1.1 Isotopic Composition

Production at RFP involved EU from 1952 to 1963. Table 5-3 lists the weight percent and fraction of alpha activity for each isotope.

Table 5-3. Weight percent and fraction of alpha activity for EU.[a]

Isotope	Weight percent	Fraction of alpha activity
U-234	1	0.97
U-235	93	0.031
U-236	0.39	0.0039
U-238	5.4	0.00028

a. Source: DOE (1980, Volume 1, Table 2.7.2-4, p. 238).

5.2.3.1.2 Enriched Uranium Solubility and Particle Size

Operations for EU paralleled those for plutonium and included chemical processing and metalworking. Compounds of uranium are generally more soluble than those of plutonium, and solubility classification is uncertain. The ICRP assigns $UO_2(NO_3)_2$ (uranyl nitrate or UNH) to inhalation type F; UO_3 (yellow cake), UF_4, and UCl_4 to inhalation type M; and UO_2 and U_3O_8 to inhalation type S (ICRP 1979, 1994, 1995). All of these compounds were involved in the recovery and recycling processes for EU in Building 881 (KHC 2000a).

In many cases, the compound of uranium in an intake was not identified. Dose reconstructors should use the solubility classification that is most favorable to claimants.

If site-specific data for particle size of uranium are not available, dose reconstructors should use the default particle size value of 5-µm AMAD (NIOSH 2002).

5.2.3.2 Depleted Uranium

5.2.3.2.1 Isotopic Composition

DU was present at RFP throughout its production history. Uranium-238 accounts for the majority of DU internal dose, but the total uranium alpha activity should be included in the dose reconstruction (Table 5-4).

Table 5-4. Weight percent and fraction of alpha activity for DU.[a]

Isotope	Weight percent	Fraction of alpha activity
U-234	0.00058	0.097
U-235	0.23	0.013
U-238	99.77	0.89

a. These values are derived from data in DOE (1980, Volume 1, Table 2.7.2-4, p. 238).

5.2.3.2.2 Depleted Uranium Solubility and Particle Size

Operations with DU involved metalworking including casting, forming, and melting with what was probably UO_3 and U_3O_8 (KHC 2000a). The solubility classification is ambiguous, falling somewhere between type S and type M (KHC 1998a, Section 6.1; HPS 1995; Lawrence 1984). Dose reconstructors should use the solubility classification that is most favorable to claimants.

If site-specific data for particle size of uranium are not available, dose reconstructors should use the default particle size value of 5-µm AMAD (NIOSH 2002).

5.2.3.3 Uranium-233

Operations with ^{233}U (thorium strikes) occurred between 1964 and 1983 (Moment, Gibbs, and Freiboth 1999). The process included the following steps:

1. Material received as nitrate solution,
2. Thorium strike (thorium fluoride precipitation > peroxide precipitation > UO_4 cake),
3. Conversion (UO_4 > UO_3 > UO_2 > UF_4),
4. Reduction to metal and casting into an ingot,
5. Rolling ingot into a sheet and producing part blanks from the sheet,
6. Machining, and
7. Sampling.

In the beginning of operations, the first two steps were performed in Building 71 (later called 771). Intermediate steps (conversion to UF_4, reduction to metal, and casting) were performed in Building 81 (later called 881). The ingot was rolled and formed into parts in Building 83 (later called 883) and then transferred back to Building 81 for final machining. Finished parts were sent to Building 77 (later called 777) where they were assembled and shipped. By the mid-1970s, the intermediate steps in Building 881 shifted to the research and development areas of Building 771.

Because of data issues and limitations, no specific methods to bound doses from ^{233}U and ^{232}U have been determined. Therefore, doses to unmonitored RFP workers from neptunium, thorium, and ^{233}U (and its associated ^{232}U and ^{228}Th contaminants) cannot be reconstructed.

5.2.3.4 Recycled Uranium

For all DOE uranium after 1952, this analysis assumed the possibility that uranium from refineries was recycled uranium or contained recycled uranium. Table 5-5 provides the activity fractions that should be applied to all uranium intakes after 1952 (NIOSH 2011).

Table 5-5. Activity fraction of contaminant in recycled uranium.

Recycled uranium contaminant	Pu-239	Np-237	Tc-99	Th-232	Th-228
Activity fraction of contaminant in uranium	0.00246	0.00182	0.379	2.73E-06	2.73E-06

Note: If plutonium intakes are assigned through bioassay or co-exposure, it should not be assigned as part of recycled uranium, as this would result in a double assignment of plutonium. However, the rest of the contaminants should still be assigned.

5.2.4 Thorium

Thorium was present at RFP facilities from the beginning of operations in 1952 at least through 1975; quantities varied from 0 or gram quantities to 238 kg in any particular month at the site (ChemRisk 1992, p. 136; Ulsh et al. 2008; Author unknown 1976a). The site used thorium in various ways including:

- Fabrication of metal parts from natural thorium or thorium alloys,
- Use of oxide ("thoria") as a mold-coating compound,
- In compounds for numerous analytical procedures and research and development programs,
- As a substitute for uranium or plutonium components in various research and development activities and programs, and
- The removal of ^{228}Th (thorium strike) performed during ^{233}U processing.

While the consensus of the contributors and authors of the thorium reference documents was that the quantities and concentrations of thorium on the site over the years at RFP were minimal, there was the potential for thorium exposures to certain populations of workers. The available documentation supports the presence of thorium on site in the early 1950s through the development of internal and external thorium-monitoring processes (Dow 1953–1963; Hammond 1956–1958, 1958).

There is no indication that the MgTh was ever used at the RFP site. Attachment F presents an analysis of available information to support this conclusion.

Because of data issues and limitations, no specific methods to bound doses from thorium have been determined. Therefore, NIOSH has determined that unmonitored thorium doses at RFP cannot be reconstructed.

5.2.5 Neptunium

Neptunium processing at the RFP included preparation of pure neptunium oxide, metal and metal alloys, and the recovery of ^{237}Np from a variety of residues (Conner and Baaso 1981). Processes included dissolution, anion exchange, precipitation, filtration, calcination, conversion to fluoride, and reduction to metal. Fabrication steps such as casting and rolling were also sometimes performed for the production of high-purity metal shapes and foils. Neptunium was recovered from residual materials including sand, slag, crucibles, casting skulls, and various alloys (with plutonium, tin, uranium, and zirconium).

Because of data issues and limitations, no specific methods to bound doses from neptunium have been determined. Therefore, unmonitored neptunium doses at the RFP cannot be reconstructed.

Attachment E provides additional details on neptunium operations.

5.2.6 Critical Mass Laboratory

Building 886 housed the Critical Mass Laboratory (CML) at RFP. Mixed fission and activation products (MFAPs) in both the fuel and containment materials present an internal dose potential for personnel who might ingest or inhale them. CML staff submitted routine bioassay (urinalysis and whole-body counts) to detect intakes of plutonium, uranium, and/or americium, but MFAPs were not routinely monitored.

Attachment H provides a detailed analysis of the CML. Based on this analysis, no significant unmonitored exposure is associated from the generation of fission or activation products.

5.3 IN VITRO

5.3.1 Plutonium Urinalysis

5.3.1.1 Methods, Units, Isotopes, and Interferences

Through 1989, the units of the results are dpm/24-hr excretion period (dpm/24-hr sample). After 1989, the units of the results are dpm/sample regardless of the sample volume or excretion period. Spot urine samples for plutonium were rarely requested and were usually associated with a significant incident, especially an incident with followup chelation using diethylene triamine pentaacetic acid (DTPA). Assume a 24-hour excretion period unless the record indicates that the actual excretion period was different.

Through 1977, samples were counted using an air proportional detector system that did not have sufficient resolution to separate the alpha energies for the plutonium alpha-emitting isotopes. Starting in 1973, an alpha pulse height analysis (PHA) system with surface barrier detectors was phased in and had completely replaced the air proportional detector system by 1978. The plutonium urine results from the air proportional detector system included activity from ^{238}Pu, ^{239}Pu, and ^{240}Pu. Plutonium urine results for samples counted by the PHA system included only ^{239}Pu and ^{240}Pu results. Intake assessments are simpler and more favorable to claimants if dose reconstructors assume ^{239}Pu and ^{240}Pu for all plutonium urine results unless the worker was involved in a special situation involving pure ^{238}Pu. If the intake is assessed using ^{239}Pu and ^{240}Pu data, the ^{238}Pu component of the intake is obtained by multiplying the ^{239}Pu and ^{240}Pu intake by 0.0235. This factor is obtained by dividing the ^{238}Pu fraction of alpha activity stated in Table 5-1 by 0.98, the sum of the fractions of alpha activity for the ^{239}Pu and ^{240}Pu isotopes.

Interferences were probably in the period from 1952 to 1962 because of a lack of specificity of the chemical procedure to isolate only the plutonium in the extract. Plutonium results probably included some americium and thorium activity. In addition, for gross alpha analyses that were assigned to plutonium through 1973, the result could have included some contribution from uranium. However, it is favorable to claimants to disregard such interferences and take the plutonium results at face value unless a value can be determined to be an outlier.

From 1963 to 1977, the ion exchange method significantly reduced interferences from americium, uranium, and thorium. As the PHA system was phased in starting in 1973, the possibility of interferences was further reduced. After 1977, these interferences were not a significant issue for plutonium urine results because all samples were counted on the PHA system. This statement is based on the property of the PHA system to separate and count the alphas by their energies. The alpha energies of the ^{239}Pu and ^{240}Pu isotopes were sufficiently different from the alpha energies of americium and thorium to allow plutonium analyses to be unaffected by the presence of americium or thorium, if any.

DTPA and ethylene diamine tetraacetic acid (EDTA) chelation treatments cause enhanced excretion of plutonium in the urine. Urine data from within 90 days of a chelation injection have historically been excluded from calculations of intakes or depositions of plutonium. Information in the medical or dosimetry records should allow dose reconstructors to discern chelation treatments, which generally followed a significant and documented incident. In the urine data reports for the Health Sciences Data System (HSDS), urine data that was affected by chelation were flagged with a code 1. Code 1 was also used to flag urine data that did not pass quality standards. Dose reconstructors should be wary of any urine result flagged with a code 1 and in general should not use these data in dose reconstruction (Various 1965–1987, p. 28).

5.3.1.2 Plutonium Reporting Levels, Minimum Detectable Amounts, and Uncertainties

The minimum reporting level for plutonium through 1961 was 0.88 dpm/24-hr sample (this was 10% of the RFP tolerance level). For 1962 to April 6, 1970, the minimum reporting level was 0.2 dpm/24-hr sample. Results less than the reporting level were reported as 0.00 dpm/24-hr sample on computer-generated reports, such as the HSDS (see Attachment C, Figures C-3 and C-4) or background (or some abbreviation; e.g., BK) when manually recorded on the Urinalysis Record Card (see Figure C-3). For some workers, results initially reported as background on the Urinalysis Record Card were superseded by the report of the actual result in reports of the HSDS, if the actual result was ≥0.00 dpm/24-hr sample. After April 6, 1970, all results ≥0.00 dpm/24-hr sample were reported. Negative results were reported as 0.00 dpm/24-hr sample through 1989. After 1989, the actual negative value was reported (KHC 2002).

The MDA for plutonium is presented here for the median conditions. By definition of the median value, half of the sample-specific MDAs are lower than the median value, and half are higher. In most cases dose reconstructors are not likely to have sufficient data to determine the sample-specific MDA, so the median values should be used.

Table 5-6 lists the MDA values for plutonium. The values for 1952 to 1977 are based on examination of urinalysis data logs for 1952 to 1971 (see Attachment A). The MDA value for 1971 was extrapolated through 1977. The MDA value for 1978 to 1989 is based on matrix blank data (Author unknown 1992) for the routine plutonium urinalysis program for August 1, 1990, to September 27, 1991, using blank values with a sample-specific recovery in the range of 0.1 to 1.1 dpm/24-hr sample. This range of recoveries mimics the range from 1978 to 1989 for a valid analysis of routine samples. For 1990 to 1992, the blank values with a sample-specific recovery in the range from 0.35 to 1.1 dpm/24-hr sample were used to determine the MDA value. For 1993 to the present, the value of the MDA is equal to the sample-specific MDA of 0.02 dpm/sample that was contractually required in the Rocky Flats Environmental Technology Site bioassay statement of work (KHC 1998b) for any laboratory that processed the sample. Note that the value of the sample-specific MDA is included in the urinalysis data reports starting in 1990.

Some urine samples could have been processed by an offsite commercial laboratory before 1993. The reports for those samples might have the sample-specific MDAs. If these are not available, the MDA in Table 5-6 should be used.

Some periods contain transitions that improved the detection of plutonium. For example, from 1964 to 1977, electrodeposition of the plutonium replaced evaporation of the extract on the planchet. In addition, starting in 1973 with four detectors, plutonium samples were processed with an internal standard and were counted on a PHA system to establish the sample-specific recovery. The count time was also increased to 720 minutes. Because of the difficulty of determining which improvements apply to each sample, the MDAs in Table 5-6 do not account for the improvement until the transition was completed for all samples (i.e., the MDAs are favorable to claimants).

Table 5-6. Median MDA values for plutonium.[a,b]

Period	dpm/24-hr sample
1952–1953	0.57[c]
1954–1962	0.51[c]
1963	0.44
1964–1977	0.54
1978–1989	0.24
1990–1992	0.24
1993–present	0.02

a. The unit of the MDA values starting in 1990 is dpm/sample.
b. Sample-specific MDA values, if found in the record starting in 1990, should be used instead of the generic MDA values in this table.
c. Note that these values of MDA are lower than the reporting level of 0.88 dpm/24-hr sample used at RFP through 1961. Many urine results in this period were rereported with the actual value if greater than zero. For those rereported results, these MDA values apply instead of the original reporting level.

The uncertainty of the result was not quantified and reported in the record until approximately 1980 based on an examination of the Comprehensive Epidemiology Data Resource (CEDR) database. To estimate the uncertainty for results without a reported uncertainty, a reasonable approach is to divide the median MDA value by 3.3, where 3.3 is the sum of k_α and k_β, and $k_\alpha = k_\beta = 1.645$ (see Attachment A).

5.3.2 Americium Urinalysis

5.3.2.1 Methods, Units, Isotopes, and Interferences

Attachment A describes the methods through 1971. After 1971, the method for ^{241}Am paralleled that for plutonium.

The units of the results are dpm/24-hr excretion period through 1989. After 1989, the units of the results are dpm/sample regardless of the sample volume or excretion period. This statement is based on the observation that the same reporting format used for plutonium results was used for americium results.

The main interference is thorium, specifically ^{228}Th, which has two alphas with energies similar to those of ^{241}Am and has chemical properties similar to those of americium. If the chemical extraction procedure for americium was not run precisely, thorium would be eluted from the ion exchange column with the americium. When the extract was counted, even with the PHA system, the ^{228}Th could not be distinguished from the ^{241}Am. This biases the result high and is considered favorable to the claimant. The intake of the ^{241}Am is then calculated from the value of the initial parts per million of ^{241}Am measured or assumed for the plutonium mixture involved in the intake.

5.3.2.2 Americium Reporting Levels, Minimum Detectable Amounts, and Uncertainties

The reporting levels for americium were ≥0.24 dpm/24 hr in 1963, ≥0.2 dpm/24 hr from 1964 to 1967, and ≥0.30 dpm/24 hr from 1968 to 1971. Results less than the reporting level were reported as zero or background (or some abbreviation; e.g., BK). The reporting practice for the period from 1972 to 1976 has not been determined. Until it is determined, dose reconstructors should assume that the reporting level for 1968 to 1971 was continued through 1976, which is considered favorable to the claimant. Starting in 1977, all results ≥0.00 dpm/24-hr sample were reported. Negative results were reported as zero through 1989. After 1989, the actual negative value was reported. As for plutonium, urine results were not normalized to a 24-hour sample starting in about 1990. Instead, the results are dpm/sample, regardless of the sample volume (ORAUT 2003).

The MDAs for americium (Table 5-7) were determined as described for plutonium (see Section 5.3.1.2 and Attachment A), with the difference that the americium analyses started in 1963.

Table 5-7. Median MDA values for americium.[a,b]

Period	dpm/24-hr sample
1963	0.44
1964–1965[c]	0.55
1965–1970[c]	0.46
1971–1977	0.76
1978–1989	0.31
1990–1992	0.3
1993–present	0.02

a. The unit of the MDA values starting in 1990 is dpm/sample.
b. Sample-specific MDA values, if found in the record starting in 1990, should be used instead of the generic MDA values in this table.
c. In overlapping years the more favorable MDAs should be assumed.

The discussions of MDA and uncertainty for plutonium urinalysis in Section 5.3.1.2 apply to americium urinalysis.

5.3.3 Uranium Urinalysis

5.3.3.1 Enriched Uranium

5.3.3.1.1 Methods, Units, Isotopes, and Interferences

The units of the results are dpm/24-hr excretion period for the entire period. Because urine samples analyzed for EU were counted with the air proportional detectors, all of the alpha-emitting isotopes of uranium are included in the result. Site-specific information about possible interferences that might have occurred for the urinalysis methods for EU is not available.

5.3.3.1.2 Reporting Levels, Minimum Detectable Amounts, and Uncertainties

Table 5-8 lists the MDAs for EU. The reporting level for EU through 1963 was ≥8.8 dpm/24-hr sample (10% of the RFP tolerance level). From 1964 to 1971, the minimum reporting level ranged from 20 to 28 dpm/24-hr sample depending on the volume of the sample as observed from the urinalysis data logs for that period. Results less than the reporting level were reported as zero or background (or some abbreviation; e.g., BK).

Table 5-8. Median MDAs for EU.

Period	dpm/24-hr sample
1952–1953	14
1954–1959	13
1960–1963	9.4
1964–1969	31
1970–1971	25

The MDAs for EU were determined as described for plutonium (see Section 5.3.1.2 and Attachment A).

Uncertainties for the EU urine results have not been quantified or reported. To estimate the uncertainty for results without a reported uncertainty, a reasonable approach is to divide the median MDA value by 3.3, where 3.3 is the sum of k_α and k_β, and $k_\alpha = k_\beta = 1.645$ (see Attachment A).

5.3.3.2 Depleted Uranium

5.3.3.2.1 Methods, Units, Isotopes, and Interferences

Attachment A describes the uranium urinalysis methods through 1971. From 1972 to 1979, DU samples were chemically processed with the uranium-specific trioctylphosphene oxide (TOPO) extraction procedure, and the electrodeposited extract was counted on the gas flow proportional counter. From 1980 to 1997, DU samples were processed with a tracer (^{232}U or ^{236}U) by ion exchange and alpha-counted with the alpha spectrometry system with surface barrier detectors in vacuum. The starting year of use of the tracer has not been determined. From 1997 to the present, DU samples were processed at an offsite commercial laboratory according to provisions of the bioassay statement of work (KHC 1998b).

The units for 1952 to April 1964 were micrograms of uranium per 24-hour excretion period. The mass measurement was for all the isotopes of uranium. From May 1964 to 1989, the units were dpm/24-hr sample. After 1989, the units of the results were dpm/sample, regardless of the sample volume or excretion period (ORAUT 2003).

The urine data logs through 1971 do not identify the involved isotopes. However, it is reasonable to assume that all the alpha-emitting uranium isotopes were included in the air proportional detector measurements. For the 1980s, ^{238}U contributes 89% of the alpha activity. Therefore, the logs have not been reviewed to determine the other uranium isotopes. In the 1990s, the urine data reports include the results separately for ^{234}U, ^{235}U, and ^{238}U.

The major interference is the contribution from natural uranium, which is ubiquitous, sometimes in concentrated pockets, in the terrain near RFP. No adjustments have been made to the reported DU urine results for this background, which was highly variable.

5.3.3.2.2 Depleted Uranium Reporting Levels, Minimum Detectable Amounts, and Uncertainties

The minimum reporting level for DU through April 1964 was 5.8 µg/24-hr sample (10% of the tolerance level). From May 1964 to 1971, the minimum reporting level was the same as that for EU (20 to 28 dpm/24-hr sample depending on the volume of the sample). The reporting level for 1972 to 1979 (TOPO procedure) has not been determined. An approach that is favorable to claimants is to use the reporting level for 1964 to 1971. In the 1980s, all results ≥0.00 dpm/24-hr sample were reported. Negative values were reported as 0.00 dpm/24-hr sample. In the 1990s and after, all actual results, including negative values, were reported.

The MDAs for DU for fluorometric measurements were determined as described in Attachment A. Median MDAs for DU from 1952 to April 1964 are listed in Table 5-9. For alpha-counting methods, the MDAs in the period from April 1964 to 1971 are the same as those for EU in Table 5-7. The MDA value for 1972 to 1979 was extrapolated from the value for the previous period. The MDAs for 1980 to the present were derived in the same manner as that for plutonium but are based on ^{238}U. Table 5-10 lists median MDAs for DU from May 1964 to the present.

Table 5-9. Median MDAs for DU from 1952 to April 1964.

Period	µg/24-hr sample
1952–1955[a]	31
1955–1959[a]	12
1960–04/1964	11

a. In overlapping years the more favorable MDAs should be assumed.

The discussion of the uncertainty for plutonium in Section 5.3.1.2 applies to DU.

Table 5-10. Median MDAs for DU from May 1964 to the present.[b]

Period	dpm/24-hr sample
05/1964–1969	31
1970–1971	25
1972–1979	25[c]
1980–1989	0.56
1990–1992	0.4[a]
1993–present	0.1[a]

a. The MDA value unit starting in 1990 is dpm/sample.
b. Sample-specific MDA values, if found in the record starting in 1990, should be used instead of the generic MDA values in this table.
c. Actual practice is unknown; assume continuation of earlier practice.

5.3.4 Gross Alpha Urinalysis

5.3.4.1 Methods, Units, Isotopes, and Interferences

Gross alpha measurement is a nonspecific analysis that was used for workers who were potentially exposed to both uranium and plutonium in the same monitoring period. Workers who were potentially exposed to other alpha-emitting radionuclides, such as neptunium and curium, might also have been monitored for gross alpha. Urinalysis methods are discussed in Attachment A. The gross alpha method was discontinued in the early 1970s, probably in 1973 based on an examination of the CEDR database. However, isolated gross alpha measurements were identified as late at January 1980. The results are reported as dpm/24-hr sample of either EU (the default analyte through 1963) or plutonium (the default analyte after 1963). Interferences are likely, because the methods were nonspecific. The analyzed isotopes were all of the alpha-emitting isotopes of the analyte.

5.3.4.2 Reporting Levels, Minimum Detectable Amounts, and Uncertainties

The reporting level for gross alpha through 1963 was ≥8.8 dpm/24-hr sample (10% of the RFP tolerance level for EU). After 1963, the reporting level was ≥0.9 dpm/24-hr sample and credited to plutonium. Gross alpha data are coded as G in the urine data reports. Urinalysis code G was observed in HSDS urinalysis reports to 1972. Code G correlates with the gross alpha B2 analysis code on the Urinalysis Record Card (see, for example, Figure C-2).

Samples with results ≥0.9 dpm/24-hr sample were typically but not always counted using a PHA system to determine whether to credit the result to EU, to plutonium, or to a portion to both. The default condition through 1963 was to credit the result to EU unless the PHA count indicated otherwise. After 1963 (when EU operations were phased out), the default condition was to credit the result to plutonium. In either case, the results is considered an upper bound because of the nonspecificity of the analysis.

The MDAs for gross alpha in Table 5-11 were determined as described in Attachment A.

Table 5-11. Median MDAs for gross alpha measurements.

Period	dpm/24-hr sample
1952	1
1953	0.88
1954–1959	0.79
1960–1962	0.55
1963	0.55
1964–1971	0.69

5.3.5 Unmonitored Tritium Exposure

The unmonitored tritium exposure rates in Table 5-12 should be assigned to all workers at RFP. Attachment G provides the basis for the exposure rates.

Table 5-12. Unmonitored tritium exposure rates.

Period	Dose rate (rem/yr)
Before 1973	0.0375
1973	0.103
After 1973	0.0

5.3.5.1 Reporting Levels, Minimum Detectable Amounts, and Uncertainties

Starting in 1973, workers were monitored for possible tritium exposures only for special projects or situations. The methods have not been reviewed but probably involved liquid scintillation measurements. It is assumed that liquid scintillation was used in the 1970s. The urine results are reported as picocurie per liter of urine, and actual results were reported, generally with the standard deviation. It has not been determined whether the reported uncertainty in the 1970s to early 1980s is 1 or 2 times the standard deviation. The sensitivity of the method was 2 to 3 orders of magnitude better than the significant level of about 1 µCi/L. Although the actual MDA has not been quantified for the methods in the 1970s and 1980s, it is probably in the range of several hundred to several thousand picocuries per liter (AEC 1973, ca. 1974; Various 1955–1976; RFP 1952–1971). The MDA for tritium should be assumed to be 600 pCi/L for all tritium bioassay (KHC 1998c, p. 176).

5.3.6 Alvin E. Blackwell Bioassay Laboratory

In late 1991, RFP identified an issue with their outside bioassay laboratory Alvin E. Blackwell Bioassay Laboratory (AEB). The RFP Internal Dosimetry Committee meeting notes indicate that the site determined that they needed to send a set of quality control samples to AEB before they send a group of baseline samples for analysis to determine if AEB is qualified. Six of the eight quality control samples sent to AEB failed the data quality objectives (EG&G 1992).

In a review of individual dosimetry records, sample results assessed by AEB included a cover letter that states that these results are not to be considered valid data. In addition, all personnel would be resampled.

Any bioassay results analyzed by AEB should not be considered valid. The individual should be considered unmonitored during this period unless non-AEB bioassay results are available.

5.4 IN VIVO

In vivo lung counts have been performed at RFP since 1964 to determine the activity of plutonium in the lungs of workers who were exposed, or had the potential to be exposed, to airborne plutonium. The method of in vivo lung counts was to place one or more detectors over the chest of the subject and count the photons that are emitted from the plutonium mixture, if any, in the chest. Plutonium was not detected directly because of the low abundance of gamma photons and the severe attenuation of the more abundant low-energy X-rays (L X-rays). Rather, the 59.5-keV gamma photon from ^{241}Am was used to detect ^{241}Am, which is present to some extent in all WG plutonium at RFP. The activity of plutonium was then calculated from the detected ^{241}Am by measuring, calculating, or assuming the fraction of the ^{241}Am in the plutonium mixture on the date of the lung count (see Section B.11 in Attachment B). At RFP, the fraction of the ^{241}Am in the plutonium mixture has historically been characterized in terms of parts per million by weight. Direct in vivo measurement of plutonium in the lungs, although investigated, was never implemented at RFP (Falk et al. 1979).

The RFP lung counter also measured ^{234}Th, using the 63-keV gamma (doublet) photon, to determine the activity of ^{238}U in the lungs of workers exposed to DU. This measurement was made possible by the improved resolution of the germanium detectors that allowed baseline separation of the 59.5-keV gamma of ^{241}Am from the 63-keV gamma doublet of ^{234}Th. The activity of ^{238}U was considered to be equal to that of the measured ^{234}Th under the assumption of equilibrium (Berger 1988a).

Attachment B, Minimum Detectable Amounts for In Vivo Lung Counts at RFP, contains more detail. Section 5.6 discusses the data, and Attachment C contains examples of the report forms.

5.4.1 Americium and Plutonium

5.4.1.1 Methods, Units, Isotopes, and Interferences

Before May 1995, lung count data were not converted to a quantified amount or activity unless there was confirmation that the count was from an actual deposition in the lungs. For unquantified results, the data are generally in units of counts per minute and accompanied by a decision that is noted as normal, background, or some abbreviation of background. For quantified results through about 1968, the unit was micrograms of plutonium. In addition, the result was converted to a fraction of the maximum permissible lung burden (MPLB) using a plutonium-specific activity of 0.07 µCi/µg and the MPLB of 0.016 µCi (16 nCi) for the alpha-emitting isotopes of plutonium. Starting in about 1973, the activities of both plutonium (including all the alpha-emitting isotopes of WG plutonium) and americium (^{241}Am) were recorded in nanocuries. Figure C-18 is an example of the implementation of these modifications. In addition, the activity of ^{241}Am was stated as a fraction of the MPLB, which was 14.7 nCi (Falk ca. 1993). After 1989, the results were no longer stated as a fraction of the MPLB.

There are two sources of interferences to consider. The first is the 63-keV gamma doublet of ^{234}Th from DU operations being mistaken for ^{241}Am in lung counts with the NaI or phoswich detector systems. The second interference is the contribution of count from ^{241}Am not in the lungs (e.g., contributions from contamination on the skin, activity in the lymph nodes, etc.).

5.4.1.2 Reporting Levels, Minimum Detectable Amounts, and Uncertainties

The decision levels varied. From 1965 to 1968, the decision level was two times the uncertainty of the matched subject's net count, although the application of this decision level was inconsistent in this period. Starting in 1969, for NaI and phoswich detector systems, the decision level was 3 times the standard deviation of the net count rate for a set of lung counts for unexposed known cold subjects based on the index method (see Attachment B). Results between 2 and 3 sigma were noted but not always investigated. For the germanium detector systems, starting in 1976, the decision level (also called the "cutoff") was equal to 1.645 times the standard deviation of the net count rate. The cutoff, as defined, is based on limiting the probability of a Type I error (false positive) in the signal domain to 5%. Figure C-22 is an example of the implementation of this decision level. The decision level for 1995 and later was calculated by ABACOS-Plus for a probability of a Type I (false positive) error of 5% (KHC 2000b, p. 90). The decision level was used as a reporting level from May 1995 to mid-February 1997. For ABACOS-Plus lung count reports, from May 1995 to mid-February 1997, when a result above the decision level is reported, the MDA is assumed to be twice the decision level. The decision level is determined as 1.645 times the 1-sigma standard deviation of the measurement. Note that during this time the 2-sigma percent error was normally reported. Therefore the MDA would equal to 1.645 × (measured activity) × (2-sigma percent error) ÷ (100).

Table 5-13 lists the MDAs for ^{241}Am, which were calculated for the evolution of lung-counting systems at RFP as described in Attachment B.

These values of MDAs are for three indices that represent the median and the approximate 5th- and 95th-percentile body statures of RFP male workers. To obtain the worker-specific MDA, dose reconstructors can calculate the value using the information in Attachment B or interpolate (or extrapolate) from the values in Table 5-13. The worker-specific index is generally stated on lung count report forms from 1969 to 1994 and can be derived from the weight and height data on report forms from mid-February 1997 and later. (The MDA values are reported on report forms from mid-February 1997 and later, but the values are not worker-specific. Dose reconstructors should disregard these MDA values.) The default MDA would be for an index of 1.35 (median value for a range of 0.9 to 1.8) if height and weight (or index) data for the worker are not available. Dose reconstructors should assume the MDA is twice the decision level for mid-February 1997 and later lung count reports that include the non-worker-specific MDA.

The MDA for plutonium should be calculated by multiplying the worker-specific value of the MDA for ^{241}Am by the MDA conversion factor (Equation B-17 in Attachment B), which is based on the value of the ppm ^{241}Am on the date of the lung count. The value of the ppm ^{241}Am on the date of the lung count, accounting for ingrowth of ^{241}Am from the nuclear transformation of ^{241}Pu and the radioactive decay of the initial ^{241}Am, is given by Equation B-18 in Attachment B.

The uncertainties of the results were reported for the net counts per minute starting with the germanium detector systems in 1976. The uncertainty was reported at 1 standard deviation and included only the contribution from counting statistics. Starting in approximately 1981, the counting statistics uncertainty was also applied to the assessed activity and to the value of the fraction of the MPLB. Figures C-20 and C-21 are examples of the implementation of these modifications. With the advent of ABACOS-Plus in 1995, the percent error at 2 standard deviations was reported for all identified nuclides until around mid-February 1997, and the percent error at 1 standard deviation was reported for all identified nuclides after that time. Beginning on October 11, 1999, a 30% systematic uncertainty, which included contributions of uncertainties in the chest wall thickness (CWT), the location of the activity in the lungs, the uncertainty in the ppm ^{241}Am, and the influence of activity deposited in other organs, was included in the total propagated uncertainty (KHC 2000b, p. 89).

5.4.2 Thorium and Depleted Uranium

5.4.2.1 Methods, Units, Isotopes, and Interferences

The method to detect DU was to detect the 63-keV gamma (doublet) photon of ^{234}Th and to calculate the activity of ^{238}U assuming equilibrium. This method was implemented manually for special cases in approximately 1978. Starting in 1983, the count data for the 63-keV doublet photon were routinely processed and reported. However, the activity of the ^{238}U was calculated only for special cases and not routinely. A supplemental method, implemented in about 1989, detected the 93-keV gamma doublet photon of ^{234}Th, and the count data were routinely processed and reported. This supplemental method was used mainly to reduce false positive results for the detection of ^{234}Th because detection of both doublet photons was required before detection of ^{234}Th was considered.

Starting in 1995, the activity of ^{238}U was calculated and reported if the 63-keV peak (or sometimes the 93-keV peak) was detected by the ABACOS-Plus peak-search software. If the peak was not detected, the activity of ^{238}U was reported as less than the decision level (the activity of the decision level was reported). Starting in mid-February 1997, the activity of ^{238}U was reported, including negative results, even if a peak was not detected. In a similar manner, the activity of ^{235}U was reported. Starting in about 1999, the activity of ^{238}U was based solely on the 63-keV peak.

The main part of the data for the 63-keV doublet photon is in units of net counts per minute. To convert to activity (nanocuries) of ^{238}U, the counts per minute is divided by the calibration factor for ^{241}Am (see Attachment B) and normalized to the ratio of photon abundances [abundance of 59.5-keV

Table 5-13. Summary of MDAs (nCi) for ^{241}Am.

Period[a]	Detector system	Index	Minimum system half time	Minimum system full time	Standard system half time	Standard system full time
1964–1968	NaI(Tl) 4 × 4	0.90	1.7	1.5	1.3	1.2
1964–1968	NaI(Tl) 4 × 4	1.35	2.8	2.5	2.1	1.9
1964–1968	NaI(Tl) 4 × 4	1.80	4.6	4.1	3.5	3.2
1969→	NaI(Tl) 4 × 4	0.90	Not applicable	Not applicable	0.8	0.76
1964–1968	NaI(Tl) 4 × 4	1.35	Not applicable	Not applicable	1.3	1.3
1964–1968	NaI(Tl) 4 × 4	1.80	Not applicable	Not applicable	2.2	2
1973→	Phoswich	0.90	Not applicable	Not applicable	1.2	1.2
1973→	Phoswich	1.35	Not applicable	Not applicable	2.0	2.
1973→	Phoswich	1.80	Not applicable	Not applicable	3.3	3.2
1976–1978	Ortec Arrays (high-purity Ge)	0.90	0.26	0.18	0.2	0.14
1976–1978	Ortec Arrays (high-purity Ge)	1.35	0.48	0.32	0.37	0.25
1976–1978	Ortec Arrays (high-purity Ge)	1.80	0.86	0.59	0.66	0.45
1979→	Ortec Arrays (high-purity Ge)	0.90	0.2	0.14	0.16	0.11
1979→	Ortec Arrays (high-purity Ge)	1.35	0.37	0.25	0.28	0.19
1979→	Ortec Arrays (high-purity Ge)	1.80	0.66	0.45	0.51	0.35
1978→	PGT I Arrays (high-purity Ge)	0.90	0.22	0.15	0.17	0.12
1978→	PGT I Arrays (high-purity Ge)	1.35	0.4	0.27	0.31	0.21
1978→	PGT I Arrays (high-purity Ge)	1.80	0.71	0.49	0.55	0.38
1979→	PGT I Arrays (high-purity Ge)	0.90	0.17	0.12	0.13	0.09
1979→	PGT I Arrays (high-purity Ge)	1.35	0.31	0.21	0.24	0.16
1979→	PGT I Arrays (high-purity Ge)	1.80	0.55	0.38	0.42	0.29
1979→	PGT II Arrays (high-purity Ge)	0.90	0.22	0.15	0.17	0.12
1979→	PGT II Arrays (high-purity Ge)	1.35	0.4	0.28	0.31	0.21
1979→	PGT II Arrays (high-purity Ge)	1.80	0.74	0.5	0.57	0.39
1985→	PGT Organ Pipe Ge Detectors	0.90	Not applicable	Not applicable	0.15	0.11
1985→	PGT Organ Pipe Ge Detectors	1.35	Not applicable	Not applicable	0.26	0.18
1985→	PGT Organ Pipe Ge Detectors	1.80	Not applicable	Not applicable	0.46	0.32
1991→	EG&G Organ Pipe Ge Detectors	0.90	Not applicable	Not applicable	0.14	0.1
1985→	PGT Organ Pipe Ge Detectors	1.35	Not applicable	Not applicable	0.26	0.18
1985→	PGT Organ Pipe Ge Detectors	1.80	Not applicable	Not applicable	0.48	0.33
1995→	Ortec 2 Organ Pipe Ge Detectors	0.90	Not applicable	Not applicable	Not applicable	0.14
1985→	PGT Organ Pipe Ge Detectors	1.35	Not applicable	Not applicable	Not applicable	0.3
1985→	PGT Organ Pipe Ge Detectors	1.80	Not applicable	Not applicable	Not applicable	0.6

a. In overlapping years, the more favorable MDAs should be assumed.

gamma, ^{241}Am, is 0.359; abundance of 63-keV doublet gamma, ^{234}Th, is 0.0381 (Lederer and Shirley 1978); the ratio (59.5-keV gamma/63-keV doublet gamma) is 9.4]. That is, nanocuries of ^{238}U equals [(^{234}Th 63-keV net cpm) divided by (^{241}Am calibration factor)] multiplied by 9.4. To calculate the activity for DU, the ^{238}U activity is divided by 0.89 (see Section 5.2.3.2.1).

5.4.2.2 Reporting Levels, Minimum Detectable Amounts, and Uncertainties

Reporting levels were not generally used for DU until 1995 with the implementation of ABACOS-Plus (see Section 5.4.2.1). Before 1995, the ^{238}U activity was generally quantified only after verification of an intake. For ABACOS-Plus lung count reports, from May 1995 to mid-February 1997, when a result above the decision level is reported, the MDA is assumed to be twice the decision level. The decision level is determined as 1.645 times the 1-sigma standard deviation of the measurement. Note that during this time the 2-sigma percent error was normally reported. Therefore the MDA would equal to 1.645 × (measured activity) × (2-sigma percent error) ÷ (100).

The MDA for ^{238}U has not been determined rigorously. However, the ^{238}U worker-specific MDA can reasonably be expected to be a multiple of the ^{241}Am worker-specific MDA because the detected photons (63 keV and 59.5 keV) are very close in energy. As described in Section 5.4.2.1 for using the calibration factor for ^{241}Am to determine the ^{238}U activity, the ^{238}U worker-specific MDA can be obtained by multiplying the ^{241}Am worker-specific MDA by 9.4. That result is divided by 0.89 to obtain the worker-specific MDA for DU. Dividing the ^{238}U MDA by 0.89 accounts for the contribution to the DU MDA from activities of the other uranium isotopes.

As noted in Section 5.4.1.2 for americium and plutonium, MDA values are reported on forms for mid-February 1997 and later, but are not worker-specific. Dose reconstructors should disregard these MDA values. Dose reconstructors should assume the MDA is twice the decision level for mid-February 1997 and later lung count reports that include the non-worker-specific MDA.

The major uncertainty is the assumption of equilibrium of the ^{234}Th with the ^{238}U before 1990, when DU was still being processed. Part of the process was to remove decay chain radionuclides, especially thorium, by heating the uranium ingot to drive the smaller atoms of thorium to the surface or top of the ingot, which was then cut off. The result was DU metal with a deficiency of ^{234}Th for several weeks plus scrap DU with an excess of ^{234}Th (super-equilibrium). The assumption of equilibrium when super-equilibrium existed is favorable to claimants. If a superequilibrium situation was operative and the ^{234}Th lung count result was used to calculate the DU assuming equilibrium, the calculated DU would be higher than the actual activity. Therefore, the approach is favorable to claimants. The effect of a deficiency of ^{234}Th (not favorable to claimants) is mitigated by the rapid ingrowth of the ^{234}Th into the DU. Fifty-percent equilibrium occurs after 24 days after a thorium strike, and 90% occurs after 80 days.

5.5 OTHER BIOASSAY DATA

5.5.1 Wound Count Data

Wounds are defined as any break in the skin (e.g., cuts, punctures, abrasions, acid burns). Any wound that occurred in a work area involving plutonium was monitored for plutonium contamination, especially after the advent of the wound counter in 1957. Counting a blood sample or directly counting the wound site with an alpha detector were also methods RFP used to monitor wounds to detect possible plutonium contamination. In RFP terminology in the 1950s and 1960s, wound counts were called “gamma specs,” and the wound counter was called a “gamma spectrometer.” Wounds in uranium work areas were monitored selectively. The record could contain an incident report, a wound count data sheet, a medical decontamination report, and a medical treatment report, depending on the era and circumstances.

The process was to attempt to decontaminate the wound in the building of the occurrence by washing and encouraging bleeding to flush any plutonium out of the wound. Then the worker was sent or escorted to the medical facility for a wound count and additional decontamination if the wound count was positive (Berger 1988b). The sequence of additional decontamination was washing with soap and water, washing with commercial bleach, scrubbing with commercial bleach, and excision.

Guidance on assessing wound intakes is provided in ORAUT-OTIB-0022, *Guidance on Wound Modeling for Internal Dose Reconstruction* (ORAUT 2005a).

5.5.2 Nasal Smears and Fecal Samples

Nasal smear (later called swab) and fecal sample data were occasionally performed throughout RFP operations as supplemental data for workers with actual or suspected significant inhalation intakes. Through the 1980s, they were used subjectively to verify that an intake did occur and to estimate the possible magnitude of the intake. The data were also used to determine or confirm the ppm ^{241}Am in the inhaled plutonium mixture. Some obstacles to using nasal smear or fecal data to quantify an intake are unknown particle size distribution, unknown fraction of the plutonium captured by the nasal smear or fecal sample, inconsistent and largely undocumented sampling technique for nasal smears (which sometimes were called “nose blows”), and unknown counting efficiency (e.g., sample geometry and alpha absorption, especially in the 1950s and 1960s). Through 1989, the requested fecal sample was the second voiding after the incident. In some cases, the second, third, and fourth voidings were requested.

Starting in the 1990s, the nasal or mouth smears were used as a workplace indicator to identify potential intakes, and fecal sampling was used to confirm and evaluate suspected intakes (KHC 1998d, p. 62).

The reported MDAs (KHC 1998d, pp. 67–68) are:

- 20 dpm/sample for routine nasal samples (gross alpha, liquid scintillation);
- 0.2 dpm/sample for fecal samples with a 21-day reporting time (plutonium alpha isotopic);
- 1.3 dpm/sample for fecal samples with a 14-day reporting time (plutonium alpha isotopic);
- 2.6 dpm/sample for fecal samples with a 7-day reporting time (plutonium alpha isotopic); and
- 100 dpm/sample for fecal samples with a 2-day reporting time (nonisotopic, rapid analysis).

The reporting times are the times for the laboratory to analyze the sample and report the results. The shorter reporting times indicate an expedited analysis, with the trade-off of a less sensitive analysis (a higher MDA).

These MDA values apply to samples starting approximately in 1993 and are specifications for the laboratory. (Note: The laboratory MDA does not depend on the time after intake that the sample was excreted.) Most reports of fecal sample results do not give the sample-specific MDA but might give the decision level, which is approximately one-half of the sample-specific MDA. MDA values for earlier years are not available.

5.6 RECORDS AND REPORTS

This section discusses the interpretation of the data and information on records and reports of bioassay data. Attachment C, Examples of Records and Reports Used at RFP, contains the figures described below.

5.6.1 Urinalysis Records and Reports

Figures C-1 to C-3 are examples of the Urinalysis Record Card and the HSDS – Urinalysis Detail report. The Urinalysis Record Card was the recording medium for the urinalysis data from 1952 to 1969 and is the primary record for urine data in this period. The urine data were manually entered on this card through 1969. These data were also entered into a database starting in about 1961. In about 1970, the HSDS was implemented to record, process, and report urinalysis data and the derived fraction of the maximum permissible systemic burden.

5.6.2 Interpretation of the Urinalysis Record Card

Urine results are presented in columns under the month for a given year (in the row). The top number is the day of the month (assumed to be the excretion day). The middle number is the sample result, either a number or BK (see Section 5.3.1.2). The bottom number is the technique code and refers to the codes in the header (see Attachment A).

The unit of the result is given in the header. Sometimes the unit is written with the result (e.g., µg in Figure C-1, analysis Code A, 1955). Be careful not to interpret µg as the number 49.

The corresponding data on the HSDS – Urinalysis Detail report should be the same as that on the Urinalysis Record Card. If not, the data on the Urinalysis Record Card should be taken as the correct data, with the exception noted in Section 5.3.1.2 (i.e., some plutonium results reported as BK on the card were rereported with the actual result). The urine data record written on the Urinalysis Record Card preceded the HSDS and was the probable source of the urine data loaded into the HSDS and its mainframe database predecessors. Because there could have been transcription errors during the preparation of the data (punched cards in the 1960s) for loading into the mainframe, the data of the Urinalysis Record Cards (the source data) should be considered the correct data, as recommended. On some cards, dose reconstructors might observe the initially reported result was crossed out and replaced by a lower value. The technical basis for that change has not been determined. In addition, that change generally was not applied to the data in the HSDS. It is reasonable and favorable to claimants to disregard the modified result. This recommendation is reasonable because the basis for the change is not known and the change was not made in the HSDS. It is favorable to claimants because the original record is the higher value.

The analyte code for DU was sometimes transcribed incorrectly from the card to the urinalysis detail report as U (see Figure C-1) rather than D (see Figures C-2 and C-3) with the unit of dpm/24-hr sample rather than µg/24-hr sample.

Figures C-4 and C-5 are two versions of urinalysis reports from the HSDS. Both versions report the data in the same way but with differences in the headers. Figure C-5 (the newer version) adds a column (the uncertainty of the result).

5.6.3 Interpretation of the Health Sciences Data System – Urinalysis Detail Report

The Activity Date is taken to be the date that the sample was excreted. However, the recorded date frequently was the date that the sample was received at the laboratory, especially for routine samples. (This applies also to the dates on the Urinalysis Record Card.)

ANAL is the code for the analyte:

P = plutonium,
A = americium,
U = EU (pre-1970, approximately),

U = DU (1970–1989, approximately),
D = DU (1952–1969, approximately), and
G = gross alpha.

NO CAL is a code used to flag the logic of the software.

0 = use normally in the calculation;
1 = do not use in the calculation; and
2 = date of a new intake.

Code 1 was used primarily for two situations to exclude a sample result from the systemic burden calculation: If the excretion of the analyte was enhanced by a chelation treatment or if the analysis of the sample did not meet quality standards (an invalid analysis or result). Sample results within 90 days of a chelation treatment were generally (or should have been) coded as 1 (Various 1965–1987, p. 28). The use of Code 2 to flag the date of a new significant intake occurred inconsistently. In reports from the 1980s, an asterisk was used instead of a Code 2 to flag the date of a new intake. Dose reconstructors should disregard the Code 2 or asterisked entries.

- ELAPSED DAYS is the number of days since the hire date. This data field is not likely to be of use.

- The EXPOSURE VALUE or DPM/24HR is the result of the urinalysis for the analyte. In general, the unit was dpm/24-hr sample, except for DU, from 1952 to April 1964.

- The column in parentheses is the uncertainty, starting in 1980. Any value or symbol in the parentheses before 1980 is only a placeholder and should be disregarded. This circumstance is evident in the example reports in Figures C-2, C-3, and especially C-5.

- The BODY BURDEN % or SYSTEM BURDEN is the fraction of the maximum permissible systemic burden that was calculated from Code 0 results for plutonium and for americium. This data field is not likely to be of use.

5.6.4 Interpretation of Other Urinalysis Reports

Figures C-6 and C-7 are examples of urinalysis reports from the onsite bioassay laboratory from 1990 to the mid-1990s. Figure C-6 is for a special urine sample for plutonium analysis, and Figure C-7 is for a routine urine sample for plutonium analysis. Both forms have the same format. The first three columns are self-explanatory; the remaining columns are:

- Dec Level is the decision level in units of dpm/sample.

- Aspec is code for the alpha spectrometry quality. The Aspec codes are defined on the lower left portion of the report. Aspec code 0 is analogous to the previous Code 0 for urine data in the HSDS. Codes 1, 3, and 4 indicate a failed analysis and disqualify the result (Various 1982–2005, p. 160).

- DQO, for “data quality objective,” is the code for status of the data quality objectives for the results of the batch blank and control samples. The DQO codes are defined on the lower center portion of the report. DQOs, in theory, were assessed for the blank, accuracy, and precision. In practice, the DQO was usually assessed only for the blank. Therefore, the ANN notation means that the blank was acceptable, the accuracy was not assessed, and the precision was not assessed. An F would indicate that the batch failed a DQO (Various 1982–

2005, p. 160). If the batch failed, every sample in the batch was conditionally failed pending further evaluation.

- Batch Val is the overall validation of the result. "V" means valid, and "I" means invalid. Do not use a result that has an "I" validation code.

- Analyte is self-explanatory.

- Recovery is the fraction of the tracer recovered by the analysis.

- DPM is the result of the sample in dpm/sample. Dose reconstructors should assume a 24-hour urine sample unless there is information that indicates otherwise.

- Error is the uncertainty at 1 standard deviation.

Figure C-8 is an example of the urinalysis data report by Quanterra, a commercial offsite laboratory, starting in 1993. The form header information, except for the collection date and the matrix, is not useful. The collection date, if not the sample excretion date, should be replaced by the sample date written on the form. The offsite laboratory sometimes was not provided the date on which the worker excreted the urine sample. In such cases, the excretion date was written on the report, as was the case for the report in Figure C-8. The guidance for dose reconstructors is to use the date written on the report in this case. The result header is largely self-explanatory.

- The primary information is the RESULT and its TOTAL ERROR (at 1 standard deviation) in dpm/sample (REPORT UNIT).

- The decision level and the sample-specific MDA are also stated.

- The YIELD is the percent recovery of the tracer.

- The RST/MDA is the ratio of the result and the sample-specific MDA.

- The RST/CNTERR is the ratio of the result and the counting error.

- The ANALYSIS DATE is the date the sample was analyzed, not the excretion date.

- The ALIQUOT SIZE is the volume of the sample in milliliters (ALQ UNIT).

- The DETECTOR ID is self-explanatory.

- The METHOD NUMBER references the document number of Quanterra's analytical procedure used to process the sample.

Figures C-9 and C-10 are examples of the analytical report of the onsite bioassay in the mid-1990s. Most of the information is self-explanatory. Some points:

- The date sampled is the excretion date.

- The data can only be used if the Alpha Spec Condition Code is 0 and if the Data Validation Code is V.

- The ^{234}U activity is approximately equal to ^{238}U activity in Figure C-9, and both results are greater than the decision level. As stated in Section 5.3.3.1.1, this is the classic pattern indicating natural uranium, not an occupational intake of DU.

Figure C-11 is an updated version of the urinalysis data report of Quanterra. The significant improvement is the validation of each result (QUAL is V). Use only results with a QUAL of V.

Figure C-12 is the urinalysis data report for General Engineering Laboratories. The header information is largely self-explanatory.

- The Date Collected is the sample excretion date. The 24-hour clock time (0600) is also noted; 0600 was used as a default end time of the 24-hour excretion period if the actual end time was not documented.

- The VF is the volume fraction, the fraction of the sample that was analyzed. A volume fraction of 1 indicates that the entire sample was analyzed.

- Use only data that have a Data Validation Code of V.

Figure C-13 is an example of the data card that was used in the 1970s and 1980s to record data manually for tritium urine samples and for other samples such as fecal samples and nasal smears. The unit of the tritium results is pCi/L. The unit of the fecal sample and nasal smear results is dpm/sample based on an examination of the CEDR database.

There might be other versions of in vitro bioassay reports. In all cases, the important data are the excretion date, the analyte, the result in the proper units, and whether the result was valid.

5.6.5 Lung Count Records and Reports

Figure C-14 is an example of an early lung count report. The aftermath of the October 15, 1965, plutonium fire in Buildings 776 and 777 was the first extensive use of the lung counter to detect americium and plutonium depositions for RFP workers.

- The in vivo lung-counting system was called the Body Counter. In RFP terminology, the lung count was called a body count through 1989. Most claimants will probably use the term “body count” instead of “lung count.” Dose reconstructors should not mistake the RFP “body count” for a whole-body count, which was widely used at other facilities to detect intakes of fission products.

- The Time field was used either for the time of the day at the start of the count or for the length of the count. In this case, the length of the count was noted as 40 minutes live time (MLT).

- The “Minus Bkg + match” notation indicates that the result is the net count rate after the room background count rate and the net count rate of a matched person was subtracted.

- The “1.4 LB” notation is the calculated plutonium deposition in terms of the multiple of the MPLB of plutonium (1 MPLB = 16 nCi plutonium alpha emitters). The value of the MPLB for plutonium alpha emitters (^{239}Pu and ^{240}Pu) was calculated using Equation 4 in ICRP Publication 2 (ICRP 1959) for an annual dose of 15 rem (0.3 rem/wk), organ mass (m) = 1,000 g, f2 = 1, and $\varepsilon = 53$ (ICRP 1959, Table 5; and based on a relative biological effectiveness = 10).

- The Body Location is the position of the detector. In this case, the detectors were over the right and left portions of the chest. In many early counts, one of the detectors was over the liver or gut or below the sternum rather than over one side of the chest.

Figure C-15 is the August 1967 revision to the Health Physics Body Counter Information form. The change was to present the results after subtraction of the room background [Net (1) c/m] and after subtraction of matched subject net cpm [Net (2) c/m]. In addition, the plutonium deposition was stated in terms of micrograms of plutonium.

Figure C-16 is the August 1968 revision to the Health Physics Body Counter Information form.

- The Net cpm is the subject's total count rate minus the room background count rate.
- The Predicted cpm replaced the net count rate of the matched subject.
- The Result is the final net cpm.
- In this example, there is no measurement for the right chest. Dose reconstructors should estimate the contribution for the right chest before using data from this count, because the lung dataset generally includes contributions from both right and left lungs.

Figure C-17 is an example of a lung count with no tabulated result. This is an example of a positive unknown case (see Section 5.4.1.2). In addition, note the tabulation of the index, which was used later to estimate the chest thickness. Sufficient information is presented here and in Attachment B to allow dose reconstructors to calculate the plutonium and americium activities for this lung count, for any assumed or actual intake date.

Figure C-18 is the December 1973 revision to the previous form, with expanded information.

- The ROOM is the designation of the counting chamber, A, B, or C, used for this count.
- The RATIO field is the ratio of the ^{241}Am photopeak region of interest (ROI) and a background ROI around 100 keV. The ratio was used as a supplemental subjective tool to improve detection of americium.
- The ppm ^{241}Am was used to record either the ppm ^{241}Am for a new incident or, as in this case, the calculated value of the ppm ^{241}Am (including of ^{241}Am) for a previous actual or assumed intake.

The form included fields to record the activity and fraction of the MPLB for both plutonium and americium. (This lung count, now quantified, is for the same positive unknown case as Figure C-17).

Figure C-19 is an example of the previous form for a count that was judged to be background. Data fields were added to capture data for measurements of the L X-ray (17-keV) ROI, especially for the phoswich detector system.

The previous lung count reports were for counts using the NaI detector system. Figure C-20 is an example of the lung count data for a germanium detector system. The data for the five to eight detectors of the germanium systems were multiplexed into a composite total count tabulated in the row for TOTAL CHEST. The standard deviation of the resultant counts per minute is based only on counting statistics. For workers with confirmed lung depositions, the calibration factors for plutonium and americium were generally written on the form, as in this case.

Figure C-21 is an example of the first computer report for the lung count results. The data are labeled appropriately. This report is for a worker with a confirmed deposition. The report for workers without a confirmed deposition does not report the calibration factors, the ppm americium, or the lung burden. Rather, it reports the cutoff, which is the decision level, and Normal if the DIFFERENCE is less than the cutoff (see also Figure C-22 for example).

Figure C-22 is an example of a computer report for the phoswich detector system, which was used as a backup screening system in the 1980s. Note the outcome statement, RESULTS ARE NORMAL. Because the phoswich system could not resolve the 60- and 63-keV photopeaks, they share a common ROI. Another feature is the tabulation of the total count for each pertinent ROI. ROI 3 is the total count for the 60-keV to 63-keV ROI, and ROI 4 is the background count for the 60- and 63-keV photopeaks. ROI 4 was also used as the count for the 93-keV photopeak, and ROI 5 was its background. ROI 2 was probably the count in the L X-ray region, but it was not used.

Figures C-23 and C-24 are examples of the next generation of reports for the germanium detector systems. The innovation is the data capture in 10 ROIs. In Figure C-24, the ROIs are labeled with the photopeak of interest. Although the data were captured, most of the data were not used, mainly because the relationship between the photopeak and its background was not established or was too variable (see also Figure C-26 for example). ROI 5 (BKG in Figure C-24) is the common background (divided by a factor) for both the 60- and 63-keV photopeaks.

Figure C-25 is an example of a report for a worker with a confirmed deposition. There are no new fields.

Figure C-26 is an example of a report for a worker with no detected deposition and illustrates a frequent problem with the L X-ray data, namely low-end electronic noise in one or more of the detectors. Dose reconstructors should disregard all L X-ray data (including the 13- and 17-keV ROIs). Dose reconstructors are advised to disregard these L X-ray data because the counts were unreliable because of low-end electronic noise. Because of this unsolved problem in real time, a calibration factor to convert from count of plutonium activity was not established.

Figure C-27 is an example of a report on which data for the 93-keV photopeak are analyzed and presented.

Figure C-28 is an example of the next generation of reports. On this report, the ROI data for each detector are tabulated separately, as is the sum. ADC #1 stands for analog-to-digital converter for detector #1, which in this case is an EG&G detector, and similarly for the other detectors. This report does not report the results in terms of the fraction of the MPLB.

Figure C-29 is an example of the lung count report from an early version of ABACOS-Plus that was used through mid-February 1997. Because this software is based on a peak-search method, no ROI data are available. In addition, if a uranium or americium peak was not found, the activity was reported as less than the decision level. This statement is the result of direct observation of information in Figure C-29. For ABACOS-Plus lung count reports, from May 1995 to mid-February 1997, when a result above the decision level is reported, the MDA is assumed to be twice the decision level. The decision level is determined as 1.645 times the 1-sigma standard deviation of the measurement. Note that during this time the 2-sigma percent error was normally reported. Therefore the MDA would equal to 1.645 × (measured activity) × (2-sigma percent error) ÷ (100).

Figure C-30 is an example of the lung count report from ABACOS-Plus after mid-February 1997, when the reporting protocol was changed. The primary change was that the activities of ^{235}U, ^{238}U, and ^{241}Am are calculated and reported, even if the peak was not detected or if the result was negative. The MDA values are for the average worker, as stated on the report. The MDA value for ^{238}U is lower

than the worker-specific decision level for this case. The worker-specific MDA should be at least twice the worker-specific decision level.

Figure C-31 is an example of the lung count report from ABACOS-Plus for a worker with a confirmed deposition. The software calculated the deposition for the plutonium isotopes based on the intake date in the header and on the calculated ppm ^{241}Am (including ingrowth), which was based on the value of the initial ppm ^{241}Am in the worker's file. The % Error for ^{241}Am was assigned to the plutonium isotopes. The value of the ppm ^{241}Am on the date of the count was not reported on lung count reports that were generated by ABACOS-Plus. This value can be calculated using Equation B-18 in Attachment B.

Much of the information from ABACOS-Plus is not useful, including Count Rate, Detector Count Rate, Analysis Limits, and the total activity.

Dose reconstructors should note the intake date. If the intake date is different from the date for Count Started, the intake date is from the file for a worker with a confirmed deposition.

Dose reconstructors should be aware that the lung counter detectors were also used for wound counts (Berger 1988b; KHC 2000b, p. 93). Reports of wound measurements, including the calibration of the detector using americium and plutonium sources, look the same as the lung count reports except for some header information (name, employer, job code, reason, height, or weight).

It is important to note that the calculated activities for plutonium for lung counts were based on a specific, actual, or assumed intake date and initial ppm ^{241}Am. The plutonium values are valid and appropriate only for that intake data. If dose reconstructors choose to use another intake date or initial ppm ^{241}Am, they should recalculate the set of plutonium lung deposition activities based on the recalculated ppm ^{241}Am for ingrowth. This is accomplished by multiplying the original activity of plutonium by the ratio of the original ppm ^{241}Am on the date of the count divided by the new value of the ppm ^{241}Am on the date of the count. The new value of the ppm ^{241}Am on the date of the count can be calculated using Equation B-18 in Attachment B. Dose reconstructors should adjust the activities for the discontinuity factors presented in Attachment B. In general, use of the discontinuity factors is favorable to claimants. The only exception to this statement is the CWT adjustment factor (Equation B-4 in Attachment B) for low indices. For indices less than 0.98, the CWT adjustment is less than 1.00.

5.7 ATTRIBUTIONS AND ANNOTATIONS

All information requiring identification was addressed via references integrated into the reference section of this document.

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GLOSSARY

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

alpha particles
See *alpha radiation*.

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.

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.

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

curie
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.

detection limit (lower)
See *limit of detection*.

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.

dosimetry
Measurement and calculation of internal and external radiation doses.

element
One of the known chemical substances in which the atoms have the same number of protons. Elements cannot be broken down further without changing their chemical properties. Chemical symbols for the elements consist of either a single letter or a combination of letters, some of which descend from the Latin names [e.g., Au from *aurum* (gold), Fe from *ferrum* (iron)]. This glossary indicates *elements* by their names. Specific *isotopes* appear as their standard chemical symbols with the number of protons and neutrons in the nucleus. For example, the isotope of uranium that contains 92 protons and 143 neutrons can appear as ^{235}U, *U-235*, or *uranium-235*. See *periodic table of the elements* and *radioactive isotope*.

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.

extremities

Portion of the arm from and including the elbow through the fingertips and the portion of the leg from and including the knee and patella through the toes.

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.

gamma ray

See gamma radiation.

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*, *gamma radiation*, *neutron radiation*, *photon 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. See *element*.

limit of detection

Minimum level at which a particular device can detect and quantify exposure or radiation. Also called lower limit of detection and detection limit or level.

maximum permissible lung burden (MPLB)

Historical occupational limit on the amount of a radionuclide present in the systemic body at the end of 50 years as a result of being exposed at the maximum permissible concentration for 50 working years.

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).

neutron (n)

Basic nucleic particle that is electrically neutral with mass slightly greater than that of a proton. There are neutrons in the nuclei of every atom heavier than normal hydrogen. See *element*.

neutron radiation

Radiation that consists of free neutrons unattached to other subatomic particles emitted from a decaying radionuclide. Neutron radiation can cause further fission in fissionable material such as the chain reactions in nuclear reactors, and nonradioactive nuclides can become radioactive by absorbing free neutrons. See *neutron*.

nuclide

Stable or unstable isotope of any element. Nuclide relates to the atomic mass, which is the sum of the number of protons and neutrons in the nucleus of an atom. A radionuclide is an unstable nuclide.

periodic table of the elements

Arrangement of the chemical elements in order of increasing atomic number from left to right and by similar chemical properties vertically. Elements of similar properties occur one under the other, which yields groups or families of elements.

photon

Quantum of electromagnetic energy generally regarded as a discrete particle having zero rest mass, no electric charge, and an indefinitely long lifetime. The entire range of electromagnetic radiation that extends in frequency from 10^{23} cycles per second (hertz) to 0 hertz.

photon radiation

Electromagnetic radiation that consists of quanta of energy (photons) from radiofrequency waves to gamma rays.

radioactive isotope

Natural or synthetic form of an atom that emits radioactivity when it decays. See *isotope*.

radiation

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

radioactive

Of, caused by, or exhibiting radioactivity.

radioactivity

Property possessed by some elements (e.g., uranium) or isotopes (e.g., ^{14}C) of spontaneously emitting energetic particles (electrons or alpha particles) by the disintegration of their atomic nuclei.

radionuclide

Radioactive nuclide. See *radioactive* and *nuclide*.

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.

site returns

At Rocky Flats, weapons components returned from other sites for disassembly and recovery of materials.

whole-body dose

Dose to the entire body excluding the contents of the gastrointestinal tract, urinary bladder, and gall bladder and commonly defined as the absorbed dose at a tissue depth of 10 millimeters (1,000 milligrams per square centimeter). Also called penetrating dose. See *dose*.

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
MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS

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MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

A.1 INTRODUCTION

Urinalysis was used at RFP since the start of operations in 1952 to detect intakes of radionuclides by workers who were exposed, or had the potential to be exposed, to plutonium, EU, or DU. Urinalysis involved the submission of a urine sample by the worker, a chemical processing of the sample to isolate the radionuclide of interest (the analyte), and measurement and calculation of the mass or activity of the analyte in the sample. The request for submission of the urine was either scheduled as part of a routine monitoring program or was specially requested after an actual or suspected intake. Routine urine samples were typically 24-hour excretions, either one continuous 24-hour period (but not taken at the RFP site) or two 12-hour periods. Special urine samples could be 24-hour samples, overnight samples, or a single voiding. The chemical processing of the sample depended on the analyte and the need for specificity and recovery. "Specificity" refers to separation of the desired radionuclide from interferences such as other radionuclides. "Recovery" refers to isolating as much of the analyte as possible in the final medium to be measured (counted). The measurement of the sample typically involved counting the alpha radiation from the processed aliquant of the sample and determining the activity of the analyte in the original sample. Also involved was the fluorometric measurement of mass of DU. The assessment of the MDA involves the determination of the activity of the analyte in the original urine sample that would be expected to be detected by the methods and systems used at RFP. The analytes of interest are plutonium, americium, EU, and DU. In addition, RFP analyzed for gross alpha using a nonspecific analysis for workers from 1952 to 1971 who were potentially exposed to any of the analytes of interest. This attachment focuses on the period from 1952 to 1971, for which many of the urinalysis logs have been found and analyzed to obtain the information necessary to assess the MDA. This also is the period when urinalysis procedures were primitive and evolving and numerous dosimetrically interesting events and intakes were occurring at RFP.

A.2 MDA METHODOLOGY

The general equation for the MDA is Equation 6 in the American National Standard, *Performance Criteria for Radiobioassay* (HPS 1996):

$$MDA = \frac{(1+\Delta_K)(2\Delta_B B + 2ks_0 + 3)}{KT} \quad \text{(A-1)}$$

where

- B = the total count of the appropriate blank
- Δ_B = the maximum expected fractional systematic error bound in the appropriate blank
- K = calibration factor
- Δ_K = the maximum fractional systematic error bound in the calibration factor K
- k = the abscissa of the standardized normal distribution corresponding to the 0.05 probability level (for α = 0.05 and β = 0.05, k = 1.645)
- T = the standard counting time for the procedure
- s_0 = the standard deviation in the net count of a sample with no additional analyte:

$$s_0 = \sqrt{s_{B1}^2 + \frac{s_{B0}^2}{m^2}}$$

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MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

where

s_{B1} = the standard deviation of the sample, where the sample contains no actual analyte above that of the appropriate blank
s_{B0} = the standard deviation in the unadjusted count of the appropriate blank
m = the adjustment factor for the appropriate blank

Applying this equation to urinalysis methods at RFP involves determining the value of each variable for measurements of the analytes (plutonium, americium, EU, DU, and gross alpha) as the methods evolved.

A.3 HISTORY OF METHODS

General Information

In the beginning of operations (1952), RFP was divided into four distinct subplants plus a general support area. The subplants were named A Plant, B Plant, C Plant, and D Plant. The designations A, B, C, and D are significant because they are also the code names for the materials that were processed in those plants as well for the urinalysis procedures that were used to analyze those materials. The records of the 1950s do not contain the words “depleted uranium,” “enriched uranium,” and “plutonium.” Instead, DU is A material processed in A Plant (buildings numbered 4##, mainly Building 444); EU is B material processed in B Plant (buildings numbered 8##, mainly Building 881); and plutonium is C material processed in C Plant (buildings numbered 7##, mainly Building 771). D Plant (buildings numbered 9##, mainly Building 991) handled all materials. D Plant (Building 991) handled “all materials” as a consequence of its function of shipping, receiving, and storage of special nuclear and classified materials for RFP, as well as final assembly and inspection of plutonium and EU products in the early years. For more information, see “Historical American Engineering Record, Rocky Flats Site, Building 991” (DOE 2011a). A nonspecific gross alpha urinalysis method was used for workers in D Plant. [Note: Building numbers were two-digit numbers until 1968, when the numbers were expanded to three digits (e.g., Building 771 was originally Building 71).] From 1962 to 1963, the EU operations were phased out at RFP, although urinalysis monitoring for EU continued through 1971.

The Urinalysis Record Card is an important and significant record for the early (1952 to 1969) urine data and for the methods that generated those data for a specific worker. A Urinalysis Record Card was established for each monitored worker and included the result of each urine sample, the date of the sample, and the code of the urinalysis method that was used to generate that result are recorded. The card is now in the worker’s Health Physics file, which is the primary RFP record of dosimetry information for a worker. Table A-1 lists the method codes. The method codes are listed at the top of the Urinalysis Record Card (see Figures C-1 to C-3 in Attachment C). Units, if not listed on the card, were discerned from the urine data logs.

Although there is some correlation of the codes with the subplants, there are some exceptions. Table A-2 summarizes the correlation of the method code and the analyte (see Figures C-1 to C-3 in Attachment C).

Tolerance levels were used at RFP in the 1950s and 1960s as an indicator of the maximum permissible amount (activity) of a radionuclide excreted per day in a worker’s urine. The technical basis for the values of tolerance levels has not been identified. The significance is that urinalysis results less than 10% of the tolerance level were recorded and reported as background (BK on the Urinalysis Record Card) or zero, regardless of the underlying sensitivity of the method, with some

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MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

Table A-1. Method codes.

Code	Meaning
A	Fluorimeter, reported in µg/L 1952–1956 and µg/24 hr 1957–1964
B_1	Electroplating, reported in dpm/24 hr. (Note: Electroplating, in RFP records, more properly should be called electrodeposition.)
B_2	Ether extraction, reported in dpm/24 hr
B_3	TBP extraction (hand-written on some cards)
C_1	Carrier precipitation, reported in dpm/24 hr
C_2	TTA extraction, reported in dpm/24 hr. (Note: On the header of cards for 1961–1965, the code C_2 is "Pu by Radio Autography." There is no indication that this method was implemented at RFP.)
D	TBP extraction

Table A-2. Correlation of method code and analyte.

Analyte	Method code
DU	A, B_1 (starting 05/01/64)
EU	B_1
Plutonium	C_1, C_2
Gross alpha	B_2, B_3, D

exceptions. Table A-3 lists the values of the tolerance levels. The tolerance levels were noted as the "working MDL" (minimum detection level) in some early urine data logs. The reporting levels were not stated explicitly in the data logs, but rather were discerned from the minimum values calculated in the data logs. These minimum values corresponded to 10% of the working MDL.

Table A-3. Values of tolerance and reporting levels.

Analyte	Tolerance level	Reporting level
DU	58 µg/24 hr	≥5.8 µg/24 hr
EU	88 dpm/24 hr	≥8.8 dpm/24 hr
Plutonium	8.8 dpm/24 hr	≥0.88 dpm/24 hr
Gross alpha	88 dpm/24 hr	≥8.8 dpm/24 hr

These reporting (and recording) levels continued through April 1964 for both DU and EU, through 1961 for plutonium, and through 1963 for gross alpha. From May 1964 through 1971, the reporting level for DU and EU was ≥20 to 28 dpm/24 hr. After 1963, the reporting level for gross alpha was ≥0.9 dpm/24 hr. This change in the reporting level for the gross alpha results corresponded to the change to using plutonium as the default analyte rather than EU. The other changes in this paragraph were discerned from the lowest values recorded in the urine data logs.

For plutonium, the reporting and recording level was ≥0.2 dpm/24 hr for 1962 to April 6, 1970 based on an examination of the CEDR database. After that date, all results ≥0.00 dpm/24 hr were recorded and reported. Negative values were recorded and reported as 0.00 dpm/24 hr. A further exception is that, for some workers, the practice implemented on April 7, 1970, was applied retroactively for their plutonium data. This retroactive application was variable depending on how far back it was applied.

In 1963, a specific analysis for ^{241}Am was implemented. The recording and reporting level for ^{241}Am was ≥0.24 dpm/24 hr in 1963, ≥0.2 dpm/24 hr from 1964 to 1967, and ≥0.3 dpm/24 hr from 1968 to 1971 (ORAUT 2003).

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MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

The general method for data analysis for alpha-counting procedures (1952 to 1971) was:

$$Activity\left(\text{dpm/24-hr sample}\right) = \frac{\left(C/T - B_{\text{Det}} - B_{\text{Blk}}\right)\left(V/A\right)}{\varepsilon R} \qquad \text{(A-2)}$$

where

- C = total count
- T = count time (min)
- B_{Det} = detector background count rate (cpm)
- B_{Blk} = reagent blank count rate (cpm)
- V = sample (or standard) volume (mL)
- A = volume of the aliquant analyzed (or volume of the sample, if the entire sample was analyzed) (mL)
- ε = efficiency (geometry) of the detector (cpm/dpm)
- R = recovery, fraction of the analyte in the aliquant or sample that is transferred to the planchet or disk to be counted

The detector background count rate was generally tabulated in the urinalysis data logs through 1961. After 1961, the value for the detector background is implicit in the data reduction but is not explicitly recorded. The same detectors were used for alpha counting for all analytes.

Reagent blanks were generally processed with each batch of samples, and the value of the blank count rate that was used in the data reduction was generally tabulated in the urinalysis data logs.

The ratio V/A is a volume adjustment factor that was used for two purposes. If the entire sample was not analyzed, this ratio normalized the result from the volume of the analyzed aliquant to the total sample. If the volume of the total sample was less than a minimum specified volume (e.g., 1,000 mL), the sample was considered to be less than a 24-hr sample, and the ratio was used to normalize the sample result to that for a 24-hr sample. This is a description of the general method. How and when the volume adjustments were made for each analyte and period are discussed later in the document. The sample volume was recorded in the urinalysis data log for each sample.

The value of ε was the geometry rating of the detector. In 1952 and 1953, ε was 0.45. After that, the detectors were called 50% detectors, and ε was 0.50. In 1964, 40% detectors (ε = 0.40) were added to the system as a supplement to the 50% detectors.

The value of R was generally a standard value. Depending on the process, spiked samples (samples to which a known activity of the analyte was added) were generally processed with each batch of samples. The recovery values that were calculated from the spiked samples were the ratios of the count rate of spiked sample to the average count rate of four to six samples deposited on the planchet or plate with minimal processing. The recovery values for the spiked samples were not normalized to the deposited activity (dpm).

The fraction of absorption of the alpha particles in the residue on the planchet or plate was not explicitly incorporated either in the efficiency or recovery.

The term εR was frequently combined, especially in the 1950s. In the 1960s, the term $1/\varepsilon R$ was occasionally tabulated in the urinalysis data logs as “R.F.” (presumably for “recovery factor”), and was used as a multiplier to convert the net count per minute to activity in the sample.

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MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

The general method for the mass measurements of uranium using the fluorimeter (1953 to 1964) was:

$$Mass(\text{µg/24-hr sample}) = \frac{S - B_{Blk}}{K} \quad \text{(A-3)}$$

where

S = signal reading of the sample aliquant
B_{Blk} = signal reading of the blank
K = *constant*/V; *Constant* is custom to each process; V = volume (mL) of the entire urine sample. If the sample volume ≤1,000 mL, V = 1,000 mL.

The history of these urinalysis methods is largely based on an interview with the [position redacted] from [date redacted] to [date redacted], [name redacted], in 1992 and on a review of the bioassay data logs from 1952 to 1971.

Plutonium:

- 1952 to 1961. The urine sample was processed using a method called carrier precipitation (also called coprecipitation). The plutonium in the urine sample (plus some americium and thorium) was carried into the precipitate with lanthanum fluoride. The precipitate was dissolved and the solution was evaporated on a planchet, which was counted with a gas-flow proportional counter. Typical count time was 150 minutes. A spike sample and a reagent blank sample were processed with the worker samples, sometimes with each batch and sometimes less frequently. The result of the spike sample might have been used to establish the value of the recovery of the analyte for the batch. Similarly, the result of the blank (counts per minute) might have been used to establish the value of the blank subtracted from the total count rate of the sample. Detector efficiency was stated to be 0.50. A volume adjustment factor (1,200/*sample volume*) was applied as a multiplier to the result if the sample volume was less than 1,000 mL. The first evidence of the use of this factor is in 1960 (ORAUT 2003).

- 1961 to 1962. Starting on December 13, 1961, a thenoyltrifluoroacetone (TTA) extraction step was added to the carrier precipitation method to improve the specificity of the process to isolate plutonium (ORAUT 2003). No other changes were made to the previous method.

- 1963 to 1978. The ion exchange method replaced the carrier precipitation/TTA extraction method in 1963 and was used, with refinements, thereafter. The method was specific to plutonium. In addition, americium could be recovered separately from the plutonium in the same sample. Evaporation of the analyte on a planchet was continued, but that method was gradually phased out and replaced by electrodeposition on a stainless-steel disk. About one-third of the samples were electrodeposited in 1964 and one-half or more from 1967 to 1971. In 1973, an alpha PHA counting system with surface barrier detectors was started with four detectors. The practice of using internal tracers (^{236}Pu or ^{242}Pu) for some plutonium samples was begun concurrently. A batch blank continued to be processed, although its use was inconsistent. For example, in 1971, a blank count rate of 0.00 cpm was used even though the median value of the batch blank was 0.06 cpm. In 1964, detectors with an efficiency of 0.4 were used as a supplement to the detectors with 0.5 efficiency (ORAUT 2003).

- 1978 to 1993. By 1978, all counting systems had been converted to the PHA system, and all plutonium samples were processed with internal tracers. The recovered fraction of the internal

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MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

tracer for that sample was applied in the analysis of the result for that sample. The acceptable range of the fractional tracer recovery was 0.10 to 1.10. The result of a sample was invalidated if the recovery was outside the acceptable range (ORAUT 2003). In 1990, the acceptable recovery range was changed to 0.35 to 1.10 (ORAUT 2003). The count time of 720 minutes was used for all samples. A batch blank continued to be processed and generally was used in the data analysis unless suspected to have been contaminated excessively (a subjective decision). In 1985, the blank method was modified. The value of the blank that was used in the analysis of the result for a sample was the average value of the last 20 valid batch blanks. To be valid, a batch blank value was tested using the Dixon outlier test and, if it passed the test, was added to the population of the last 20 blanks. In 1988, the blank process was further modified by use of the Winsorized trimmed mean of the population of 20 blanks instead of the average value. The purpose of these modifications was to minimize the influence of laboratory contamination artifacts, which were considered to be nonrandom events that, if incorporated in the blank, would inappropriately bias the results of the other samples on the low side. In addition, the reagent blank was replaced by a matrix blank, either real or artificial urine. The volume of the analyzed sample (aliquant) was 800 mL if the volume of the sample was greater than 800 mL or, if the volume of the sample was less than 800 mL, the entire sample. The result of the aliquant was divided by the volume fraction (800 mL/volume of the sample) if the volume of the sample was ≥800 mL. The efficiency of the detectors was typically in the range of 0.25 to 0.35.

- 1993 and After. Upgrades to procedures occurred in 1993 to achieve a process MDA less than or equal to 0.020 dpm/sample (ORAUT 2003). Count time was increased to 1,400 minutes. The entire sample was analyzed so that the volume fraction was unity for all samples. In addition, a contract was established with a commercial bioassay laboratory, with a requirement that an MDA of ≤0.02 dpm/sample be achieved. In 1997, the onsite bioassay laboratory was shut down.

Americium (1963 and After)
Except for the details of the chemistry, the process for americium was similar to that for plutonium. A solvent extraction process, specific for americium, was first used in 1963 (ORAUT 2003). A new process (not defined in the data log) was started in November 1965. At some point, not defined in the examined data logs, the ion exchange method was implemented for americium.

Enriched Uranium (1952 to 1971)
Urine samples were analyzed for EU according to a process called electroplating. A 50-mL aliquant of urine was extracted from the 24-hour sample and chemically processed to minimize impurities. The resulting solution was poured into an electrodeposition column, and the uranium was deposited on a stainless-steel disk. The disk then was counted for alpha radiation with the gas-flow proportional counters, as described for plutonium. Counting times in this period were 30, 40, 60, 70, 90, 120, and 150 minutes.

From 1952 to 1955, one aliquant per sample was used. In 1960, a second aliquant was processed if the result of the first aliquant was ≥7 dpm/24-hr sample. If the second result was within a specified range of the first result, the average of the two results was recorded and reported. If the second result was out of the specified range, a third aliquant was processed, and the average of the two results that best confirmed each other was used. If that average was less than the reporting level of 8.8 dpm/24 hr, the result was recorded and reported as background. From 1961 to 1971, two aliquants were routinely processed for each urine sample, with a third aliquant (1961 to 1969) processed if the spread of the results of the first set was outside the specified range. The recording

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MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

and reporting logic was the same as that for 1960. From 1964 to 1971, the recording and reporting limit appears to have been ≥20 to 28 dpm/24-hr sample, depending on the volume of the sample (ORAUT 2003).

Blank data were not used to adjust the sample count rate except sporadically in 1963 and 1964. Detector background was usually subtracted, but not always. Spike samples were processed, although it is not obvious how those data were used, if at all. Instead, a constant value of the product of the detector efficiency ε and the recovery *R* was used: 0.40 (1953 to 1955 and 1971), 0.30 (1960 to 1970), and 0.24 (1964 to 1970 for detectors with ε = 0.40) (ORAUT 2003).

Depleted Uranium (1952 to 1971)
Two methods were used to analyze urine samples for DU. From 1952 to April 1964, a fluorimeter was used to measure the mass (micrograms) of uranium in a 100λ (0.1-mL) aliquant of the 24-hour urine sample. The result was extrapolated to the total sample and reported in the unit of µg/24-hr sample. A volume adjustment was made if the sample volume was less than 1,000 mL. If less than 1,000 mL, the volume was set equal to 1,000 mL.

Screening was done with one aliquant. A second aliquant was processed if the net reading of the first aliquant was greater than or equal to a value in a chart that correlated with the volume of the 24-hr urine sample. A third aliquant was processed if the results (net readings) of the first two aliquants varied by 20% or more. The average result of the two aliquants that agreed was converted to µg/24-hr sample and reported only if the result was greater than or equal to the reporting level of 5.8 µg/24-hr sample. Otherwise, the result was reported as background.

After April 1964, the urine sample was analyzed using the electroplating procedure described above for EU, and the results were reported in dpm/24-hr sample (or background) (ORAUT 2003).

Gross Alpha (1952 to 1971)
Two methods were used to analyze urine samples for gross alpha counts from either plutonium or uranium. The ether extraction method was used from 1952 to December 12, 1962, and the tributyl phosphate (TBP) extraction method was used from December 12, 1962, to 1964. The TBP method was replaced by the TOPO method. All methods were nonspecific in extracting plutonium and uranium as well as americium and natural thorium (ORAUT 2003).

In all methods, the entire urine sample was processed, and the final extract was evaporated on a planchet and counted on the gas-flow proportional counter. Counting time was typically 150 minutes, although from 1952 to 1955 count times of 55, 60, and 75 minutes, and in 1971 40 and 60 minutes, were also used.

Samples with results ≥0.9 dpm/24-hr sample were typically, but not always, counted using a PHA system to determine whether to credit the result to EU or to plutonium, or a portion to each. The default assumption through 1963 was to credit the result to EU unless the PHA count indicated otherwise. After 1963 (when EU operations were phased out), the default assumption was to credit the result to plutonium.

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MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

A.4 ASSESSMENT OF MDA

General Considerations
The MDA is assessed for plutonium, americium, EU, DU, and gross alpha, based on Equation A-1 and the values of parameters for the methods. Some considerations are:

- The probabilities of Type I (false positive) and Type II errors (false negative) are each 5% ($\alpha = \beta = 0.05$).
- The MDA is assessed for the typical, average, or median condition. If appropriate, the MDA is also assessed for the 5th- or 95th-percentile conditions.
- The MDA is assessed for the methods as they should have been performed, with consideration of such factors as alpha transmission factor, blank subtraction, recovery fraction, and volume adjustment.
- For methods with two or more options in the same period (e.g., evaporation vs. electrodeposition, 40% detectors vs. 50% detectors), the option that gives the higher MDA is used.

The value of the MDA for the typical, average, or median condition pertains to the process and indicates the amount or activity in the population of urine samples that would have been detected with a 95% probability, given a properly set decision criterion that allows a 5% probability of a Type I error. In reality, the decision criterion (and method) at RFP was not based on the probability of a Type I error. Instead, an arbitrary level (10% of the tolerance level or any nonnegative value) was used as the decision criterion for recording and reporting detected amounts or activities.

The value of the MDA for the 5th- or 95th- percentile conditions pertains to individual samples for which the conditions of the sample (e.g., low volume) or conditions of the processing (low recovery, high blank, high alpha self-absorption) were marginal. The conditions of low recovery, low volume, and high alpha self-absorption are associated with the calibration factor K and can be incorporated either in the value of K or in the Δ value of Δ_K.

Table A-4 lists sample volumes for routine 24-hour urine samples.

Table A-4. Sample volumes (mL) for routine 24-hour urine samples.

Percentile	Volume
5th	700
Median	1,350
95th	1,750

The values for the parameter values for the processes were obtained through review of the urine data logs for the periods from 1952 to 1955 and 1960 to 1971. For some years in these periods, logs for only a part of the year were available.

Data for Alpha-Counting Systems
Table A-5 lists the detector background (cpm) for the gas flow proportional counters, based on tabulations in the urine data logs from 1952 to 1955 and from 1960 to 1963, for a sample count time

ATTACHMENT A
MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

of 150 minutes. To have a coherent dataset, only background count data for samples counted for 150 minutes were extracted from the urine data logs.

Table A-5. Detector background (cpm) for gas flow proportional counters.

Date	Average	5th percentile	Median	95th percentile
1950s	0.060±0.022	0.02	0.06	0.10
1960s	0.054±0.014	0.03	0.05	0.08
Composite	0.056±0.017	0.03	0.05	0.08

No documentation was found about the count time that was used to measure the detector background, but the count time was probably 150 minutes or longer. For the purpose of assessing the MDA, the composite average is used for the value of the detector background count rate B_{Det} = 0.056 cpm with the standard deviation s_{Det} = 0.017 cpm for all alpha-counting methods (except for americium) and for all sample count times. The composite value was used because the detector background appeared to be reasonably stable in the 1950s and 1960s, as observed in the previous table. For americium, the values for the 1960s are used because the americium process was not implemented in the 1950s. This exception was made for americium because the detector backgrounds for the 1950s did not apply.

The blank count rate was method-specific, and the application of the blank in the data analysis was variable between methods and within a method over time. A complication that was intermittent, but persistent, was laboratory contamination artifacts that were introduced into blanks and worker samples. These artifacts caused false positives from a worker exposure viewpoint but real positives from a detection viewpoint.

For the purpose of this MDA analysis, the median value of the blank is used to determine the process MDA and the 95th-percentile (low to high) value is used to determine the MDA for the more extreme conditions. Table A-6 summarizes the median and 95th-percentile blank count rates.

Table A-6. Median and 95th-percentile blank count rates (cpm).

Analyte	Period	Median	95th percentile
Plutonium	1952–1971	0.06	0.28
EU	1952–1971	0.05	0.22
DU	1964–1971	0.05	0.22
Americium	1963–1971	0.07	0.26
Gross alpha	1952–1971	0.08	0.30

These values are the average of the yearly values extracted from available urine data logs (as reviewed by R. Falk in 2003 (the initial author of this TBD; see Section 5.7). For each of the analytes, the yearly median and 95th-percentile values did not differ enough over the period to warrant a separate MDA analysis. The blank values for EU and DU are based on log entries in 1963 and 1964 for cell blank checks for the electrodeposition process.

The value of the blank count rate B_{Blk} is taken from Table A-6 for the given analyte. The standard deviation s_{Blk} is taken to be the square root of the blank count for the process divided by the count time of the process:

$$s_{Blk} = \frac{\sqrt{B_{Blk}T}}{T} \qquad \text{(A-4)}$$

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The values for B, s_{B0}, s_{B1}, and s_0 in the MDA equation (A-1) are derived from the detector background and blank values:

$$B = T\left(B_{Det} + B_{Blk}\right) \quad \text{(A-5)}$$

$$s_{B0} = T\sqrt{s_{Det}^2 + s_{Blk}^2} \quad \text{(A-6)}$$

$$s_{B1} = \sqrt{B} \quad \text{(A-7)}$$

$$s_0 = \sqrt{s_{B1}^2 + s_{B0}^2} \quad \text{(A-8)}$$

For some analytes (EU, DU) and periods, the detection decision was based on the average of two aliquants. In this case, the value of s_0 for the average of two aliquants is equal to the value of s_0 for one aliquant divided by the square root of 2.

The value of Δ_B is taken to be zero. This variable could be used to account for high blank values. Instead, the effect of high blank values is determined by using the 95th-percentile value of the blank.

The calibration factor K is a combination of the detector efficiency ε, the recovery R, and the volume adjustment factor (V/A). Also included is a factor that accounts for absorption of alpha particles in the residue of planchets or plates.

Common detectors were used for all alpha-counting methods. Table A-7 lists the efficiencies of the detectors (as noted above):

Table A-7. Efficiencies of alpha-counting detectors.

Period	Detector efficiency
1952–1953	0.45
1954–1963	0.50
1964–1971	0.40 and 0.50

For 1964 to 1971, the value of 0.40 is used as the efficiency for the MDA calculation. This approach is consistent with this section under the General Considerations subsection.

Table A-8 lists the recoveries that were used in the MDA assessment, which are taken to be the median recovery and the 5th-percentile (low to high) value discerned from the spike data for the process.

Table A-8. Recovery fractions used in MDA assessments.

Analyte	Period	Median	5th percentile
Plutonium	1952–1962	0.57	0.25
Plutonium	1963–1971	0.67	0.28
EU	1952–1971	0.60	0.21
DU	1964–1971	0.60	0.21
Americium	1963–1965	0.67	0.29
Americium	1965–1971	0.80	0.26
Gross alpha	1952–1971	0.57	0.24

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The recovery values are based on incomplete datasets and involve extrapolations to cover the total period. For plutonium from 1952 to 1962, the values are based on data for 1961 and 1962. For plutonium from 1963 to 1971, the values are based on data for 1963 to 1965 and 1969 to 1971. For EU and DU, recoveries were not calculated for the spiked samples. The median value is based on the value that was used for most of the period. The 5th-percentile value is based on the relative standard deviation (0.40) of the average count rate of the spiked samples from 1963 to 1966. For americium from 1963 to November 1, 1965, the values are based on a complete set for that period.

For 1965 to 1971, the values are based on data from November 1, 1965, to 1966, and 1968 to 1970. For gross alpha, the values are based on data from 1962 to 1969 for the TBP method.

The volume adjustment factor V/A is incorporated into the calibration factor K as the reciprocal $1/(V/A)$, s_0 it becomes a multiplier with the efficiency and recovery. For convenience, the reciprocal of the volume adjustment factor is designated V_f.

For plutonium, americium, and gross alpha, the median condition is $V = A$ and $V_f = 1$. The extreme condition is a low sample volume normalized to 1,200 mL: V = 1,200 mL, A = 700 mL (the 5th-percentile volume), and V_f = 0.58 (ORAUT 2003).

For EU and DU (for the electrodeposition process), A = 50 mL, the median V = 1,350 mL, and V_f = 0.037. The extreme condition is a high sample volume: V = 1,750 mL (the 95th-percentile volume), A = 50 mL, and V_f = 0.029 (ORAUT 2003).

The absorption of the alpha particles in the residue that was evaporated on the planchets or electrodeposited on the plates should be incorporated into the value of the calibration factor. The factor to incorporate this effect is the fraction of the alphas that are emitted by the deposited analyte that successfully escape from the residue. Let this factor be designated F_a, where F_a = (1 – fraction of alphas absorbed in the residue), and let the fraction of alphas absorbed in the residue be f_{abs}. Table A-9 lists the values of f_{abs}, based on judgments of experienced bioassay chemists, for the extreme (95th-percentile) condition, and the corresponding values of F_a.

Table A-9. Fractions of alphas absorbed in residue at the 95th percentile.

Analyte	Period	f_{abs}	F_a
Plutonium (evaporated)	1952–1962	0.4	0.6
Plutonium (evaporated)	1963–1971	0.3	0.7
Plutonium (electrodeposited)	1963–1971	0.05	0.95
EU (electrodeposited)	1952–1971	0.05	0.95
DU (electrodeposited)	1964–1971	0.05	0.95
Americium (evaporated)	1964–1971	0.3	0.7
Americium (electrodeposited)	1964–1971	0.05	0.95
Gross alpha (evaporated)	1952–1962	0.1	0.9
Gross alpha (evaporated)	1962–1971	0.3	0.7

From 1963 to 1971, approximately half of the plutonium and americium samples were electrodeposited. However, the identities of samples that were electrodeposited are not discernible from the databases and reports of urinalysis results that are readily accessible. Electrodeposited plutonium and americium samples were marked in the data logs with an E. No similar designation has been observed by the author in any reports of these urinalysis results. For the purpose of the MDA assessment, dose reconstructors should use the value of F_a for the evaporation process.

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For the median condition, the value of F_a is taken to be 1 under the assumption that the absorption of alphas for the median condition of the planchet or plate was incorporated in the recovery value at the time.

The calibration factor K is the product of ε, R, V_f, and F_a:

$$K = \varepsilon R V_f F_a \quad \text{(A-9)}$$

The values of Δ_B and Δ_K are considered to be zero. No evidence of a systematic bias in the background or the calibration factor was discerned by the author. Therefore, Δ_B and Δ_K were set equal to zero.

Data for Fluorimetric Mass Measurements

Applying the MDA equation (A-1) to fluorimetric mass measurements involves setting the value of T to unity and eliminating the term "3".

The value of s_{B0} is the standard deviation of the blank flux readings that are subtracted for the signal of the aliquant reading. The value of s_{B1} is set equal to s_{B0}, and s_0 is equal to the value of s_{B0} multiplied by the square root of 2:

$$s_0 = s_{B0}\sqrt{2} \quad \text{(A-10)}$$

The value of s_{B0} was determined from a review by R. Falk of the urine data logs for 1955 and 1960 to 1962. One discontinuity was noted on September 14, 1955. The value of s_{B0} before the discontinuity was 0.37 and, after the discontinuity, averaged 0.15.

The calibration factor K converts the fluorimeter net reading to the µg U/24-hr sample (see Equation A-3). In 1955, the calibration factor was applied to the uncorrected net reading. In 1960 and later, the calibration factor was applied to the corrected reading, which was the net reading multiplied by the factor 1.15 (ORAUT 2003). The factor 1.15 is incorporated into the value of K starting in 1960. For the 1950s, the calibration factor for 1955 is used, as listed in Table A-10.

Table A-10. Gross alpha calibration factor.

Period	*K*
1952–1959	75/*V*
1960–1964	87/*V*

For the median condition, the volume V is equal to 1,350 mL. For the extreme condition, the 95th-percentile volume of 1,750 mL is used.

The values of Δ_B and Δ_K are considered to be zero.

A.5 MDA VALUES

The value of the MDA is presented to two significant figures for information purposes. In most cases, the value of the MDA should be considered only to one significant figure.

ATTACHMENT A
MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

Plutonium

The MDA for plutonium is assessed for the median condition and for the extreme (5th- or 95th-percentile) condition for the blank, the recovery, the volume factor V_f, and the alpha transmission factor F_a, individually and in combination. A count time of 150 minutes is used for all assessments.

Table A-11 lists the values of the variables and the median MDA (dpm/24-hr sample).

Table A-11. Values of variables and MDA for plutonium for median conditions.

Period	s_0	ε	R	V_f	F_a	MDA (dpm/24-hr sample)
1952–1953	5.74	0.45	0.57	1.0	1.0	0.57
1954–1962	5.74	0.50	0.57	1.0	1.0	0.51
1963	5.74	0.50	0.67	1.0	1.0	0.44
1964–1971	5.74	0.40	0.67	1.0	1.0	0.54

Table A-12 lists the values of the variables for the extreme (5th- or 95th-percentile) conditions and the resulting MDA (dpm/24-hr sample) for all of the extreme conditions occurring for the same sample.

Table A-12. Values of variables and MDA for plutonium for extreme conditions.

Period	s_0	ε	R	V_f	F_a	MDA (dpm/24-hr sample)
1952–1953	7.98	0.45	0.25	0.58	0.6	5.0
1954–1962	7.98	0.50	0.25	0.58	0.6	4.5
1963	7.98	0.50	0.28	0.58	0.7	3.4
1964–1971	7.98	0.40	0.28	0.58	0.7	4.3

The value of s_0 incorporates the 95th-percentile value of the blank.

It is unlikely that the four extreme conditions (high blank, low recovery, low volume, and cruddy residue on the planchet) all occurred for the same sample. Table A-13 lists the MDA for each of the extreme conditions individually, as well as for combinations of two and three extreme conditions.

Table A-13. MDA for plutonium for one, two, or three extreme conditions.

MDA (dpm/24-hr sample) for one extreme condition

Period	s_0	R	V_f	F_a
1952–1953	0.76	1.3	0.98	0.95
1954–1962	0.68	1.2	0.88	0.85
1963	0.58	1.0	0.75	0.62
1964–1971	0.73	1.3	0.94	0.78

MDA (dpm/24-hr sample) for two extreme conditions

Period	s_0, R	s_0, V_f	s_0, F_a	R, V_f	R, F_a	V_f, F_a
1952–1953	1.7	1.3	1.3	2.2	2.2	1.6
1954–1962	1.6	1.2	1.1	2.0	2.0	1.5
1963	1.4	1.0	0.97	1.8	1.5	1.1
1964–1971	1.7	1.3	1.2	2.3	1.9	1.3

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MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

MDA (dpm/24-hr sample) for three extreme conditions

Period	s_0, R, V_f	s_0, R, F_a	s_0, V_f, F_a	R, V_f, F_a
1952–1953	3.0	2.9	2.2	3.7
1954–1962	2.7	2.6	2.0	3.4
1963	2.4	2.0	1.4	2.6
1964–1971	3.0	2.5	1.8	3.2

Uranium

The MDA for EU is assessed for the median condition and for the extreme (5th- or 95th-percentile) condition for the blank, the recovery, the volume factor V_f, and the alpha transmission factor F_a, individually and in combination. A count time of 150 minutes is used for MDA assessments from 1952 to 1963.

For 1964 to 1969, the count time of 30 minutes is used and, for 1970 to 1971, the count time of 40 minutes is used. For 1952 to 1959, the value of s_0 is calculated for one aliquant, and for 1960 to 1971 the value of s_0 is calculated based on the average of two aliquants.

Table A-14 lists the values of the variables and the median MDA (dpm/24-hr sample).

Table A-14. Values of variables and MDA for EU for median conditions.

Period	s_0	ε	R	V_f	F_a	MDA (dpm\24-hr sample)
1952–1953	5.45	0.45	0.60	0.037	1.0	14
1954–1959	5.45	0.50	0.60	0.037	1.0	13
1960–1963	3.85	0.50	0.60	0.037	1.0	9.4
1964–1969	1.57	0.40	0.60	0.037	1.0	31
1970–1971	1.83	0.40	0.60	0.037	1.0	25

Table A-15 lists the values of the variables for the extreme (5th- or 95th-percentile) conditions and the resulting MDA (dpm/24-hr sample) for all of the extreme conditions occurring for the same sample.

Table A-15. Values of variables and MDA for EU for extreme conditions.

Period	s_0	ε	R	V_f	F_a	MDA (dpm\24-hr sample)
1952–1953	6.72	0.45	0.21	0.029	0.95	64
1954–1959	6.72	0.50	0.21	0.029	0.95	58
1960–1963	4.75	0.50	0.21	0.029	0.95	43
1964–1969	2.18	0.40	0.21	0.029	0.95	150
1970–1971	2.48	0.40	0.21	0.029	0.95	120

The value of s_0 incorporates the 95th-percentile value of the blank.

It is unlikely that the four extreme conditions (high blank, low recovery, low volume, and cruddy residue on the planchet) all occurred for the same sample. Table A-16 lists the MDA for each of the extreme conditions individually, as well as for combinations of two and three extreme conditions. Table A-17 lists the values of variables and MDAs for fluorimetric measurements of DU for median and extreme conditions.

ATTACHMENT A
MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

Table A-16. MDA for EU for one, two, or three extreme conditions.

MDA (dpm/24-hr sample) for one extreme condition

Period	s_0	R	V_f	F_a
1952–1953	17	40	18	15
1954–1959	15	36	16	13
1960–1963	11	27	12	9.9
1964–1969	38	88	39	32
1970–1971	31	74	32	27

MDA (dpm/24-hr sample) for two extreme conditions

Period	s_0, R	s_0, V_f	s_0, F_a	R, V_f	R, F_a	V_f, F_a
1952–1953	48	21	18	51	42	19
1954–1959	43	19	16	46	38	17
1960–1963	32	14	12	34	28	13
1964–1969	110	49	40	110	92	41
1970–1971	90	40	33	93	76	34

MDA (dpm/24-hr sample) for three extreme conditions

Period	s_0, R, V_f	s_0, R, F_a	s_0, V_f, F_a	R, V_f, F_a
1952–1953	61	50	23	54
1954–1959	55	45	20	48
1960–1963	41	34	15	43
1964–1969	140	120	51	150
1970–1971	120	94	42	120

Table A-17. Values of variables and MDA for fluorimetric measurements of DU for median and extreme conditions.

Period	s_{B0}	Median K	Extreme K	Median (µg/24-hr sample)	Extreme (µg/24-hr sample)
1952–1955	0.37	0.056	0.043	31	40
1955–1959	0.15	0.056	0.043	12	16
1960–1964	0.15	0.064	0.050	11	14

Americium

The MDA for americium is assessed for the median condition and for the extreme (5th- or 95th-percentile) condition for the blank, the recovery, the volume factor V_f, and the alpha transmission factor F_a, individually and in combination. A count time of 150 minutes is used for assessments from 1963 to 1970. In 1971, the typical (and minimum) count time is 60 minutes.

Table A-18 lists the MDA to two significant figures.

Table A-18. Values of variables and MDA for americium for median conditions.

Period	s_0	ε	R	V_f	F_a	MDA (dpm/24-hr sample)
1963	5.82	0.50	0.67	1.0	1.0	0.44
1964–1965	5.82	0.40	0.67	1.0	1.0	0.55
1965–1970	5.82	0.40	0.80	1.0	1.0	0.46
1971	3.51	0.40	0.80	1.0	1.0	0.76

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MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

Table A-19 lists the values of the variables for the extreme (5th- or 95th-percentile) conditions and the resulting MDA (dpm/24-hr sample) for all of the extreme conditions occurring for the same sample.

Table A-19. Values of variables and MDA for americium for extreme conditions.

Period	s_0	ε	R	V_f	F_a	MDA (dpm/24-hr sample)
1963	9.95	0.50	0.26	0.58	0.7	4.3
1964–1965	9.95	0.40	0.26	0.58	0.7	5.4
1965–1970	9.95	0.40	0.26	0.58	0.7	5.4
1971	5.94	0.40	0.26	0.58	0.7	8.9

The value of s_0 incorporates the 95th-percentile value of the blank.

It is unlikely that the four extreme conditions (high blank, low recovery, low volume, and cruddy residue on the planchet) all occurred for the same sample. Table A-20 lists the MDA for each of the extreme conditions individually, as well as for combinations of two and three extreme conditions.

Table A-20. Values of the MDA for americium for one, two, or three extreme conditions.

MDA (dpm/24-hr sample) for one extreme condition

Period	s_0	R	V_f	F_a
1963	0.68	1.1	0.76	0.63
1964–1965	0.86	1.4	0.95	0.79
1965–1970	0.72	1.4	0.80	0.66
1971	1.2	2.3	1.3	1.1

MDA (dpm/24-hr sample) for two extreme conditions

Period	s_0, R	s_0, V_f	s_0, F_a	R, V_f	R, F_a	V_f, F_a
1963	1.8	1.2	0.98	2.0	1.6	1.1
1964–1965	2.2	1.5	1.2	2.4	2.0	1.4
1965–1970	2.2	1.2	1.0	2.4	2.0	1.1
1971	3.6	2.0	1.7	4.0	3.3	1.9

MDA (dpm/24-hr sample) for three extreme conditions

Period	s_0, R, V_f	s_0, R, F_a	s_0, V_f, F_a	R, V_f, F_a
1963	3.0	2.5	1.7	2.8
1964–1965	3.8	3.2	2.1	3.5
1965–1970	3.8	3.2	1.8	3.5
1971	6.2	5.2	2.9	5.7

Gross Alpha

The MDA for gross alpha measurements is assessed for the median condition and for the extreme (5th- or 95th-percentile) condition for the blank, the recovery, the volume factor V_f, and the alpha transmission factor F_a, individually and in combination. A count time of 55 minutes is used for 1952, 75 minutes for 1953 to 1959, and 150 minutes for 1960 to 1971 for assessments of the MDA for both the median and extreme conditions, except for 1971, when a count time of 40 minutes is also used for the extreme condition. See Table A-21.

ATTACHMENT A
MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

Table A-21. Values of variables and MDA for gross alpha measurements for median conditions.

Period	s_0	ε	R	V_{fee}	F_a	MDA (dpm/24-hr sample)
1952	3.26	0.45	0.57	1.0	1.0	1.0
1953	4.23	0.45	0.57	1.0	1.0	0.88
1954–1959	4.23	0.50	0.57	1.0	1.0	0.79
1960–1963	6.23	0.50	0.57	1.0	1.0	0.55
1964–1971	6.23	0.40	0.57	1.0	1.0	0.69

Table A-22 lists the values of the variables for the extreme (5th- or 95th-percentile) conditions and the resulting MDA (dpm/24-hr sample) for all of the extreme conditions occurring for the same sample.

Table A-22. Values of variables and MDA for gross alpha measurements for extreme conditions.

Period	s_0	ε	R	V_f	F_a	MDA (dpm/24-hr sample)
1952	6.09	0.45	0.24	0.58	0.9	7.4
1953	7.12	0.45	0.24	0.58	0.9	6.2
1954–1959	7.12	0.50	0.24	0.58	0.9	5.6
1960–1962	10.27	0.50	0.24	0.58	0.9	3.9
1963	10.27	0.50	0.24	0.58	0.7	5.0
1964–1971	10.27	0.40	0.24	0.58	0.7	6.3
1971 (T = 40 min)	5.18	0.40	0.24	0.58	0.7	13

The value of s_0 incorporates the 95th-percentile value of the blank.

It is unlikely that the four extreme conditions (high blank, low recovery, low volume, and cruddy residue on the planchet) all occurred for the same sample. Table A-23 lists the MDA for each of the extreme conditions individually, as well as for combinations of two and three extreme conditions.

Table A-23. Values of the MDA (dpm/24-hr sample) for gross alpha measurements for one, two, or three extreme conditions.

One extreme condition

Period	s_0	R	V_f	F_a
1952	1.6	2.5	1.8	1.2
1953	1.4	2.1	1.5	0.98
1954–1959	1.2	1.9	1.4	0.88
1960–1962	0.86	1.3	0.95	0.61
1963	0.86	1.3	0.95	0.79
1964–1971	1.1	1.6	1.2	0.98
1971 (T = 40 min)	2.2	3.4	2.4	2.0

ATTACHMENT A
MINIMUM DETECTABLE AMOUNTS FOR URINALYSIS (continued)

Two extreme conditions

Period	s_0, R	s_0, V_f	s_0, F_a	R, V_f	R, F_a	V_f, F_a
1952	3.9	2.8	1.8	4.3	2.8	2.0
1953	3.3	2.4	1.5	3.6	2.3	1.7
1954–1959	2.9	2.1	1.4	3.2	2.1	1.5
1960–1962	2.0	1.5	0.96	2.3	1.5	1.1
1963	2.0	1.5	1.2	2.3	1.9	1.4
1964–1971	2.6	1.9	1.5	2.8	2.3	1.7
1971 (T = 40 min)	5.2	3.8	3.1	5.8	4.8	3.5

Three extreme conditions

Period	s_0, R, V_f	s_0, R, F_a	s_0, V_f, F_a	R, V_f, F_a
1952	6.7	4.3	3.1	4.7
1953	5.6	3.6	2.6	4.0
1954–1959	5.1	3.3	2.4	3.6
1960–1962	3.5	2.3	1.6	2.5
1963	3.5	2.9	2.1	3.1
1964–1971	4.4	3.6	2.6	4.0
1971 (T = 40 min)	9.0	7.5	5.4	8.3

A.6 DISCUSSION

The MDA is an a priori concept that can be applied a posteriori to a sample under certain circumstances: That the parameter values for the sample (e.g., volume, recovery, detector efficiency, count time) are or can be known before the processing of the sample result, and that the information is used conceptually to determine the subpopulation of conditions of which that sample is a member. Then the a priori MDA value for that subpopulation can be assigned to that sample. The sample volume, the characteristics of the detector that is used to count the sample, and the count time are all known before the analysis of the sample measurement. In theory, but generally not in practice, the recovery could also be known before the analysis of the sample measurement.

The MDA values in this attachment represent overall process MDAs for the median and extreme conditions. However, sufficient information is presented to allow the determination of the MDA for a specific sample if the sample-specific parameter values are known. The sample-specific parameter values, except recovery, are generally recorded in the urine data logs, but not all of the urine data logs have been found and some might not have been archived.

The recoveries for 1952 to 1971 were determined by batch spikes. Not until 1973 were some plutonium samples spiked with an internal tracer (first ^{236}Pu and later ^{242}Pu). All plutonium samples were spiked with an internal tracer after 1978. Experience has shown that a significant variability of recovery can exist within a batch of samples. Therefore, the recovery of a batch spike does not necessarily indicate the recovery of each sample in the batch.

Whether to use the median or extreme value of the MDA or the extreme value depends on the purpose. By definition, the median value implies that half of the samples will have a sample-specific MDA that is lower, and half higher. If the purpose is to define a sample-specific conservative bound, the MDA for the extreme condition should be considered. In general, the recovery fraction was the variable that had the most influence on the sample-specific MDA.

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS

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ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

B.1 INTRODUCTION

In vivo lung counts have been performed at RFP since 1964 to determine the activity of plutonium in the lungs of workers who were exposed, or had the potential to be exposed, to airborne plutonium. The method of in vivo lung counts was to place one or more detectors over the chest of the subject and count the photons that were emitted from the plutonium mixture, if any, in the subject's chest (Boss and Mann 1967). Plutonium was not detected directly because of the low abundance of gamma photons and because of the severe attenuation of the more abundant L X-rays. Instead, the 59.5-keV gamma photon from ^{241}Am was used as a surrogate. Americium-241 was present to some extent in all WG plutonium at RFP. The activity of plutonium was then calculated from the detected ^{241}Am by measuring, calculating, or assuming the fraction of the ^{241}Am in the plutonium mixture on the date of the lung count. At RFP, the fraction of the ^{241}Am in the plutonium mixture has historically been characterized in terms of parts per million by weight. Direct in vivo measurement of plutonium in the lungs, although investigated, was never implemented at RFP. The RFP lung counter detected ^{241}Am. The assessment of the MDA, therefore, is focused on the MDA for ^{241}Am. The MDA for plutonium can then be derived from the ^{241}Am MDA based on the value of the ppm ^{241}Am for the plutonium mixture.

B.2 MDA METHODOLOGY

The general equation for the MDA is Equation 6 in the American National Standard, *Performance Criteria for Radiobioassay* (HPS 1996):

$$MDA = \frac{(1+\Delta_K)(2\Delta_B B + 2ks_0 + 3)}{KT} \quad \text{(B-1)}$$

where

- Δ_K = the maximum fractional systematic error bound in the calibration factor K
- Δ_B = the maximum expected fractional systematic error bound in the appropriate blank
- B = the total count of the appropriate blank
- k = the abscissa of the standardized normal distribution corresponding to the 0.05 probability level (for $\alpha = 0.05$ and $\beta = 0.05$, $k = 1.645$)
- T = the standard subject counting time for the procedure
- K = calibration factor
- s_0 = the standard deviation in the net sample count of a subject with no additional analyte

$$s_o = \sqrt{s_{B1}^2 + \frac{s_{B0}^2}{m^2}} \quad \text{(B-2)}$$

where

- s_{B1} = the standard deviation of the subject, where the subject contains no actual analyte above that of the appropriate blank
- s_{B0} = the standard deviation in the unadjusted count of the appropriate blank
- m = the adjustment factor for the appropriate blank

Applying this equation to in vivo lung counting at RFP involves determining the value of each of these variables for the counting systems and procedures used at RFP as the systems and procedures evolved. The MDA for in vivo measurements is necessarily individual-specific because the detectability of ^{241}Am in the chest is a significant function of the CWT of the subject.

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MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

The MDA can also be determined empirically from replicate measurements on an appropriate blank. This approach is used for the systems starting in 1995 at RFP.

B.3 HISTORY OF COUNTING SYSTEMS AND PROCEDURES

The in vivo lung-counting systems at RFP consisted of photon detectors in a shielded room (6-in.-thick low-background steel lined with layers of lead, tin, and zinc) with electronic equipment (amplifiers and multichannel analyzers) to process and record the data.

There were three counting rooms (Mann 1968, p. 60; 1969, p. 10):

- Room A, built in 1964, operational in 1965;
- Room B, built in 1968, operational in 1969; and
- Room C, built in 1975, operational in 1976.

Each room was equipped with a detector system. When a new detector system was implemented, the previous system was usually maintained as a backup system. As a result, end dates for use of a given detector system are not known. In the era of the germanium detector systems, two or more detector systems could have been operational simultaneously. In that situation, the detector system is identified in the record for each lung count.

1964 to 1968 (Author unknown 2002; Boss and Mann 1967)
There was one counting room. The detector system consisted of two NaI(Tl) scintillation detectors (there was a third detector used for cesium and potassium measurements); each detector was round with a diameter of 4 in. and was 4 mm thick with a surface area of 80 cm^2. These detectors were known as the "4×4 detectors." In most situations, the detectors were configured with one detector above the left portion of the upper chest; the second detector was over the liver and gut region. The chest detector was sometimes placed over the right portion of the upper chest rather than the left position. In other cases, both detectors were placed over the chest. The chest detectors were placed in a framework called a jig to allow a standard and reproducible position for all subjects. Count time was either 40 MLT or 20 MLT. Two backgrounds were used: (1) room background and (2) matched subject background. The room background was the count rate in the empty counting room at the start of the day. The matched subject background was the count rate of an unexposed subject with matched ^{137}Cs and ^{40}K count rates. Calibration was based on ^{241}Am-impregnated epoxy lungs in the chest cavity of a water-filled Radiation-Equivalent Man, Absorption phantom from Alderson Research Laboratories. No adjustment was made for CWT.

1969 to 1976 (Author unknown 2002)
During this period, two counting rooms were operational with three 4- by 4-in. NaI(Tl) scintillation detectors, two over the upper chest (right and left portions) and one over the liver/gut region. The liver/gut detector was eliminated in 1974.

Changes during this period include:

- The ROI of the 59.5-keV photopeak of ^{241}Am was expanded.
- The use of the jig for positioning the detectors was discontinued. Instead, the detectors were positioned in light contact with the surface of the chest.
- The standard count time was changed to 2,000 s (1,000 s for expedited counts).

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MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

- The method of the matched subject background based on ^{137}Cs and ^{40}K was replaced by the index method.

The index method had the following features (Bistline 1968):

- Subjects were characterized by an index *I* equal to the ratio of the subject's weight *W* in pounds divided by twice the subject's height *H* in inches.
- A population of at least 20 known cold (unexposed) subjects of a diversity of indices was counted to generate a dataset of net count rate versus index.
- A curve fit to the dataset generated a prediction equation with the index as the variable.
- The subject's index was used to determine the predicted net count rate for the subject.

This approach was applied separately for the right chest, the left chest, and the liver/gut.

In 1973, a phoswich detector system [a detector with a primary scintillation NaI(Tl) layer backed by a CsI layer for coincidence counting] was implemented and used intermittently into the 1980s. The NaI(Tl) layer of the phoswich detectors was dimensionally the same as the 4- by 4-in. detectors.

This system lacked the stability of the NaI(Tl) detector system and was used mainly as a backup system. Use of the phoswich system to detect plutonium directly using the plutonium L X-rays was not successfully implemented at RFP.

In about 1972, room background was measured at the start of the day shift, at noon, and at the start of the night shift. The value of the room background RFP used was the five-point moving average of the last five counts.

Starting in 1969 (RFP 1965–1986), the ppm ^{241}Am was measured routinely from a representative sample of the plutonium mixture associated with incidents with the potential for inhalation exposure of workers. This situation was called a "PI" (for possible inhalation) and refers both to the incident and to the worker involved in the incident.

In this period, the use of a lithium-drifted germanium detector system was investigated but was never implemented.

1976 to 1985 (Author unknown 2002)
This period is the era of the high-purity germanium detector array systems. Three counting rooms were operational. When the germanium systems were implemented, most, if not all, quantitative measures were accomplished with that system. The NaI(Tl) and phoswich systems were used only as screening systems, and later only as backup systems. The germanium systems in this period featured four detectors mounted in an array attached to a single cryostat containing liquid nitrogen. The system had two of these arrays, one over the upper right chest and the other over the upper left chest. A full system consisted of eight detectors. However, occasionally one or more of the detectors failed and were electronically eliminated from the system. A minimum system was five detectors, three in the right array and two in the left. To maintain a minimum functional system, a hybrid system consisting of two arrays of different characteristics was frequently used.

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

The germanium system implementation timeline was:

- 1976. Ortec detectors, 10 cm^2 per detector, two arrays,
- 1977. PGT I detectors, 15 cm^2 per detector, two arrays,
- 1979. First array, PGT II detectors, 18 cm^2 per detector, and
- 1980. Second array, PGT II detectors, 18 cm^2 per detector.

Other changes in this period were:

- The calibration factor for the germanium systems was adjusted for the CWT of the subject. The thickness (centimeters) was equal to twice the index value minus 0.1 ($CWT = 2I - 0.1$).
- Calibration was accomplished using a Masonite phantom from 1976 to 1978.
- Calibration was accomplished using the LLNL phantom starting in 1979.
- The method of determining the background changed for the germanium systems. Room and subject background were determined as a unit from the subject's own spectrum using an ROI in the range of 65 to 72 keV.

1985 to 1995 (Author unknown 2002; Walraven 1991)
In this period, germanium detectors in an organ pipe configuration were implemented. Instead of clustering four detectors in an array with a common cryostat, each detector was attached to its own cryostat, which was tall and slender. The detectors with the cryostats were then clustered in arrays, two to four detectors per array, over the right and left portions of the upper chest. If a detector malfunctioned, it was physically replaced with a backup functional detector. A minimum system from 1985 to 1991 was five detectors, three on the right and two on the left. The full system was seven detectors, four on the right and three on the left, although the routine system generally consisted of six, either four on the right and two on the left or three on each side. In 1991, the full system was six detectors with either four on the right and two on the left or three on each side.

The germanium system implementation timeline was:

- 1985. PGT organ pipe detectors, 20 cm^2 per detector, and
- 1991. EG&G Ortec organ pipe detectors, 20 cm^2 per detector.

No other significant changes were made during this period.

1995 to 2005 (KHC 2000b)
In May 1995, the lung counter hardware, software, and detectors were upgraded. The data acquisition and analysis were accomplished using the Canberra Industries program ABACOS-Plus. Instead of the ROI approach that was used previously, this program used a peak-search method to detect activity of a radionuclide. The value of the MDA was established by replicate measurements on an appropriate blank. The germanium detectors were replaced by EG&G Ortec organ pipe detectors with 38 cm^2 per detector. The standard system was four detectors, two on each side. The minimum system was three detectors, two on the right and one on the left.

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MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

Another significant change (KHC 2000b) was the equation to determine *CWT*. ABACOS-Plus incorporates the equation developed at LLNL:

$$CWT\ (\text{cm}) = \frac{1.973W}{H} - 2.0038 \qquad \text{(B-3)}$$

where

W = subject's weight (pounds)
H = subject's height (inches)

The effect of this change is an adjustment factor given by:

$$CWT\ Adjustment\ Factor = 0.5364e^{0.635I} \qquad \text{(B-4)}$$

This adjustment factor is a multiplier to the activity of ^{241}Am, detected using the 59.5-keV gamma, for all previous detector systems at RFP. Equation B-4 can also be applied as a divisor to calibration factors for previous systems at RFP.

B.4 ASSESSMENT OF MDA

The value of the MDA for ^{241}Am is assessed here for each detector system and for each significant change in the procedure. It is assessed not only for the typical RFP male (*I* = 1.35, *CWT* = 3.3 cm) but also for a reasonable range of statures (*I* = 0.90, *CWT* = 1.5 cm and *I* = 1.80, *CWT* = 5.1 cm). The assessment is also done for the minimally configured system as well as for the standard system and for half of the normal count time (for expedited lung counts) as well as the full count time.

Discontinuities, which were significant changes in methods that affected the interpretation of the raw data (and therefore the MDA), were identified through review of available records and were incorporated into the value of the calibration factor. This process was done starting with the most recent calibration method, assumed to be the most accurate. The factors for each discontinuity were then applied as divisors to the calibration factor through the history of the systems. As an alternative, the product of the factors, for the appropriate period, can be used in place of the term $(1 + \Delta_K)$ in Equation B-1. Table B-1 lists the discontinuity factors.

Table B-1. Discontinuity factors.

Year	Discontinuity	Factor
1995	New CWT method, Index = 0.90	0.95
1995	Index = 1.35	1.26
1995	Index = 1.80	1.68
1979	Calibration using LLNL phantom	1.30
1969	Fixed positioning discontinued, ROI for 59.5-keV photopeak increased	1.45

Values of the Variables, 1964 to 1968
The minimum system was one NaI(Tl) detector over the left chest.

Count time *T* = 20 MLT or 40 MLT (Various 1952–1982, p. 360; Various 1954–1996, p. 121; Various 1958–2003, p. 115)

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The appropriate blank B was the net subject background (after room background was subtracted) estimated from matched unexposed subjects based on ^{137}Cs and ^{40}K measurements.

B = 600 for T = 20 MLT
B = 1,200 for T = 40 MLT
Δ_B = 0.2, estimated as the upper bound for this method

The value of s_0 is calculated from counting statistics, including the total subject count, which is taken as the sum of B and the room background R.

R = 500 for T = 20 MLT

R = 1,000 for T = 40 MLT

Because the decision of detection was based on the comparison of the net subject count rate (after subtraction of room background) with the predicted net count rate of the appropriate blank, the calculation of s_{B1} includes an extra component of the room background.

s_{B1}^2 = Total subject count + R = B + 2R
= 1,600 for T = 20 MLT
= 3,200 for T = 40 MLT
s_{B0}^2 = B
= 600 for T = 20 MLT
= 1,200 for T = 40 MLT
m = 1
s_0 = 44.9 for T = 20 MLT
= 66.3 for T = 40 MLT

The ^{241}Am calibration factor K for two detectors, normalized to the calibration with the LLNL phantom and incorporating the discontinuity factors (Equation B-4, 1.30, and 1.45) is given by:

$$K = \frac{55.13e^{-0.2359(2I-0.1)}}{e^{0.635I}} \quad \text{(B-5)}$$

For Equation B-5 and for subsequent equations of the calibration factor K, the calibration factors were normalized to Lawrence Livermore National Laboratory (LLNL) phantom, also called the Lawrence Livermore Torso Phantom and described in KHC (2000b). Normalizing to this phantom is consistent with the approach described in the second paragraph in Section B.4 of Attachment B because this phantom was used in "the most recent calibration method."

The calibration factor for the system with only one detector over the left portion of the chest is given by Equation B-5 multiplied by 0.43. This factor is the fraction of the total activity in the calibration lungs of the RFP LLNL phantom that is in the left portion of the lung. The MDA, therefore, pertains to the activity in the total lung based on the detection of activity only in the left portion of the lung.

K = 8.96 for I = 0.90
K = 5.45 for I = 1.35
K = 3.31 for I = 1.80

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

Because K is normalized to the calibration with the LLNL phantom and the discontinuity factors are incorporated into K, the value of Δ_K is taken to be zero. Because the term $(1 + \Delta_K)$ in Equation B-1 is a multiplier to the MDA and because the value of Δ_K is estimated based on the professional judgment of the analyst, one can easily adjust the values of the MDA in this attachment if another analyst has a different judgment.

For the standard system of two detectors, over both the right and left portions of the lungs, the counts are basically doubled and the values of the variables are:

B = 1,200 for T = 20 MLT
B = 2,400 for T = 40 MLT
Δ_B = 0.2, estimated as the upper bound for this method
R = 1,000 for T = 20 MLT
R = 2,000 for T = 40 MLT
s_{B1}^2 = Total subject count + $R = B + 2R$
= 3,200 for T = 20 MLT
= 6,400 for T = 40 MLT
s_{B0}^2 = B
= 1,200 for T = 20 MLT
= 2,400 for T = 40 MLT
m = 1
s_0 = 69.3 for T = 20 MLT
= 93.8 for T = 40 MLT
K = 20.85 for I = 0.90
= 12.67 for I = 1.35
= 7.70 for I = 1.80

B.5 VALUES OF THE VARIABLES, NaI(Tl) AND PHOSWICH DETECTOR SYSTEMS, 1969

The standard system was two detectors over the left and right portions of the chest. This is also the minimum system.

Count time T = 1,000 s or 2,000 s (RFP 1965–2005, pp. 22, 30, 83, 368, 803, 811, 817)

The appropriate blank was the net subject background (after room background was subtracted) estimated from matched, unexposed subjects based on the subject's index:

B = 1,100 for T = 1,000 s
B = 2,200 for T = 2,000 s
Δ_B = 0 for the NaI(Tl) detector system
Δ_B = 0.1, estimated for the phoswich detector system, because the system was less stable than the NaI(Tl) detector system

The value of s_0 is calculated from counting statistics, including the total subject count, which is taken as the sum of B and the room background R. The value of s_{B0} is taken to be 10% of the value B, based on the typical relative standard deviation of the predicted subject net count rate.

R = 833 for T = 1,000 s
R = 1,667 for T = 2,000 s
s_{B1}^2 = Total subject count + $R = B + 2R$

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

= 2,767 for T = 1,000 s
= 5,533 for T = 2,000 s
s_{B0}^2 = $(0.1B)^2$
= 12,100 for T = 1,000 s
= 48,400 for T = 2,000 s
m = 1
s_0 = 121.9 for T = 1,000 s
= 232.2 for T = 2,000 s

The ^{241}Am calibration factor K for two detectors, normalized to the calibration with the LLNL phantom and incorporating the discontinuity factors (Equation B-4 and 1.30) is given by:

$$K = \frac{79.94e^{-0.2359(2I-0.1)}}{e^{0.635I}} \quad \text{(B-6)}$$

Which yields the following results:

K = 30.23 for I = 0.90
= 18.37 for I = 1.35
= 11.16 for I = 1.80

B.6 VALUES OF THE VARIABLES, ORTEC GERMANIUM DETECTOR SYSTEMS, 1976

The standard system was two arrays, each array with four detectors over the left and right portions of the chest. The minimum system was two arrays with a total of eight detectors.

Count time T = 1,000 s or 2,000 s

The appropriate blank was the count in the subject's spectrum (composite for all detectors) in the range of 65 keV to 72 keV, divided by eight. The subject, in essence, was his own blank with essentially no bias. Room background was no longer assessed separately for germanium systems.

Δ_B = 0
m = 8

For eight detectors:

B = 341 for T = 1,000 s (unadjusted by m = 8)
B = 682 for T = 2,000 s (unadjusted by m = 8)

For five detectors:

B = 213 for T = 1,000 s (unadjusted by m = 8)
B = 427 for T = 2,000 s (unadjusted by m = 8)

For the calculation of s_{B1}, the subject background is B/8.

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

For eight detectors:

T = 1,000 s:
s_{B1} = 6.53 s_{B0} = 18.5 s_0 = 6.93
T = 2,000 s:
s_{B1} = 9.23 s_{B0} = 26.1 s_0 = 9.79

For five detectors:

T = 1,000 s:
s_{B1} = 5.17 s_{B0} = 14.6 s_0 = 5.48
T = 2,000 s:
s_{B1} = 7.30 s_{B0} = 20.7 s_0 = 7.75

The ^{241}Am calibration factor K for two arrays with a total of eight detectors, based on the calibration with the LLNL phantom and incorporating the discontinuity factors [Equation B-4 and 1.30 (for pre-1979 systems)] is given by:

$$K = \frac{24.12e^{-0.3398(2I-0.1)}}{e^{0.635I}} \quad \text{(B-7)}$$

and, for Ortec systems 1979 and after:

$$K = \frac{31.36e^{-0.3398(2I-0.1)}}{e^{0.635I}} \quad \text{(B-8)}$$

For the minimum system of five detectors, adjust the calibration factor by multiplying by (5/8).

Table B-2 lists the calibration factors for the Ortec germanium detector system.

Table B-2. Calibration factors (K) for the eight-detector Ortec germanium system.

Index	Pre-1979	1979
0.90	7.64	9.94
1.35	4.23	5.50
1.80	2.34	3.04

B.7 VALUES OF THE VARIABLES, PGT I GERMANIUM DETECTOR SYSTEMS, 1978

The PGT I germanium system is basically the same as the Ortec germanium system.

Δ_B = 0
m = 4

For eight detectors:

B = 240 for T = 1,000 s (unadjusted by m = 4)
B = 480 for T = 2,000 s (unadjusted by m = 4)

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

For five detectors:

B = 150 for T = 1,000 s (unadjusted by m = 4)
B = 300 for T = 2,000 s (unadjusted by m = 4)

For the calculation of s_{B1}, the subject background is B/4.

For eight detectors:

T = 1,000 s:
s_{B1} = 7.75 s_{B0} = 15.5 s_0 = 8.67

T = 2,000 s:
s_{B1} = 10.95 s_{B0} = 21.9 s_0 = 12.2

For five detectors:

T = 1,000 s:
s_{B1} = 6.12 s_{B0} = 12.2 s_0 = 6.84

T = 2,000 s:
s_{B1} = 8.66 s_{B0} = 17.3 s_0 = 9.68

The ^{241}Am calibration factor K for two arrays with a total of eight detectors, based on the calibration with the LLNL phantom and incorporating the discontinuity factors [Equation B-4 and 1.30 (for pre-1979 systems)] is given by:

$$K = \frac{34.09e^{-0.3292(2I-0.1)}}{e^{0.635I}} \quad \text{(B-9)}$$

and, for PGT I systems 1979 and after:

$$K = \frac{44.318e^{-0.3292(2I-0.1)}}{e^{0.635I}} \quad \text{(B-10)}$$

For the minimum system of five detectors, adjust the calibration factor by multiplying by 0.625 (5/8).

Table B-3 lists calibration factors for the PGT I germanium detector system.

Table B-3. Calibration factors (K) for the eight-detector PGT I germanium system.

Index	Pre-1979	1979 →
0.90	11.00	14.30
1.35	6.15	7.99
1.80	3.43	4.46

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

B.8 VALUES OF THE VARIABLES, PGT II GERMANIUM DETECTOR SYSTEMS, 1979

The PGT II germanium system is basically the same as the Ortec and PGT I systems.

$\Delta_B = 0$
$m = 4$

For eight detectors:

$B = 273$ for $T = 1{,}000$ s (unadjusted by $m = 4$)
$B = 546$ for $T = 2{,}000$ s (unadjusted by $m = 4$)

For five detectors:

$B = 170$ for $T = 1{,}000$ s (unadjusted by $m = 4$)
$B = 341$ for $T = 2{,}000$ s (unadjusted by $m = 4$)

For the calculation of s_{B1}, the subject background is $B/4$.

For eight detectors:

$T = 1{,}000$ s:
$s_{B1} = 8.26$ $s_{B0} = 16.5$ $s_0 = 9.23$

$T = 2{,}000$ s:
$s_{B1} = 11.7$ $s_{B0} = 23.4$ $s_0 = 13.1$

For five detectors:

$T = 1{,}000$ s:
$s_{B1} = 6.53$ $s_{B0} = 13.1$ $s_0 = 7.31$

$T = 2{,}000$ s:
$s_{B1} = 9.23$ $s_{B0} = 18.5$ $s_0 = 10.3$

The ^{241}Am calibration factor K for two arrays with a total of eight detectors (incorporating Equation B-4), is given by:

$$K = \frac{38.65e^{-0.3579(2I-0.1)}}{e^{0.635I}} \quad \text{(B-11)}$$

For the minimum system of five detectors, adjust the calibration factor by multiplying by 0.625 (5/8).

Table B-4 lists calibration factors for the PGT II germanium detector system.

Table B-4. Calibration factors (K) for the eight-detector PGT II germanium system.

Index	Calibration factor (K)
0.90	11.88
1.35	6.47
1.80	3.52

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

B.9 VALUES OF THE VARIABLES, PGT ORGAN PIPE GERMANIUM DETECTOR SYSTEMS, 1985

The PGT organ pipe germanium system is basically the same as the previous germanium array systems. The main difference is the ability to maintain a stable, standard configuration with six detectors.

$\Delta_B = 0$
$m = 4$

Table B-5 lists the values of variables for the PGT organ pipe germanium detector system.

Table B-5. Values of variables for the PGT organ pipe germanium detector system.

Variable	T = 1,000 s	T = 2,000 s
B	215	429
s_{B1}	7.33	10.4
s_{B0}	14.7	20.7
s_0	8.20	11.6

The ^{241}Am calibration factor K for two arrays with a total of six detectors (incorporating Equation B-4) is given by:

$$K = \frac{34.32e^{-0.2946(2I-0.1)}}{e^{0.635I}} \quad \text{(B-12)}$$

Table B-6 lists calibration factors for the PGT organ pipe germanium detector system.

Table B-6. Calibration factors (K) for the six-detector PGT organ pipe germanium detector system.

Index	Six-detector calibration factor (K)
0.90	11.74
1.35	6.77
1.80	3.90

B.10 VALUES OF THE VARIABLES, EG&G ORGAN PIPE GERMANIUM DETECTOR SYSTEMS, 1985

The EG&G organ pipe germanium system is basically the same as the previous PGT organ pipe germanium array system.

$\Delta_B = 0$
$m = 4$

Table B-7 lists the values of variables for the EG&G organ pipe germanium detector system.

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

Table B-7. Values of variables for the EG&G organ pipe germanium detector system.

Variable	*T* = 1,000 s	*T* = 2,000 s
B	204	408
s_{B1}	7.14	10.1
s_{B0}	14.3	20.2
s_0	7.98	11.3

The ^{241}Am calibration factor K for two arrays with a total of six detectors, incorporating Equation B-4, is given by:

$$K = \frac{42.36e^{-0.3708(2I-0.1)}}{e^{0.635I}} \quad \text{(B-13)}$$

Table B-8 lists calibration factors for the EG&G organ pipe General Electric detector system.

Table B-8. Calibration factors (K) for the six-detector EG&G organ pipe germanium detector system.

Index	Calibration factor (*K*)
0.90	12.73
1.35	6.85
1.80	3.69

Values of the Variables, 1995

The MDA for the system at RFP was not determined analytically using Equation B-1. Instead, the MDA was determined empirically from replicate measurements on an appropriate blank that simulated the counts of the average RFP worker (CWT = 3.36 cm). Therefore, there are no values of the variables to be listed here. The value of the MDA for the average RFP worker (CWT = 3.36 cm, I = 1.35) is 0.3 nCi ^{241}Am.

To extrapolate this value to the range of workers (CWT = 1.15 cm, I = 0.90 to CWT = 5.10 cm, I = 1.80), the following approach was used to establish the calibration factor equation as a function of CWT. The efficiency equation is:

$$\varepsilon = a_1^{a_2 CWT} \quad \text{(B-14)}$$

where

ε = counts per minute per gamma from ^{241}Am
a_1 = 0.045 (factor determined from calibration)
a_2 = -0.41 (factor determined from calibration)

The efficiency equation converts to the style of historical calibration equations using the conversion factors of 0.359 gamma photons (59.5 keV) per ^{241}Am nuclear transformation and 797 γ/min per nCi ^{241}Am. The derived calibration equation is:

$$K = 35.9^{-0.41CWT} \quad \text{(B-15)}$$

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

The MDA for any value of *CWT* is then obtained from the product of 0.3 nCi (the MDA for the average RFP worker) and the ratio (9.05/*K* for the value of *CWT*).

B.11 MDA FOR RFP PLUTONIUM

The MDA for RFP plutonium is derived from the MDA of ^{241}Am based on the value of the ppm ^{241}Am in the plutonium mixture at the time of the lung count. To convert the MDA for ^{241}Am to the MDA for plutonium (^{239}Pu and ^{240}Pu), the MDA for ^{241}Am is multiplied by the factor:

$$MDA\ Conversion\ Factor = \frac{1\times 10^{6} - ppm^{241}Am}{48.2} \qquad \text{(B-16)}$$

Table B-9 lists MDA conversion factors for some typical values of ppm ^{241}Am.

Table B-9. MDA conversion factors for values of ppm ^{241}Am.

ppm Am-241	MDA conversion factors
100	207
1,000	20.7
10,000	2.05

The task is to determine the value of the ppm ^{241}Am at the time of the lung count. The practice at RFP was to measure the ppm ^{241}Am in a representative sample of material from a possible inhalation incident. If a representative sample was not obtained or the origin of the intake was not known, a default value of 1,000 ppm ^{241}Am was used and was assigned to the date of the intake or to the date of the first positive lung count if the date of the intake was not known. For subsequent lung counts, the value of the ppm ^{241}Am was updated to account for the ingrowth of the ^{241}Am from the nuclear transformation of ^{241}Pu and for the radioactive decay of the ^{241}Am. The rate of ingrowth of ^{241}Am in the plutonium mixture depends on the fraction by weight of the ^{241}Pu in the mixture. The initial weight fraction of ^{241}Pu in RFP plutonium was taken to be 0.005 in the 1950s and 1960s and 0.0036 in the 1970s and later (See Table 5-1). The value of 0.0049 had been historically used at the RFP body counter as the rounded value of 0.005. Table B-10 lists values of the ppm ^{241}Am at times (years) after the intake for initial values of ppm ^{241}Am of 100, 1,000, and 10,000.

Table B-10. Americium-241 ingrowth in plutonium.

Initial fraction Pu-241 = 0.0036

Years	100	1,000	10,000
1	270	1,200	10,200
2	430	1,300	10,300
4	730	1,600	10,600
6	1,000	1,900	10,800
10	1,500	2,400	11,200
20	2,300	3,200	11,900
30	2,800	3,600	12,200
40	3,000	3,900	12,300
50	3,200	4,000	12,300

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

Initial fraction Pu-241 = 0.0050

Years	100	1,000	10,000
1	340	1,200	10,200
2	560	1,500	10,400
4	980	1,900	10,800
6	1,400	2,200	11,100
10	2,000	2,900	11,700
20	3,100	4,000	12,700
30	3,800	4,700	13,200
40	4,200	5,000	13,500
50	4,400	5,200	13,500

The appropriate value of the ppm ^{241}Am should be applied for lung counts that occurred after a known or assumed intake.

The equation to calculate the ppm ^{241}Am for any time (years) after the intake is:

$$A = L_1 P_0 \left(e^{-\lambda_{Pu241} T} - e^{-\lambda_{Am241} T} \right) + \frac{1 \times 10^6 A_0 L_2}{A_0 L_2 + e^{-\lambda_{Pu239} T}} \quad \text{(B-17)}$$

where

A = ppm ^{241}Am at time T (yr)
L_1 = $\lambda_{Pu241} \div (\lambda_{Am241} - \lambda_{Pu241})$
P_0 = initial ^{241}Pu ppm = (initial ^{241}Pu fraction by weight) × $(1 \times 10^6 - A_0)$
λ_{Pu241} = decay constant for ^{241}Pu (half-life = 14.4 yr) = 0.0481
λ_{Am241} = decay constant for ^{241}Am (half-life = 433 yr) =0.00160
A_0 = initial ppm ^{241}Am
L_2 = $\dfrac{e^{-\lambda_{Am241} T}}{1 \times 10^{-6} - A_0}$
λ_{Pu239} = decay constant for ^{239}Pu (half-life = 24,100 yr) = 0.0000288

Half-times are from *Table of Isotopes, Seventh Edition* (Lederer and Shirley 1978).

Table B-11 summarizes the americium MDAs for RFP in vivo lung counts.

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

Table B-11. Americium-241 MDA for in vivo lung counts.[a,b]

Period	Detector system	Index[c]	Minimum system half time[d]	Minimum system full time	Standard system half time[d]	Standard system full time
1964–1968[e]	NaI(Tl) 4×4	0.90	1.7	1.5	1.3	1.2
1964–1968[e]	NaI(Tl) 4×4	1.35	2.8	2.5	2.1	1.9
1964–1968[e]	NaI(Tl) 4×4	1.80	4.6	4.1	3.5	3.2
1969–[f]	NaI(Tl) 4×4	0.90	Not applicable	Not applicable	0.80	0.76
1969–[f]	NaI(Tl) 4×4	1.35	Not applicable	Not applicable	1.3	1.3
1969–[f]	NaI(Tl) 4×4	1.80	Not applicable	Not applicable	2.2	2.0
1973–[g]	Phoswich	0.90	Not applicable	Not applicable	1.2	1.2
1973–[g]	Phoswich	1.35	Not applicable	Not applicable	2.0	2.0
1973–[g]	Phoswich	1.80	Not applicable	Not applicable	3.3	3.2
1976–1978[h,i]	Ortec Arrays (high-purity Ge)	0.90	0.26	0.18	0.20	0.14
1976–1978[h,i]	Ortec Arrays (high-purity Ge)	1.35	0.48	0.32	0.37	0.25
1976–1978[h,i]	Ortec Arrays (high-purity Ge)	1.80	0.86	0.59	0.66	0.45
1979–[i]	Ortec Arrays (high-purity Ge)	0.90	0.20	0.14	0.16	0.11
1979–[i]	Ortec Arrays (high-purity Ge)	1.35	0.37	0.25	0.28	0.19
1979–[i]	Ortec Arrays (high-purity Ge)	1.80	0.66	0.45	0.51	0.35
1978–[i]	PGT I Arrays (high-purity Ge)	0.90	0.22	0.15	0.17	0.12
1978–[i]	PGT I Arrays (high-purity Ge)	1.35	0.40	0.27	0.31	0.21
1978–[i]	PGT I Arrays (high-purity Ge)	1.80	0.71	0.49	0.55	0.38
1979–[i]	PGT I Arrays (high-purity Ge)	0.90	0.17	0.12	0.13	0.09
1979–[i]	PGT I Arrays (high-purity Ge)	1.35	0.31	0.21	0.24	0.16
1979–[i]	PGT I Arrays (high-purity Ge)	1.80	0.55	0.38	0.42	0.29
1979–[i]	PGT II Arrays (high-purity Ge)	0.90	0.22	0.15	0.17	0.12
1979–[i]	PGT II Arrays (high-purity Ge)	1.35	0.40	0.28	0.31	0.21
1979–[i]	PGT II Arrays (high-purity Ge)	1.80	0.74	0.50	0.57	0.39
1985–[j]	PGT Organ Pipe (OP) Ge Detectors	0.90	Not applicable	Not applicable	0.15	0.11
1985–[j]	PGT Organ Pipe (OP) Ge Detectors	1.35	Not applicable	Not applicable	0.26	0.18
1985–[j]	PGT Organ Pipe (OP) Ge Detectors	1.80	Not applicable	Not applicable	0.46	0.32

ATTACHMENT B
MINIMUM DETECTABLE AMOUNTS FOR IN VIVO LUNG COUNTS (continued)

Period	Detector system	Index[c]	Minimum system half time[d]	Minimum system full time	Standard system half time[d]	Standard system full time
1991–[k]	EG&G Organ Pipe Ge Detectors	0.90	Not applicable	Not applicable	0.14	0.10
1991–[k]	EG&G Organ Pipe Ge Detectors	1.35	Not applicable	Not applicable	0.26	0.18
1991–[k]	EG&G Organ Pipe Ge Detectors	1.80	Not applicable	Not applicable	0.48	0.33
1995–[l]	Ortec 2 Organ Pipe Ge Detectors	0.90	Not applicable	Not applicable	Not applicable	0.14
1995–[l]	Ortec 2 Organ Pipe Ge Detectors	1.35	Not applicable	Not applicable	Not applicable	0.3
1995–[l]	Ortec 2 Organ Pipe Ge Detectors	1.80	Not applicable	Not applicable	Not applicable	0.6

a. Americium-241 grows into the plutonium mixture from the nuclear transformation of Pu-241. The initial weight fraction of Pu-241 in RFP plutonium was 0.0050 in the 1950s and 1960s and 0.0036 in the 1970s and 1980s. For freshly purified plutonium, with a residual of approximately 100 ppm Am-241, the ppm Am-241 would be 270 to 340 after the 1st year, 430 to 560 after the 2nd year, 730 to 980 after the 4th year, 1,000 to 1,400 after the 6th year, 1,500 to 2,000 after the 10th year, 2,300 to 3,100 after the 20th year, 2,800 to 3,800 after the 30th year, 3,000 to 4,200 after the 40th year, and 3,200 to 4,400 after the 50th year.

b. To convert the MDA for Am-241 to the MDA for Pu-239,240, multiply the MDA for $Am^{239,240}$ by $[(1 \times 10^6 - \text{ppm } Am^{239,240}) \div (48.2 \times \text{ppm } Am^{239,240})]$, where ppm $Am^{239,240}$ is the parts per million by weight of the $Am^{239,240}$ in the plutonium mixture at the time of the lung count.

c. The index is the ratio of the weight (pounds) of the subject divided by twice the height (inches) and is correlated with the CWT. The index of 1.35 represents the typical RFP male subject, with a reasonable range of 0.90 (CWT = 1.5 cm) to 1.80 (CWT = 5.1 cm).

d. Halved count times were usually used for nonscheduled counts or when a large number of subjects needed to be counted expeditiously.

e, Full time = 40 MLT; minimum system is one detector over the left chest; standard system is two detectors, over right and left chests.

f. Full time = 2,000 s; standard system is two detectors, over right and left chests.

g Full time = 2,000 s; standard system is two detectors, over right and left chests, NaI sensitive layer is the same as the NaI 4×4.

h. Starting in 1978, hybrid germanium systems were used that combined two different germanium arrays or detector types. For hybrid systems, use the higher of the MDA values for the involved detector types.

i. Full time = 2,000 s; standard system is eight detectors in two array; minimum system is five detectors in two arrays.

j. Standard system = six detectors; standard count time = 2,000 s; occasionally, five or seven detectors were used.

k. Standard system = six detectors; standard count time = 2,000 s.

l. Standard system = four detectors; standard count time = 2,000 s.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS

LIST OF FIGURES

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-1 Urinalysis Record Card and Health Sciences Data System - Urinalysis Detail (1) (first activity date on the HSDS portion: 10-29-54)

THE DOW CHEMICAL COMPANY — URINALYSIS RECORD CARD

TECHNIQUE CODING: A - [illegible] REPORTED IN MICROGRAMS/ ; B1: ELECTROPLATING B2: ETHER EXTRACT REPORTED IN 24 HRS. ; C1: CARRIER PRECIPITATION C2: TTA EXTRACTION REPORTED IN DPM/24 HRS.

		JAN.		FEB.		MAR.	APR.	MAY	JUNE	JULY	AUG.	SEPT.	OCT.	NOV.	DEC.
52	DAY														
	COUNT														
	TECH.														
53	DAY								19				16	11	
	COUNT								BK				BK	BK	
	TECH.								B1				B1	B2	
54	DAY	4	19	26	4	4	28	3	23	23	19	30	29		3
	COUNT	46	BK	111	81 30	81 30	1500	32	15	73	11.5	165	15		16
	TECH.	B2 RD	C2	B2	B2	B2	B2	B2	B2	B2	B2	A	A		A
55	DAY	14		25		24		20	24	26	19	22			2 29
	COUNT	9.2mg		10mg		8.1mg		BK	15mg	649	749	1249			BK
	TECH.	A		A		A		A	A	A	A	A			A
56	DAY	12	19	20		21	19	26	13	18	20	19	17	14	12
	COUNT	11mg	10mg	BK		8.8mg	BK	BK	BK	BK	BK	BK	BK	BK	BK
	TECH.	A	A	A		A	A	A	A	A	A	A	A	A	A

MFR No. [redacted] .359.24.1M.4

NAME OF EMPLOYEE: [redacted] 1414-336

BADGE NO. 327-110

AREA 3E44

HEALTH SCIENCES DATA SYSTEM - URINALYSIS DETAIL

SPECIFY EMPLOYEE NO. [redacted]

EMPLOYEE NO. [redacted] EMPLOYEE NAME [redacted]

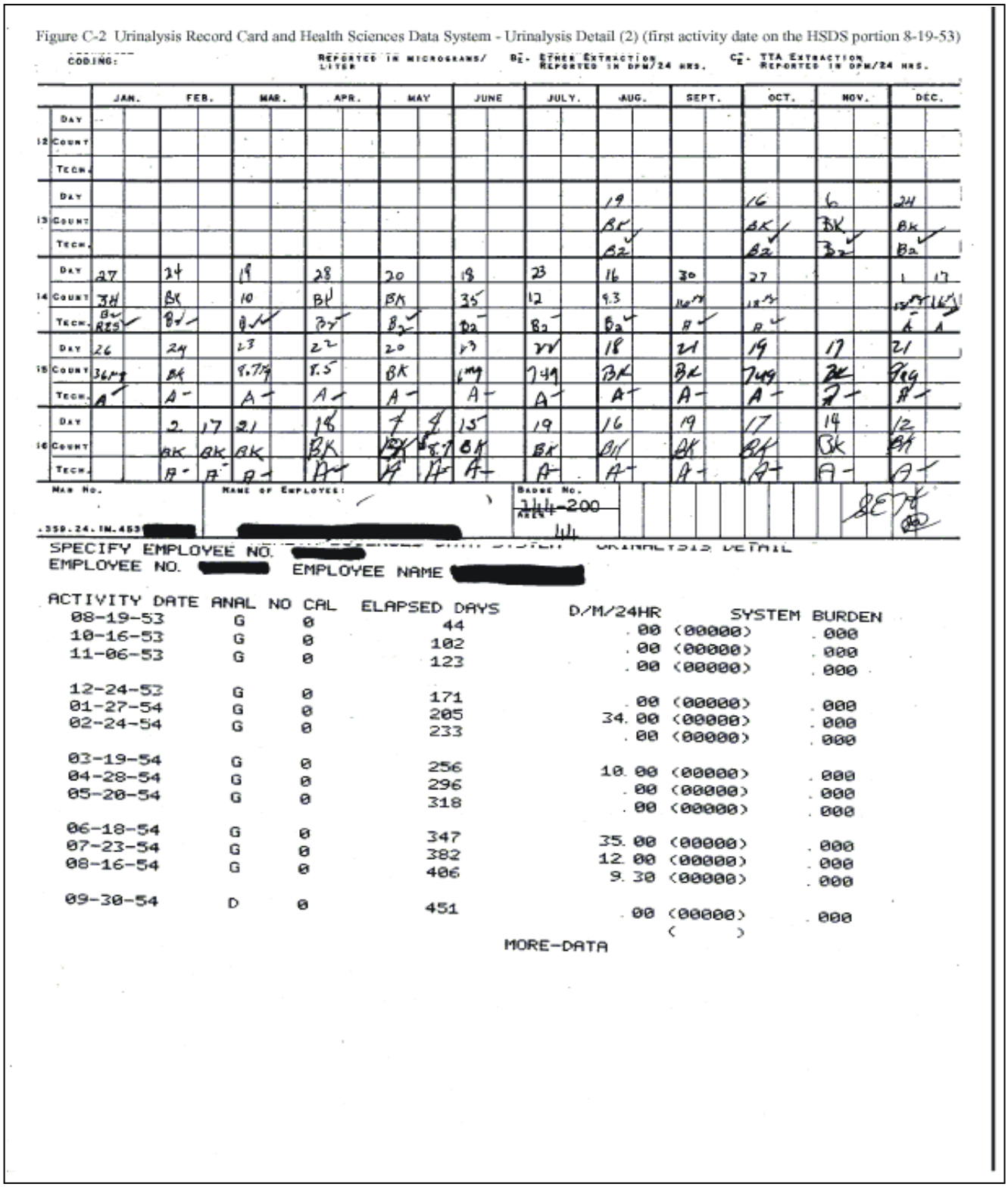
ACTIVITY DATE	ANAL	NO CAL	ELAPSED DAYS	D/M/24HR	SYSTEM BURDEN
10-29-54	U	0	550	15.00	.000
12-03-54	U	0	585	16.00	.000
01-14-55	U	0	627	9.20	.000
02-25-55	U	0	669	10.00	.000
03-24-55	U	0	696	8.10	.000
05-20-55	U	0	753	.00	.000
06-24-55	U	0	788	15.00	.000
07-29-55	U	0	823	6.00	.000
08-19-55	U	0	844	7.00	.000
09-22-55	U	0	878	12.00	.000
12-02-55	U	0	949	.00	.000
12-29-55	U	0	976	.00	.000
01-12-56	U	0	990	11.00	.000

MORE-DATA

Figure C-1. Urinalysis Record Card and HSDS – Urinalysis Detail (1) (first activity date on the HSDS portion: 10-29-54).

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-2 Urinalysis Record Card and Health Sciences Data System - Urinalysis Detail (2) (first activity date on the HSDS portion 8-19-53)



CODING: REPORTED IN MICROGRAMS/LITER B_2- ETHER EXTRACTION REPORTED IN DPM/24 HRS. C_2- TTA EXTRACTION REPORTED IN DPM/24 HRS.

	JAN.	FEB.	MAR.	APR.	MAY	JUNE	JULY.	AUG.	SEPT.	OCT.	NOV.	DEC.
12 DAY												
12 COUNT												
12 TECH.												
13 DAY								19		16	6	24
13 COUNT								BK		BK	BK	BK
13 TECH.								B_2		B_2	B_2	B_2
14 DAY	27	24	19	28	20	18	23	16	30	27		1 17
14 COUNT	3H	BK	10	BK	BK	35	12	9.3				
14 TECH.	B_2 RES	B_2	B_2	B_2	B_2	B_2	B_2	B_2	B	B		A A
15 DAY	26	24	23	22	20	19	22	18	21	19	17	21
15 COUNT	36mg	BK	8.7mg	8.5	BK	1mg	749	BK	BK	749		9mg
15 TECH.	A	A	A	A	A	A	A	A	A	A	A	A
16 DAY		2 17	21	18	7 4	15	19	16	19	17	14	12
16 COUNT		BK BK	BK	BK	10X 8.7	BK	BK	BK	BK	BK	BK	BK
16 TECH.		A A	A	A	H H	A	A	A	A	A	A	A

MAN NO. [redacted] NAME OF EMPLOYEE: [redacted] BADGE NO. 144-200 AREA 44

SPECIFY EMPLOYEE NO.

EMPLOYEE NO. [redacted] EMPLOYEE NAME [redacted]

ACTIVITY DATE	ANAL	NO CAL	ELAPSED DAYS	D/M/24HR	SYSTEM BURDEN
08-19-53	G	0	44	.00 (00000)	.000
10-16-53	G	0	102	.00 (00000)	.000
11-06-53	G	0	123	.00 (00000)	.000
12-24-53	G	0	171	.00 (00000)	.000
01-27-54	G	0	205	34.00 (00000)	.000
02-24-54	G	0	233	.00 (00000)	.000
03-19-54	G	0	256	10.00 (00000)	.000
04-28-54	G	0	296	.00 (00000)	.000
05-20-54	G	0	318	.00 (00000)	.000
06-18-54	G	0	347	35.00 (00000)	.000
07-23-54	G	0	382	12.00 (00000)	.000
08-16-54	G	0	406	9.30 (00000)	.000
09-30-54	D	0	451	.00 (00000)	.000

( )

MORE-DATA

Figure C-2. Urinalysis Record Card and HSDS – Urinalysis Detail (2) (first activity date on the HSDS portion 8-19-53).

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-3 Urinalysis Record Card and Health Sciences Data System - Urinalysis Detail (3) (first activity date on the HSDS portion: 1-6-58)

TECHNIQUE CODING: A-Fluorimeter Reported in micrograms/24-hrs. B_1- Electroplating B_2- Ether Extraction Reported in d/m/24 hrs. C_1- Carrier Precipitation, d/m/24-hrs.

		Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.
56	Day												
	Count												
	Tech												
57	Day	16	14	16	17	15	27	18	15	13	11	14	11
	Count	BK	BK	BK	BK	BK	BK	BK	BK	BK	BK	BK	BK
	Tech	A+	A+	A+	A+	A+	A+	A+	A+	A+	A+	A+	A+
58	Day	6	6	26	24						10		
	Count	BK	BK	BK	BK						BK		
	Tech	A+	B1+	C1+	C1+						A+		
59	Day		24								9		
	Count		BK								BK		
	Tech		A+								A+		
60	Day	13			8			12			12		14
	Count	BK			BK			BK			BK		BK
	Tech	A+			A+			A+			A+		A+

Man No. [redacted] Name of Employee: [redacted] Badge No. Area

HEALTH SCIENCES DATA SYSTEM - URINALYSIS DETAIL

SPECIFY EMPLOYEE NO. [redacted]

EMPLOYEE NO. [redacted] EMPLOYEE NAME [redacted]

ACTIVITY DATE	ANAL	NO CAL	ELAPSED DAYS	D/M/24HR	SYSTEM BURDEN
01-06-58	D	0	1,644	.00 (00000)	.000
02-06-58	U	0	1,675	.00 (00000)	.000
03-26-58	P	0	1,723	.00 (00000)	.000
04-24-58	P	0	1,752	.00 (00000)	.000
10-10-58	D	0	1,921	.00 (00000)	.000
02-24-59	D	0	2,058	.00 (00000)	.000
10-09-59	D	0	2,285	.00 (00000)	.000
01-13-60	D	0	2,381	.00 (00000)	.000
04-08-60	D	0	2,467	.00 (00000)	.000
07-12-60	D	0	2,562	.00 (00000)	.000
10-12-60	D	0	2,654	.00 (00000)	.000
12-14-60	D	0	2,717	.00 (00000)	.000
01-13-61	D	0	2,746	.00 (00000)	.000

MORE-DATA

Figure C-3. Urinalysis Record Card and HSDS – Urinalysis Detail (3) (first activity date on the HSDS portion: 1-6-58).

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-4 Health Sciences Data System - Urinalysis Detail (1) (first activity date 9-17-58)

HEALTH SCIENCES DATA SYSTEM - URINALYSIS DETAIL

SPECIFY EMPLOYEE NO. [redacted]

EMPLOYEE NO. [redacted] EMPLOYEE NAME [redacted]

ACTIVITY DATE	ANAL	NO CAL	ELAPSED DAYS	EXPOSURE VALUE	BODY BURDEN %
09-17-58	P	0	1,968	.00	.000
12-12-58	P	0	2,054	.00	.000
03-13-59	P	0	2,145	.00	.000
04-17-59	P	0	2,180	6.00	.258
04-20-59	P	0	2,183	.40	.139
04-21-59	P	0	2,184	1.94	.135
04-29-59	P	0	2,192	.18	.112
05-08-59	P	0	2,201	.12	.096
06-05-59	P	0	2,229	.07	.075
09-18-59	P	0	2,334	.24	.074
12-11-59	P	0	2,418	.22	.075
03-18-60	P	0	2,516	.16	.075
05-16-60	P	0	2,575	.07	.073

MORE-DATA

Figure C-4. HSDS – Urinalysis Detail (1) (first activity date 9-17-58).

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-5 Health Sciences Data System - Urinalysis Detail (2) (first activity date: 3-19-73)

HEALTH SCIENCES DATA SYSTEM - URINALYSIS DETAIL

SPECIFY EMPLOYEE NO. [redacted]

EMPLOYEE NO. [redacted] EMPLOYEE NAME [redacted]

ACTIVITY DATE	ANAL	NO CAL	ELAPSED DAYS	D/M/24HR	SYSTEM BURDEN
03-19-73	P	0	860	.09 (708)	.000
03-14-75	P	0	1,585	.00 (708)	.000
09-04-75	P	0	1,759	.00 (708)	.000
09-04-75	A	0	1,759	.00 (708)	.000
04-26-78	P	0	2,723	.08 (00000)	.005
01-30-79	P	0	3,002	.03 (00000)	.004
03-23-79	P	0	3,054	.14 (00000)	.012
03-23-79	A	0	3,054	.17 (00000)	.006
05-20-80	A	0	3,478	.03 (.109)	.003
05-20-80	P	0	3,478	.09 (.0752)	.010
03-24-81	A	0	3,785	.00 (0.62)	.002
03-24-81	P	0	3,785	.05 (0.03)	.009
05-01-81	P	0	3,823	.00 (.1060)	.008

END-OF-DATA

Figure C-5. HSDS – Urinalysis Detail (2) (first activity date 3-19-73).

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-6 Analytical Report - Bioassay Analysis Data 3-15-93

Employee Name : [redacted]
Employee Number: [redacted]
Worksheet ID : 123PU242_594

ANALYTICAL REPORT
Bioassay Analysis Data

Date: 15-MAR-1993

Analysis Type	Sample Date	Laboratory Sample Number	Det Level	Aspec	DQO	BLANK CHKR Val	Analyte	Recovery	Result (DPM)	Error
Special Urine	26-JAN-1993 13:33	14630/1	0.048	0	ANN	V	PU239	0.865	0.030	0.030
			0.053				PU238		-0.015	0.014

ASPEC CODES

0 = OK
1 = Analytical Failure
3 = Low recovery
4 = Poor Planchet

DQO'S - BLANK,ACCURACY,PRECISION

A = Acceptable
C = Conditional
F = Failed
U = Unassessed
N = Not assessed

Jo Ann Read, QA
3-17-93

Figure C-6. Analytical Report – Bioassay Analysis Data 3-15-93.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-7 Analytical Report - Bioassay Analysis Data 10-28-93

Employee Name : [redacted]
Employee Number: [redacted]
Worksheet ID : 123PU242_902

ANALYTICAL REPORT
Bioassay Analysis Data

Date: 28-OCT-1993

Analysis Type	Sample Date	Laboratory Sample Number	Dec Level	Aspec	DQO	Batch Val	Analyte	Recovery	Result (DPM)	Error
Routine Urine	8-SEP-1993 09:24	25747/1	0.016	0	ANN	V	PU239	0.896	0.002	0.006

ASPEC CODES

0 = OK
1 = Analytical Failure
3 = Low recovery
4 = Poor Planchet

DQO'S - BLANK,ACCURACY,PRECISION

A = Acceptable
C = Conditional
F = Failed
U = Unassessed
N = Not assessed

Jo Ann Read, QA
DATE OCT 29 1993

Figure C-7. Analytical Report – Bioassay Analysis Data 10-28-93.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-8 Form 1 - Sample Results 1-29-96

Date: 1/29/96

FORM I

SAMPLE RESULTS

LAB NAME: QUANTERRA, Richland SDG: 5915 COLLECTION DATE: 11/21/95

LAB SAMPLE ID: 51128827 REPORT NBR: 955 RECEIVED DATE: 11/22/95 10:30:00 AM

CLIENT ID: [redacted] ORDER NBR: MATRIX: URINE

ISOTOPE	RESULT	COUNTING ERROR (1s)	TOTAL ERROR (1s)	Lc	MDA	REPORT UNIT	YIELD	RST/MDA	RATIO/INTERR	ANALYSIS DATE	ALIQUOT SIZE	ALIQ UNIT	DETECTOR ID	METHOD NUMBER
PU239/40	9.69E-03	3.25E-03	3.92E-03	4.44E-03	1.21E-02	dpm/sa	94.90%	0.8	(U)	12/28/95	1039.3	ML	ALPHA-SPEC	RICHRC5010

Number of Results: 1

[redacted]

Sample Date: 10-16-95

Comments:

Figure C-8. Form 1 – Sample Results 1-29-96.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-9 Rocky Flats Environmental Technology Site (1) 8-27-96 (analytes: U238, U235, U234)

ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE
ANALYTICAL LABORATORIES--BIOASSAY 123/ENVIRONMENTAL 123

ANALYTICAL REPORT Date: 27-AUG-1996

Customer: DOSIMETRY
Sample Type: Routine Urine

Employee Number: [redacted]
Employee Name: [redacted]

Lab Sample #: 117598/1
Worksheet ID: 123U232_1483

Date Sampled: 3-JUL-1996 16:32:10.02
Date Received: 23-JUL-1996

Sample Size: 1200.000 ML
Aliquot Frac: 1200.000/ 1200.000

QA Data:

Alpha Spec Condition Code: 0
Chemical Recovery: 0.730
Data Qual Objective Codes: AAN

Analyte Results:

Analyte	RESULT (DPM)	ERROR (DPM)	DECISION LEVEL(DPM)	MDA (DPM)
U238	0.0282 >DL	0.0149	0.0281	0.0630
U235	0.0025	0.0078	0.0128	0.0324
U234	0.0295 >DL	0.0142	0.0225	0.0519

Comments: In March of 1995, the statistical method used for computing bioassay results was evaluated and EG&G Rocky Flats Internal Dosimetry initiated the use of a more appropriate statistical method for calculating the blank population variance. This report uses the new methodology for calculating the Decision Level, MDA, and the Results.

ASPEC CODES

0 = OK
1 = Analytical Failure
3 = Low recovery
4 = Poor Planchet
5 = High Recovery

DQO'S - BLANK, ACCURACY, PRECISION

A = Acceptable
C = Conditional
F = Failed
U = Unassessed
N = Not assessed

Michael M. Salmans
Michael M. Salmans AUG 29 1996
QA Officer

Data Validation Code: V Reviewed by [illegible] Date: 8-28-96

Figure C-9. Rocky Flats Environmental Technology Site (1) 8-27-96 (analytes: ^{238}U, ^{235}U, ^{234}U).

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-10 Rocky Flats Environmental Technology Site (1) 8-8-96 (analyte: Pu239)

ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE
ANALYTICAL LABORATORIES--BIOASSAY 123/ENVIRONMENTAL 123

ANALYTICAL REPORT Date: 28-AUG-1996

Customer: DOSIMETRY
Sample Type: Routine Urine

Employee Number: [redacted]
Employee Name:

Lab Sample #: 117598/2
Worksheet ID: 123PU242_3038

Date Sampled: 3-JUL-1996 16:32:10.02
Date Received: 23-JUL-1996

Sample Size: 1200.000 ML
Aliquot Frac: 1200.000/ 1200.000

QA Data:

Alpha Spec Condition Code: 0
Chemical Recovery: 0.838
Data Qual Objective Codes: AAN

Analyte Results:

Analyte	RESULT (DPM)	ERROR (DPM)	DECISION LEVEL(DPM)	MDA (DPM)
PU239	-0.0024	0.0038	0.0072	0.0192

Comments: In March of 1995, the statistical method used for computing bioassay results was evaluated and EG&G Rocky Flats Internal Dosimetry initiated the use of a more appropriate statistical method for calculating the blank population variance. This report uses the new methodology for calculating the Decision Level, MDA, and the Results.

ASPEC CODES

0 = OK
1 = Analytical Failure
3 = Low recovery
4 = Poor Planchet
5 = High Recovery

DQO'S - BLANK,ACCURACY,PRECISION

A = Acceptable
C = Conditional
F = Failed
U = Unassessed
N = Not assessed

Michael M. Salmans AUG 29 1996
Michael M. Salmans
QA Officer

Data Validation Code: V Reviewed by MJE Date: 8-28-96

Figure C-10. Rocky Flats Environmental Technology Site (1) 8-8-96 (analyte: ^{239}Pu).

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

FORM I

SAMPLE RESULTS

QUANTERRA, Richland

Date: 7/31/98

LAB SAMPLE ID: 80708501 SDG: 10398 COLLECTION DATE: 7/2/98 7:00:00 AM

CLIENT ID: 98M5156-001.001 REPORT NBR: 5725 RECEIVED DATE: 7/7/98 10:30:00 AM

EMPLOYEE NBR [redacted] RIN NBR: 98M5156 MATRIX: URINE

EMPLOYEE: [redacted]

ISOTOPE	RESULT	COUNTING ERROR (1 s)	TOTAL ERROR (1 s)	Lc	MDA	QUAL*	REPORT UNIT	YIELD	ANALYSIS DATE	TOTAL SA SIZE	ALIQUOT SIZE	UNIT	DETECTOR ID	METHOD NUMBER
U-234	5.75E-02	1.45E-02	1.51E-02	1.60E-02	4.10E-02	V	dpm/samp	62.80%	7/25/98	1594.7	1594.7	ML	ALPHA-SPEC	RICHRC5030
U-235	1.45E-03	4.90E-03	4.94E-03	6.34E-03	2.17E-02	V	dpm/samp	62.80%	7/25/98	1594.7	1594.7	ML	ALPHA-SPEC	RICHRC5030
U-238	2.54E-02	1.12E-02	1.15E-02	1.53E-02	3.95E-02	V	dpm/samp	62.80%	7/25/98	1594.7	1594.7	ML	ALPHA-SPEC	RICHRC5030
PU239/40	-2.91E-03	1.65E-03	1.80E-03	7.23E-03	1.94E-02	V	dpm/samp	75.20%	7/24/98	1594.7	1594.7	ML	ALPHA-SPEC	RICHRC5010

Number of Results: 4

0018

* (V=Valid; I=Invalid)

Comments:

(Validated by)

Date 8/3/98

Figure C-11 Form 1 - Sample Results - Quanterra, Richland 7-31-98

Figure C-11. Form 1 – Sample Results – Quanterra, Richland 7-31-98.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-12 General Engineering Laboratories, Inc. 6-28-99

General Engineering Laboratories, Inc.

Employee Name:

Employee Number: [redacted]

Lab ID: 9906900-06

RIN #: 99M8334

Sample #: 99M8334-006.001

Date Received: 28-JUN-99

Date Collected: 6/24/99 0600

Date Reported: 7/21/99

Sample Type: Urine

Parameter	Result	Uncertainty (1-Sigma)	Lc	MDA	Units	VF	Yield	Run Date	Sample Volume mL	Batch	Data Validation Code
Uranium-238	0.0244	0.0137	0.0258	0.0663	DPM/S	1.00	48.31	11-JUL-99	1802	152239	V 7/22/99
Uranium-235	0.0018	0.0092	0.0257	0.0662	DPM/S	1.00	48.31	11-JUL-99	1802	152239	V 7/22/99
Uranium-233/234	-0.0054	0.0110	0.0315	0.0776	DPM/S	1.00	48.31	11-JUL-99	1802	152239	V 7/22/99
Plutonium-239/240	-0.0008	0.0030	0.0060	0.0157	DPM/S	1.00	94.69	11-JUL-99	1802	152239	V 7/22/99

Comments:

This data report has been prepared and reviewed in accordance with General Engineering Laboratories, Inc. standard operating procedures.

Validated By: J. P. Connell

Page 1 of 1

019

Figure C-12. General Engineering Laboratories 6-28-99.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-13 Health Sciences Uranalysis Record (with tritium, fecal and nasal smear results)

HEALTH SCIENCES Urinalysis Record

PD = URINE DISTILLED
PR = URINE RAW
H_3 & C/R

NAME [redacted]
MAN NO. [redacted]

DATE	Element	Type Sample	Sample Type/ Anal	RESULTS
1-18-74	H_3	PD	6	7990 ± 486
2-5-74	"	"	150	21603 ± 5801
8-25-74	"	"		1753 ± 426
10-5-74	"	"	37	13201 ± 498
4-11-78	PU	FECAL	52	32.3 ± 1.8
1	AM	1	1	0.15 ± 0.13
1		1	1	PPM AM 100
5-17-78	PU	NASE	93	1.0 ± 0.28
1	AM	1	1	0.43 ± 0.20
1		1	1	PPM AM 9.200

Figure C-13. Health Sciences Urinalysis Record (with tritium, fecal, and nasal smear results).

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-14 Health Physics - Body Counter Information 12-8-65

Circulate:
~~[redacted]~~
~~CWP~~
~~EAP~~
CRL
~~SCH~~
~~JRCM~~
Pers. File

HEALTH PHYSICS

BODY COUNTER INFORMATION

Name [redacted] Man No. [redacted] Date 12-8-65 Time 4 PM CT

Reason for Counting: 76 Fire Exposure

minus Bby + Matth

Detectors	Body Location	Isotope	Results
# 1 : 4"x4mm NaI Crystal	R Chest	Am	33.8 c/m
# 2 : 4"x4mm NaI Crystal	L Chest	Am	31.2 c/m
# 3 : 9"x4" NaI Crystal			
			1.4 L B

☐ Collect all urine until further notice. Each void should be put in a separate jar. Mark the covers with the date and time of day.

☐ Collect all urine until further notice. All jars used in a 24-hour sampling period (midnite to midnite) should be marked with the date.

☐ Collect all fecal samples until further notice and mark date on the box.

☐ Collect fecal samples occasionally as per instructions and mark each box with the date.

Pennock
Operator

White copy: Circulate
Pink copy: To H. P. area office

RF-27740

Figure C-14. Health Physics – Body Counter Information 12-8-65.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-15 Health Physics - Body Counter Information 5-16-68

Circulate:
~~[redacted]~~
~~CHP~~
~~[redacted]~~
~~[redacted]~~
~~[redacted]~~
~~JHM~~
Pers. File

HEALTH PHYSICS

BODY COUNTER INFORMATION

Name [redacted] Man No. [redacted] Date 5-16-68 Time 1000

Reason for Counting: 76 Fire

Detector	Body Location	Isotope	Kev	Net (1) c/m	Mean	Net (2) c/m	µ g
#1 : 4"x4mm NaI Crystal	R chest	Am	60	52.2	34.9	17.3	0.065
#2 : 4"x4mm NaI Crystal	L chest	Am	60	46.6	34.5	11.7	0.044
							0.44
#3 : 9"x4" NaI Crystal				___ nci Cs^{137} ; ___ g Pu			

(1) Gross c/m — Background

(2) Gross c/m — Bkgd. — Match subject

Tally
Operator

White copy: Circulate
Blue copy: To H. P. area office

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Figure C-15. Health Physics – Body Counter Information 5-16-68.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-16 Health Physics - Body Counter Information 8-26-68

HEALTH PHYSICS
BODY COUNTER INFORMATION

CIRCULATE:
~~C.W.P.~~
~~E.A.P.~~
~~G.R.L.~~
~~[illegible]~~
~~J.R.M.~~
PERS. FILE

NAME: [redacted] BADGE NO. [redacted] DATE: 8-26-68 TIME: 1030

REASON FOR COUNTING:

BODY LOCATION	NET C/M	PREDICTED C/M	RESULT	INTERPRETATION OF DATA
LEFT CHEST	48.6	24.4	24.2	0.092 mg.
RIGHT CHEST				
LIVER	34.1	28.9	5.2	0.015 mg.
				X

REMARKS:

ECAL SAMPLING:
☐ NON REQUESTED ☐ NEXT SAMPLE ☐ CONTINUOUSLY

DISTRIBUTION:
WHITE-CIRCULATE
BLUE-TO H.P. AREA OFFICE

OPERATOR: Tally

27740 (Rev. 7-68) PREVIOUS ISSUE MAY BE USED

Figure C-16. Health Physics – Body Counter Information 8-26-68.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-17 Health Physics - Body Counter Information 9-16-70

HEALTH PHYSICS
BODY COUNTER INFORMATION

CIRCULATE:
C.W.P.
E.A.P.
C.R.L.
S.P.H.
J.R.M.
PERS. FILE

NAME: [redacted] BAR NO.: [redacted] DATE: 9-16-70 TIME: 1100

REASON FOR COUNTING: Routine

BODY LOCATION	NET C/M	PREDICTED C/M	RESULT	INTERPRETATION OF DATA
LEFT CHEST	41.3	36.2		
RIGHT CHEST	46.8	"		
LIVER	42.8	40.0		

REMARKS: Std Def 1.40

FECAL SAMPLING:
☐ NON REQUESTED ☐ NEXT SAMPLE ☐ CONTINUOUSLY

DISTRIBUTION:
WHITE- CIRCULATE
BLUE- TO H.P. AREA OFFICE

OPERATOR: Tsely

Figure C-17. Health Physics – Body Counter Information 9-16-70.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-18 Radiation Dosimetry - Body Count Results 10-3-74

PERSONAL & CONFIDENTIAL

RADIATION DOSIMETRY
BODY COUNT RESULTS

NAME: [redacted] MAN NO.: [redacted] DATE: 10-3-74 TIME: 0930

INDEX NUMBER: 165 ROOM: A (B) C

REASON FOR COUNTING: ☐ NEW Quarterly ☒ RECOUNT ☐ ROUTINE ☐ TERMINATION

☐ POSSIBLE INHALATION ☐ REQUEST BY: ______

BUILDING: ROOM: LINE OR OPERATION:

BODY LOCATION	NET C/M	PREDICTED C/M	RESULT C/M	nCi Pu	MPLB Pu	nCi Am	MPLB Am	RATIO		
RIGHT CHEST	45.2	34.3	10.9	3.42				1.361		
LEFT CHEST	42.2	33.5	8.7	2.73				1.218		
GUT										
TOTAL CHEST			19.6	6.15	0.38	0.66	0.04			

REMARKS:

INCIDENT SAMPLE: ppm ^{241}Am 2041 (calc). Chemical Form ______. Solubility ______.

URINE SAMPLING: ☐ NONE REQUESTED ☐ OVERNIGHT SAMPLE ☐ CONTINUOUSLY

FECAL SAMPLING: ☐ NONE REQUESTED ☐ NEXT SAMPLE ☐ CONTINUOUSLY

TECHNICIAN: RGF R.V. Reuchesk SUPERVISOR:

RET-286 (12-73)

PERSONAL & CONFIDENTIAL

Figure C-18. Radiation Dosimetry – Body Count Results 10-3-74.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-19 Radiation Dosimetry = Body County Results 5-30-75

RADIATION DOSIMETRY
BODY COUNT RESULTS

NAME: [redacted] MAN NO.: [redacted] DATE: 5-30-75 TIME: 1000

INDEX NUMBER: 1,55 ROOM: (A) B C

REASON FOR COUNTING: ☐ NEW ☐ RECOUNT ☒ ROUTINE ☐ TERMINATION

☐ POSSIBLE INHALATION ☐ REQUEST BY: _____

BUILDING: ROOM: LINE OR OPERATION:

BODY LOCATION	NET C/M	PREDICTED C/M	RESULT C/M	nCi Pu	MPLB Pu	nCi Am	MPLB Am	RATIO		
60 KEV RIGHT CHEST	36.9	33.2	Bkgd					1.13		
60 KEV LEFT CHEST	36.2	32.6	Bkgd					1.10		
17 KEV RIGHT CHEST										
17 KEV LEFT CHEST										
TOTAL CHEST										

REMARKS:

INCIDENT SAMPLE: ppm ^{241}Am _____. Chemical Form _____. Solubility _____.

URINE SAMPLING: ☐ NONE REQUESTED ☐ OVERNIGHT SAMPLE ☐ CONTINUOUSLY

FECAL SAMPLING: ☐ NONE REQUESTED ☐ NEXT SAMPLE ☐ CONTINUOUSLY

TECHNICIAN: RSF SUPERVISOR:

PERSONAL & CONFIDENTIAL

RFT-285 (12-73)

Figure C-19. Radiation Dosimetry – Body Count Results 5-30-75.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-20 Radiation Dosimetry - Body Count Results 1-9-78

PERSONAL - PRIVILEGED INFORMATION

RADIATION DOSIMETRY
BODY COUNT RESULTS

NAME: [redacted]	MAN NO: [redacted]	DATE: 1-9-78	TIME: 0843
INDEX NUMBER: 1,80	ROOM: A B C		
REASON FOR COUNTING: ☐ NEW Quarterly ☑ RECOUNT ☐ ROUTINE ☐ TERMINATION			
☐ POSSIBLE INHALATION ☐ REQUEST BY:			
BUILDING:	ROOM:	LINE OR OPERATION:	

BODY LOCATION	NET C/M	PREDICTED C/M	RESULT C/M	nCi Pu	MPLB Pu	nCi Am	MPLB Am	RATIO		
60 KEV RIGHT CHEST										
60 KEV LEFT CHEST										
17 KEV RIGHT CHEST										
17 KEV LEFT CHEST										
TOTAL CHEST	8.19	3.11	5.08 ± 0.50	6.4	0.40	0.81	0.055			

REMARKS:
Ge Array Hybrid
Calibration Factor Pu: 4.90 c/m per 16 nC Pu @ 1000 ppm Am; 12.67 @ 2585 ppm Am
Am: 6.25 c/m per nC Am

INCIDENT SAMPLE: ppm ^{241}Am 2585 (est.). Chemical Form _____. Solubility _____.

URINE SAMPLING: ☐ NONE REQUESTED ☐ OVERNIGHT SAMPLE ☐ CONTINUOUSLY

FECAL SAMPLING: ☐ NONE REQUESTED ☐ NEXT SAMPLE ☐ CONTINUOUSLY

TECHNICIAN: RBF | SUPERVISOR:

RFT-285 (12-73)

PERSONAL - PRIVILEGED

Figure C-20. Radiation Dosimetry – Body Count Results 1-9-78.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-21 Body Counter Results 12-8-81

BODY COUNTER RESULTS FILE

[redacted] MAN# [redacted] 12 / 8 / 1981

DETECTOR RIGHT PGT II INDEX 1.9
LEFT PGT II COUNTING TIME 2000 SE

COUNT RATE IN THE AMERICIUM REGION OF THE SPECTRUM IS + 8.49 C/M
COUNT RATE IN THE BACKGROUND REGION OF THE SPECTRUM IS + 4.07 C/M
DIFFERENCE + 4.42 C/M
STANDARD DEVIATION 0.51 C/M

CALIBRATION FACTORS
+ 3.55 C/M PER 16 NANOCURIES PU AT 1000 PPM AMERICIUM
+ 4.562 C/M PER NANOCURIE AM

PARTS PER MILLION AM FOR THIS COUNT- 3142 PPM

LUNG BURDEN CALCULATED FOR THIS COUNT

NANOCURIES FRACTION OF A LUNG BURDEN

PLUTONIUM+ 6.74 + OR - 0.74 + 0.40 + OR - 0.05
AMERICIUM+ 0.868 + OR - 0.112 + 0.066 + OR - 0.008

BODY COUNTER TECHNICIAN:

APPROVED BY: Roger B. Falk

Figure C-21. Body Counter Results 12-8-81.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-22 Radiation Dose Assessment - Body Count Results 7-22-83

PERSONAL-PRIVILEGED INFORMATION

RADIATION DOSE ASSESSMENT
BODY COUNT RESULTS

NAME: [redacted] EMPLOYEE #: [redacted]
DATE: 7/22/83 12:45 PM ROOM # B
COUNT TIME 2000 SEC INDEX #: 1.2318
BLDG #: 334 ROOM#: OPERATION:

REASON FOR COUNT: -- NEW -X- ROUTINE -- TERMINATION
DETECTORS: RIGHT: -X- PHOSWICH LEFT: -X- PHOSWICH

60 & 63 KEV

		93 KEV	
GROSS CT/MIN =	33.78	GROSS CT/MIN =	33.66
BKG CT/MIN =	33.66	BKG CT/MIN =	32.46
NET CT/MIN =	0.12	NET CT/MIN =	1.20
STD DEV =	1.42	STD DEV =	1.41
CUTOFF =	2.34	CUTOFF =	2.32

	ROI 2	ROI 3	ROI 4	ROI 5
SUM	593	1126	1122	1082

RESULTS ARE NORMAL
URINE SAMPLING: -- X NONE -- OVERNIGHT
-- X 24 HOUR -- THREE 24 HOUR
FECAL SAMPLING: -X- NONE -- ONE SAMPLE
-- 3 SAMPLES

BODY COUNTER TECHNICIAN: [signature]

APPROVED BY: Roger B. Falk

Figure C-22. Radiation Dose Assessment – Body Count Results 7-22-83.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-23 Radiation Dose Assessment - Body Count Results 5-13-83

PERSONAL-PRIVILEGED INFORMATION

RADIATION DOSE ASSESSMENT
BODY COUNT RESULTS

NAME: [redacted] EMPLOYEE #: [redacted]
DATE: 5/13/83 10:38 AM ROOM # A
COUNT TIME 2000 SEC INDEX #: 1.4758
BLDG #: 778 ROOM#: OPERATION:

REASON FOR COUNT: -- NEW -- RECOUNT X ROUTINE -- TERMINATION
-- POSSIBLE INHALATION (_____ PPM)
-- OTHER _____

DETECTORS: RIGHT: PGT-1 LEFT: PGT-1

AM-241/PLUTONIUM TH-234/U-238

GROSS CT/MIN =	4.98	GROSS CT/MIN =	5.16
BKG CT/MIN =	4.39	BKG CT/MIN =	4.99
NET CT/MIN =	0.59	NET CT/MIN =	0.17
STD DEV =	0.43	STD DEV =	0.43
CUTOFF =	0.70	CUTOFF =	0.71

	ROI 2	ROI 3	ROI 4	ROI 5	ROI 6	ROI 7	ROI 8	ROI 9	ROI10	ROI11
SUM	870	166	172	585	543	407	129	221	68	94

RESULTS ARE NORMAL

URINE SAMPLING: X NONE -- OVERNIGHT
-- 24 HOUR -- THREE 24 HOUR

FECAL SAMPLING: X NONE -- ONE SAMPLE
-- 3 SAMPLES

BODY COUNTER TECHNICIAN: [signature]

APPROVED BY: Roger D. Falk

Figure C-23. Radiation Dose Assessment – Body Count Results 5-18-83.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-24 Radiation Dose Assessment - Body Count Results 2-21-84

PERSONAL-PRIVILEGED INFORMATION

RADIATION DOSE ASSESSMENT
BODY COUNT RESULTS

NAME: [redacted] EMPLOYEE #: [redacted]
DATE: 2/21/84 1:27 PM ROOM # A
COUNT TIME 2000 SEC INDEX #: 1.2291
BUILD #: 444 ROOM#: OPERATION:
REASON FOR COUNT: __ NEW __ RECOUNT __ ROUTINE ___ TERMINATION
__ POSSIBLE INHALATION (_____ PPM)
OTHER _____
DETECTORS: RIGHT: PGT-1 LEFT: PGT-1

	AM-241/PLUTONIUM		TH-234/U-238	
GROSS CT/MIN	=	4.56	=	5.07
BKG CT/MIN	=	3.98	=	4.51
NET CT/MIN	=	0.58	=	0.56
STD DEV	=	0.41	=	0.43
CUTOFF	=	0.67	=	0.70

RESULTS ARE NORMAL

	L-XRAY	60KEV	63KEV	BKG	13KEV	17KEV	93KEV	93BKG	185KEV	185BKG
SUM	739	152	169	530	682	290	188	119	89	151

URINE SAMPLING: ✓ NONE __ OVERNIGHT __ 24 HOUR __ THREE 24 HOUR
FECAL SAMPLING: ✓ NONE __ ONE SAMPLE __ 3 SAMPLES

BODY COUNTER TECHNICIAN: W. Jacobs
APPROVED BY: Roger B. Falk

Figure C-24. Radiation Dose Assessment – Body Count Results 2-21-84.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-25 Radiation Dose Assessment - Body Count Results 3-22-84

PERSONAL-PRIVILEGED INFORMATION

RADIATION DOSE ASSESSMENT
BODY COUNT RESULTS

NAME: [redacted] EMPLOYEE #: [redacted]
DATE: 3/22/84 2:55 PM ROOM # A
COUNT TIME 2000 SEC INDEX #: 1.8115
BUILD #: 881 ROOM#: OPERATION:
REASON FOR COUNT: __ NEW __ RECOUNT (X) ROUTINE ___ TERMINATION
__ POSSIBLE INHALATION (_____ PPM)
OTHER _____
DETECTORS: RIGHT: PGT-1 LEFT: PGT-1

	AM-241/PLUTONIUM		TH-234/U-238	
GROSS CT/MIN	=	9.12	=	4.26
BKG CT/MIN	=	4.25	=	4.78
NET CT/MIN	=	4.87	=	-0.52
STD DEV	=	0.55	=	0.40

	L-XRAY	60KEV	63KEV	BKG	13KEV	17KEV	93KEV	93BKG	185KEV	185BKG
SUM	794	304	142	567	715	306	196	129	74	128

CALIBRATION FACTORS

CT/MIN PER 16 NANOCURIE PU @ 1000 PPM AM-241 = 4.19
CT/MIN PER NANOCURIE AM-241 = 5.392
PPM TODAY AM-241 = 3420 PPM

LUNG BURDEN CALCULATED FOR THIS COUNT

	NANOCURIES		FRACTION OF LUNG BURDEN	
PU-239 =	5.43 +-	0.62	0.34 +-	0.04
AM-241 =	0.903 +-	0.103	0.061 +-	0.007

URINE SAMPLING: /NONE __ OVERNIGHT __ 24 HOUR __ THREE 24 HOUR
FECAL SAMPLING: /NONE __ ONE SAMPLE __ 3 SAMPLES

BODY COUNTER TECHNICIAN: [signature]
APPROVED BY: Roger B. Falk

Figure C-25. Radiation Dose Assessment – Body Count Results 3-22-84.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-26 Radiation Dose Assessment - Body Count Results 10-10-85

PERSONAL-PRIVILEGED INFORMATION

RADIATION DOSE ASSESSMENT
BODY COUNT RESULTS

NAME: [redacted] EMPLOYEE #: [redacted]
DATE: 10/10/85 9:51 AM ROOM # C
COUNT TIME 2000 SEC INDEX #: 1.2238
BUILD #: 778 ROOM#: OPERATION:
REASON FOR COUNT: __ NEW __ RECOUNT _✓_ ROUTINE ___ TERMINATION
__ POSSIBLE INHALATION (_____ PPM)
OTHER _____
DETECTORS: RIGHT: PGT-2 LEFT: PGT-2

	AM-241/PLUTONIUM		TH-234/U-238
GROSS CT/MIN	= 3.84	GROSS CT/MIN	= 4.71
BKG CT/MIN	= 4.51	BKG CT/MIN	= 4.76
NET CT/MIN	= -0.67	NET CT/MIN	= -0.05
STD DEV	= 0.39	STD DEV	= 0.42
CUTOFF	= 0.63	CUTOFF	= 0.69

RESULTS ARE NORMAL

	L-XRAY	60KEV	63KEV	BKG	13KEV	17KEV	93KEV	93BKG	185KEV	185BKG
SUM	119407	128	157	601	90860	88124	144	187	56	78

URINE SAMPLING: _✓_ NONE __ OVERNIGHT __ 24 HOUR __ THREE 24 HOUR
FECAL SAMPLING: __ NONE __ ONE SAMPLE __ 3 SAMPLES

BODY COUNTER TECHNICIAN: C Barnett

APPROVED BY: Roger B. Falk

Figure C-26. Radiation Dose Assessment – Body Count Results 10-10-85.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-27 Radiation Dose Assessment - Body Count Results 3-6-89

SUM 638 1619 16 1595

PERSONAL-PRIVILEGED INFORMATION

RADIATION DOSE ASSESSMENT
BODY COUNT RESULTS

NAME: [redacted] EMPLOYEE #: [redacted]
DATE: 2/09/87 11:09 AM ROOM # B
COUNT TIME 2000 SEC INDEX #: 1.2910
BLDG #: 778 ROOM#: OPERATION:

REASON FOR COUNT: __ NEW __ ROUTINE ___ TERMINATION
DETECTORS: RIGHT: PHOSWICH LEFT: PHOSWICH

	60 & 63 KEV		93 KEV
GROSS CT/MIN =	47.57	GROSS CT/MIN =	49.71
BKG CT/MIN =	49.71	BKG CT/MIN =	47.85
NET CT/MIN =	-2.14	NET CT/MIN =	1.86
STD DEV =	1.72	STD DEV =	1.71
CUTOFF =	2.82	CUTOFF =	2.81

	ROI 2	ROI 3	ROI 4	ROI 5
SUM	638	1619	1657	1595

RESULTS ARE NORMAL

URINE SAMPLING: __ NONE __ OVERNIGHT
__ 24 HOUR __ THREE 24 HOUR
FECAL SAMPLING: __ NONE __ ONE SAMPLE
__ 3 SAMPLES

BODY COUNTER TECHNICIAN: _____

APPROVED BY: B.C. Baker

Figure C-27. Radiation Dose Assessment – Body Count Results 3-6-89.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-28 Internal Dosimetry - Lung Count Results 11-23-93

EG&G ROCKY FLATS
RADIOLOGICAL HEALTH

ANALYTICAL REPORT

PERSONAL-PRIVILEGED INFORMATION

INTERNAL DOSIMETRY
LUNG COUNT RESULTS

NAME: [redacted] EMPLOYEE #: [redacted]
DATE: 11/23/93 11:08 AM ROOM # B
COUNT TIME 2000 SEC INDEX #: 1.3615
BUILD #: [redacted] ROOM#: OPERATION:
REASON FOR COUNT: __ NEW __ RECOUNT __ ROUTINE ___ SEPARATION
__ POSSIBLE INHALATION (_____ PPM)
OTHER Accident
DETECTORS: RIGHT: EG&G 1, PGT OP 2,3,4 LEFT: EG&G 5, PGT OP 6

	AM-241/PLUTONIUM	TH-234/U-238 63KEV	93 KEV
GROSS CT/MIN =	12.69	4.38	3.78
BKG CT/MIN =	3.82	4.12	3.41
NET CT/MIN =	8.87	0.26	0.37
STD DEV =	0.64	0.40	0.40

	L-XRAY	60KEV	63KEV	BKG	13KEV	17KEV	93KEV	93BKG	185KEV	185BKG
ADC #1	91	80	21	86	30	39	19	29	18	24
ADC #2	126	94	19	73	49	60	17	42	15	26
ADC #3	194	53	29	92	529	106	22	32	8	22
ADC #4	121	40	23	102	49	52	25	40	16	29
ADC #5	83	113	17	78	27	43	18	34	18	35
ADC #6	119	43	37	82	52	66	25	38	17	26
ADC #7	0	0	0	0	0	0	0	0	0	0
ADC #8	0	0	0	0	0	0	0	0	0	0
SUM	734	423	146	513	736	366	126	215	92	162

CALIBRATION FACTORS

CT/MIN PER 16 NANOCURIE PU @ 1000 PPM AM-241 = 6.44
CT/MIN PER NANOCURIE AM-241 = 8.283
PPM TODAY AM-241 = 4339 PPM

ACTIVITY CALCULATED FOR THIS COUNT
NANOCURIES
PU-239 = 5.08 +- 0.37
AM-241 = 1.071 +- 0.077

DID EMPLOYEE SHOWER BEFORE COUNT? YES__ NO✓

LUNG COUNTER TECHNICIAN: YSA

APPROVED BY:_____

Figure C-28. Internal Dosimetry – Lung Count Results 11-23-93.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-29 ABACOS-Plus 3-6-96

ABACOS-Plus | Printed:6-MAR-1996 15:37:14. | Page: 1

Subject name: [redacted]
Employer: [redacted]
Job Code: [redacted]
Reason: Routine
Count type: Individual
Height: 77.000
Sex: Male
CWT: 3.4540
Comment:

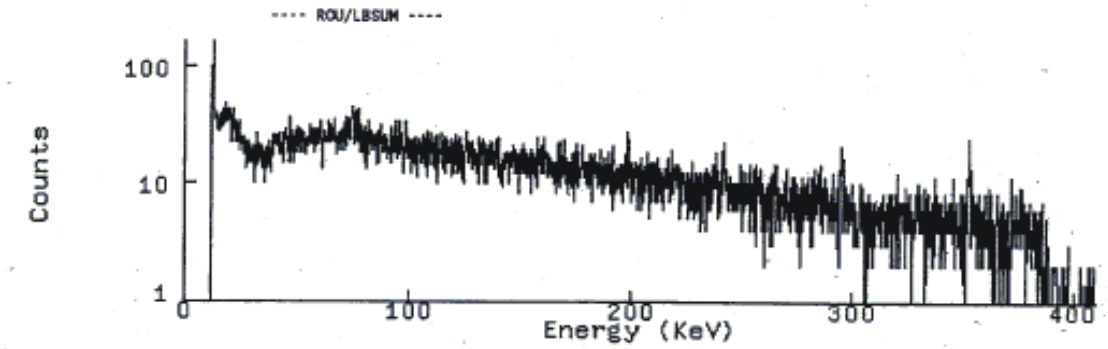
Identification #: [redacted]
Count Started: 6-MAR-1996 15:06:55.
Intake Date: 6-MAR-1996 15:06:55.
Frequency:
Operator: K.J. Hannie / K.J. Moul
Weight: 213.00
Date of Birth: [redacted]

Counter: Room B4 Sum
Arrangement/Geometry: ROU/LBSUM
Detector: LUNGS
Analysis limits: 200 to 1900

Facility: Rocky Flats ETS
Elapsed Live Time: 0 00:30:00.00
Elapsed Real Time: 0 00:30:00.35
Count Rate: 2.942 3.152 2.994 3.145 0.00 0.00 0.00 0.00

Nuclide Results for ROU/LBSUM

Nuclide	Activity (nCi)	%Error (2 SD)	Comments
U-235	<7.245E-02		Decision level reported
U-238	< 1.00		Decision level reported
AM-241	<0.117		Decision level reported
			----- No Nuclides Identified -----
totals:	0.000E+00		



Energy Calibration Performed: 14-FEB-1996 11:03:20 ROU/LBSUM Efficiency Calibration Performed: 12-FEB-1996 19:42:30
Libraries Used: ND_WBC_LIB:PU_OPTION.WLB,ND_WBC_LIB:PU_OPTION.WLB,ND_WBC_LIB:PU_OPTION.WLB
Analyses By: GAMMAN V1.2,PEAKEFF V2.2,NID V3.1,UTMEAN V1.8,NINACT V2.5

Reviewed by: Dennis Brown Date: March 7, 1996

Figure C-29. ABACOS-Plus 3-6-96.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-30 ABACOS-Plus 11-15-01

ABACUS-Plus | Printed:15-NOV-2001 08:31 | Page: 1

ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE IN-VIVO COUNTER

SENSITIVE - UNCLASSIFIED INFORMATION

Subject name: [redacted]
Employer: [redacted]
Job Code: [redacted]
Sex: Male
Height: 68.000
CWT: 3.6251
Reason: OTHER
Comment: MED RECALL

Identification #: [redacted]
Count Started: 15-NOV-2001 08:00:11
Intake Date: 15-NOV-2001 08:00:11
Date of Birth: [redacted]
Weight: 194.00

Operator: KJN Initials: [redacted]

Counter: Room A Lung
Analysis Limits (channels): 100 to 2000
Detector Count Rate (gross cps): 3.778 3.548 3.580 3.359
Analysis Energies (keV) Am-241: 59.54 U-235: 143, 185, and 204 U-238: 63 and 93

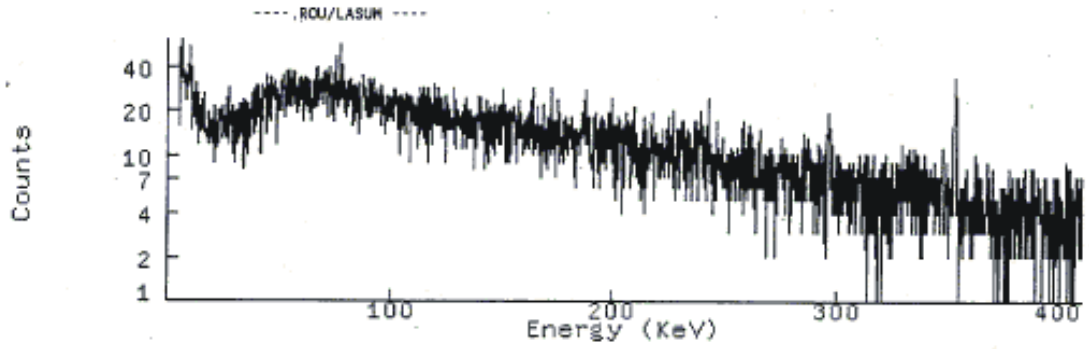
Arrangement/Geometry: RCU/LASUM
Elapsed Live Time: 0 00:30:00.00

MDA values are for an average RFETS worker. Results for an individual count may be greater than the stated MDA.

Nuclide Results for RCU/LASUM

Nuclide	Activity (nCi)	DL (nCi)	MDA (nCi)	%Error (1SD)	> DL	> MDA	Comments
U-235	-3.915E-02	0.654	0.790	-787.			
U-238	-1.38	2.12	2.10	-81.0			
AM-241	-0.119	0.230	0.290	-98.5			
			----- No Nuclides Identified -----				
totals:	-1.53						

Flags: "+" = nuclide identified.



Energy Calibration Performed: 1-OCT-2001 08:17:16. RCU/LASUM Efficiency Calibration Performed: 9-AUG-2001 13:10:09.
Libraries Used: ND_WBC_LIB:PU_OPTION_A.WLB,ND_WBC_LIB:PU_OPTION_A.WLB,ND_WBC_LIB:PU_OPTION_A.WLB
Analyses By: PEAK V16.9,PEAKEFF V2.2,NID V3.3,WTMEAN V1.8,MINACT V2.8

Reviewed by: [redacted] Date: 11-15-01

Figure C-30. ABACOS-Plus 11-15-01.

ATTACHMENT C
EXAMPLES OF BIOASSAY DATA RECORDS AND REPORTS (continued)

Figure C-31 ABACOS-Plus 6-14-01

ABACOS-Plus | Printed:14-JUN-2001 11:36 | Page: 1

ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE IN-VIVO COUNTER

SENSITIVE - UNCLASSIFIED INFORMATION

Subject name: [redacted]
Employer: [redacted]
Job Code: [redacted]
Sex: Male
Height: 68.000
CWT: 3.0738
Reason: OTHER
Comment: MEDICAL RECALL

Identification #: [redacted]
Count Started: 14-JUN-2001 11:05:57
Intake Date: 15-OCT-1965 00:00:00
Date of Birth: [redacted]
Weight: 175.00

Operator: JSH Initials: [redacted]

Counter: Room A Lung
Analysis Limits (channels): 100 to 2000
Detector Count Rate (gross cps): 4.062 3.619 3.782 3.563
Analysis Energies (keV) Am-241: 59.54 U-235: 143, 185, and 204 U-238: 63 and 93

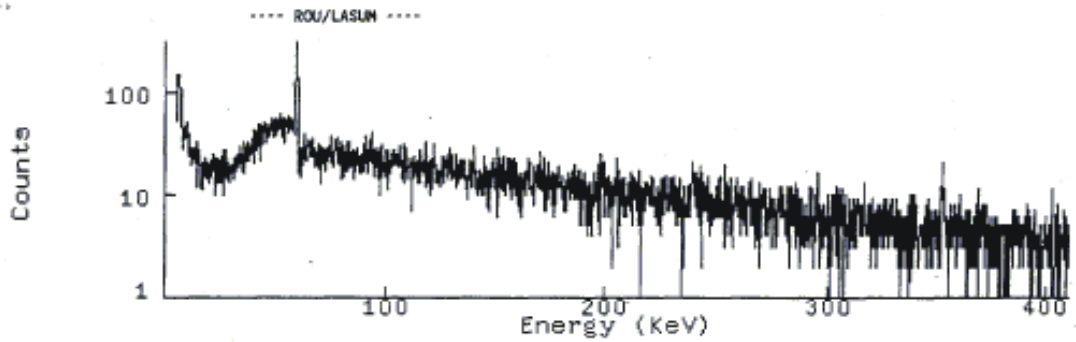
Arrangement/Geometry: ROU/LASUM
Elapsed Live Time: 0 00:30:00.00

MDA values are for an average RFETS worker. Results for an individual count may be greater than the stated MDA.

Nuclide Results for ROU/LASUM

Nuclide	Activity (nCi)	DL (nCi)	MDA (nCi)	%Error (1SD)	> DL	> MDA	Comments
U-235	0.397	0.561	1.10	70.6			
+PU-238	0.204	40.1	0.000E+00	30.5		>	
U-238	-0.327	1.66	2.90	-241.			
+PU-239	9.21	45.6	0.000E+00	30.5		>	
+PU-240	2.07	42.3	0.000E+00	30.5		>	
+AM-241	3.15	0.214	0.270	30.5	>	>	
+PU-241	13.7	71.2	0.000E+00	30.5		>	
+PU-242	1.865E-04	51.1	0.000E+00	30.5		>	
totals:	28.4						

Flags: "+" = nuclide identified



Energy Calibration Performed: 7-JUN-2001 06:43:59. ROU/LASUM Efficiency Calibration Performed: 15-FEB-2001 13:26:36
Libraries Used: ND_WBC_LIB:PU_OPTION_A.WLB,ND_WBC_LIB:PU_OPTION_A.WLB,ND_WBC_LIB:PU_OPTION_A.WLB
Analyses By: PEAK V16.9,PEAKEFF V2.2,NID V3.3,WTMEAN V1.8,MINACT V2.8

Reviewed by: [redacted] Date: 6/15/01

Figure C-31. ABACOS-Plus 6-14-01.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA

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ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

D.1 PURPOSE

Some employees at DOE sites were not monitored for internal ionizing radiation exposure, or the records of such monitoring are incomplete or unavailable. In such cases, data from monitored coworkers can be used to estimate an individual's possible exposure. The purpose of this attachment is to provide monitored co-exposure information for calculating and assigning occupational internal doses to employees at RFP for whom no or insufficient bioassay monitoring records exist.

D.1.1 Data Overview

This section provides information on the general selection characteristics of the data and the methods of analysis. More detailed radionuclide-specific information is provided in Section D.2.

D.1.2 Bioassay Data Selection

Urinalysis data used in this assessment for uranium and plutonium from 1952 to 1988 were extracted from the CEDR database. There were just over 300,000 records in the urinalysis database. Four cases had a date before 1952: one each in 1950 and 1951 and two that appeared to be date errors (years incorrectly entered as 1911 and 1923). Urinalysis data used in this assessment for uranium and plutonium from 1989 to 2005 were extracted from the HIS-20 database.

A comparison of CEDR and HIS-20 was made. The databases for the pre-1989 data were comparable but provide slightly differing results in some cases. These differences sometimes suggest CEDR could be slightly more favorable to the claimant while, in other cases, the data suggest HIS-20 could be slightly more favorable to the claimant. For the majority of the data, the results are similar. In addition, concern was expressed by the Rocky Flats Working Group that the number of samples in HIS-20 and CEDR were different in some cases. NIOSH demonstrated that the intakes that were predicted by either database were almost identical. However the concern on the part of the Working Group persisted. It was suggested and agreed that the use of the 95th percentile internal co-exposure intake for unmonitored workers with nontrivial exposure potential would satisfy this concern. It was also agreed that this situation and this policy were specific to RFP, and would set no precedent to be applied elsewhere.

In most cases, both the uranium and plutonium results were recorded as dpm/24 hr. However, the DU units are date-dependent. Through April 1964 the units were µg/24 hr; from May 1964 to 1988 the units were dpm/24 hr. Micrograms of uranium were converted to dpm by a 0.89 multiplier determined from the IMBA isotopic abundances for DU. Once converted to dpm, the uranium data were assumed to be entirely ^{234}U (Note: This assumption has no impact the statistics). See Section D.3.1 for additional discussion on using ^{234}U for the analyses.

All of the uranium and plutonium urinalysis results were recorded either as positive numbers or zeros. In general, a zero entry meant the result was less than some reporting level, but actual results were reported after April 6, 1970. Zeros were reported in 176,900 records, a little over half of the results for all measurements. Uranium and plutonium urinalysis data with a "1" flag in the "nocalc" field of the database (about 2,500 records out of roughly 300,000) should be (and were) excluded from analysis because the data did not meet quality objectives. For the post-1988 data, values were recorded as indicated by the analysis; that is, the actual results of the analyses were recorded even if it was a negative number. Because this analysis assumes the data can be represented by a lognormal distribution, the data had to be log-transformed. This required the zero and negative results to be treated as censored values.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

In vivo ^{241}Am lung data from 1965 to 1988 were extracted from a database table named "RFFACW02_RFWB." There were just fewer than 80,000 ^{241}Am records in the lung database. From 1965 through 1971, all results (about 4,000) were reported as zero, with no explanation of what those values might have meant. Therefore, no analyses were performed on those data. The ^{241}Am activities were quantified only if a known plutonium incident occurred. However, the results were sometimes recorded (in counts per minute) when no known incident had occurred. Some results were recorded in micrograms or nanocuries. Therefore, careful interpretation of the data units was imperative. Positive values began to appear after 1971, but there still were no exclusion instructions for when zero values were reported (see the "nocalc" discussion above). Therefore, zero results were treated as zeros because no better information was available. Calculations of the lung plutonium values that were recorded with the ^{241}Am lung data were determined by using the ^{241}Am data and an assumed concentration of 100 ppm (by weight) of ^{241}Am in the plutonium.

In both the urinalysis and lung-counting datasets, badge numbers (the ID column) are associated with most records. However, in the urinalysis data, 55,200 records had a "0" in the badge number column. It was not determined what a "0" badge ID meant other than, perhaps, to identify unbadged personnel. For the urinalysis data, about 34,000 of the "0" badges were plutonium records; 15,000 were gross alpha (A) and 6,000 were uranium (U). It was decided to treat "0" badge numbers as one individual when counting the number of unique individuals in any period. The "sdate" column provided the date of each analysis in YYMMDD order.

D.1.3 Analysis

Bioassay data were analyzed by quarter or year, depending on the amount of data available during the periods. A lognormal distribution was assumed. As mentioned in the previous section, a large fraction of the uranium and plutonium urinalysis data were entered as zeros. In many cases, this fact made analysis of the data difficult because so few positive values were reported. Therefore, where a reporting level was specified and where zeros were inserted for the actual values in the original data (below the reporting level), a linear distribution between zero and the reporting level was substituted for the zeros. The linear distribution had the form c/n, $2c/n$, $3c/n$,…, nc/n where n is the number of zero values less than the reporting level c. Using R^2 as the fit criterion, this linear distribution (alone) fits a lognormal transformation by better than 80% and typically significantly improves the goodness of fit for the entire dataset. Furthermore, the linear distribution has an average equal to half of the reporting value, consistent with the general dose reconstruction practice of assigning half of the lower limit of detection for missed dose calculations. As a consequence, substituting a linear distribution for these zero entries appears reasonable.

Whenever a linear distribution was substituted for values below a reporting level, the reporting levels were used. For EU, these reporting levels were 8.8 dpm/24 hr through 1963, and 20 to 28 dpm/24 hr after 1963. For DU, the reporting levels were 5.8 dpm/24 hr through April 1964, 20 to 28 dpm for May 1964 to 1979, and actual measured values thereafter. For plutonium, these reporting levels were 0.88 dpm/24 hr through 1961, 0.2 dpm/24 hr for 1962 to April 1970, and actual measured values after April 1970. The reporting level for gross alpha through 1963 was 8.8 dpm/24 hr (assigned as EU) and 0.9 dpm/24 hr thereafter (assigned as plutonium). No reporting level was given for americium-in-lung measurements.

After log-transforming the data, the 50th- and 84th-percentile values were determined for each period using the method described in ORAUT (2004). Tables D-1, D-2, and D-3 show the statistical analysis results for uranium, plutonium, and ^{241}Am, respectively.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

D.2 INTAKE MODELING

This section discusses intake modeling assumptions, intake fitting, and the intake materials (uranium and plutonium).

D.2.1 Assumptions

Each result in the intake calculations was assumed to be normally distributed. A uniform absolute error of 1 was applied to all results, thus assigning the same weight to each result. IMBA and IDOT require results to be in units of activity per day; therefore, all urinalysis results were normalized, as needed, to 24-hour samples, using 1,400 mL, the volume of urine excreted by Reference Man in a 24-hour period.

Because of the nature of work at RFP, it is possible that intakes could have been either chronic or acute. However, a series of acute intakes can be approximated as a chronic intake. Therefore, intakes were assumed to be chronic and were assumed to occur through inhalation, using a default breathing rate of 1.2 m^3/hr and a 5-µm AMAD particle size distribution (ICRP 1995).

For intake modeling purposes, all uranium activity was assumed to be ^{234}U. This assumption does not affect the fitting of the data for intake determination because all uranium isotopes behave the same biokinetically and the isotopes considered in this analysis all have long half-lives in relation to the assumed intake period. ICRP Publication 68 dose coefficients (also referred to as dose conversion factors or DCFs) for ^{234}U are 7% to 31% larger than those for ^{235}U, ^{236}U, and ^{238}U (ICRP 1995). Therefore, the assumption that the intake is 100% ^{234}U provides a result favorable to the claimant.

For plutonium, ^{239}Pu was assumed for the intake modeling. Before the mid-1970s, plutonium urinalysis was performed by chemical separation followed by the counting of all alpha-emitting isotopes of plutonium (i.e., ^{238}Pu, ^{239}Pu, and ^{240}Pu). In the mid-1970s, alpha spectroscopy was used to differentiate between them. For this modeling, the gross plutonium alpha results are assumed to represent only alphas from ^{239}Pu, which results in approximately a 2% overestimate of the modeled intakes. This assumption is made to enable consistent modeling of data from both types of urinalysis.

Starting in 1972, lung counts were performed to determine the lung burden of ^{241}Am. These lung counts can be used to determine the intake of plutonium. For each plutonium material type, the more limiting value of the intakes as determined by the americium lung counts or plutonium urinalysis was used. Use of the higher value (from the less sensitive bioassay method for a given material type) would be inconsistent with the available bioassay records because a higher intake would result in higher-than-observed bioassay results from the more sensitive bioassay method.

D.2.2 Bioassay Fitting

IMBA and IDOT were used to fit the bioassay results to a series of inhalation intakes. Data from 1952 through 2005 were fit as a series of chronic intakes. The intake assumptions were based on observed patterns in the bioassay data. Periods with constant chronic intake rates were chosen by selecting periods where the bioassay results were similar. A new chronic intake period was started if the data indicated a significant sustained change in the bioassay results. By this method, 1952 through 2005 were divided into multiple chronic intake periods.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

D.2.3 Material Types

See Section 5.2 for source term solubilities.

D.2.3.1 Uranium

Because the uranium isotopes at RFP have very long radiological half-lives and the material is retained in the body for long periods, excretion results 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 RFP for relatively short periods, each chronic intake was fit independently, using only the bioassay results from the single intake period for type S solubility. For type M and F solubilities, the approach used determined that earlier intake rates significantly biased later intake rates. This method results in a potential overestimate of intakes for exposures extending through multiple assumed intake periods. Only the results within the intake period were selected for use in fitting each period. Excluded results are shown in light gray in the figures at the end of this attachment.

Uranium urinalysis results were analyzed with IMBA to derive intake rates for 1953 to 2005. Excretion data are shown in Table D-1.

Uranium Type F
The solid lines in Figures D-1 through D-4 show the individual fits to the 50th-percentile excretion rates for type F material. Figure D-5 is the combined fit for all the intake periods. Figures D-6 through D-9 show the individual fits to the 84th-percentile excretion rates for type F uranium material. Figure D-10 shows the overall fit to the 84th-percentile excretion rates for type F uranium material. Table D-4 tabulates the derived intake rates at both the 50th- and 84th-percentile levels along with the associated geometric standard deviations (GSDs).

Uranium Type M
Figures D-11 through D-14 show the individual fits to the 50th-percentile excretion rates for type M uranium material. Figure D-15 is the combined fit for all the intake periods. Figures D-16 through D-18 show the individual fits to the 84th-percentile excretion rates for type M uranium material. Figure D-19 shows the overall fit to the 84th-percentile excretion rates for type M uranium material. Table D-5 tabulates the derived intake rates at both the 50th- and 84th-percentile levels along with the associated GSDs.

Uranium Type S
Figures D-20 through D-26 show the individual fits to the 50th-percentile excretion rates for type S uranium material. Figure D-27 is the combined fit for all the intake periods. Figures D-28 through D-34 show the individual fits to the 84th-percentile excretion rates for type S uranium material. Figure D-35 shows the overall fit to the 84th-percentile excretion rates for type S uranium material. Table D-6 tabulates the derived intake rates at both the 50th- and 84th-percentile levels along with the associated GSDs.

D.2.3.2 Plutonium

Plutonium urinalysis results were analyzed with IMBA using type M and S materials to derive intake rates for 1952 to 2005 and IDOT using type SS material to derive intake rates for 1952 to 2005. As with type S uranium, plutonium isotopes at RFP have very long radiological half-lives and the material is retained in the body for long periods, so excretion results are not independent. To avoid potential

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

underestimation of intakes for people who worked at RFP for relatively short periods, each chronic intake was fit independently, using only the bioassay results from the single intake period for type M, S and SS solubility. This method results in a potential overestimate of intakes for exposures extending through multiple assumed intake periods. Only the results within the intake period were selected for use in fitting each period. Excluded results are shown in light gray in the figures. Tables D-2 and D-3 provide the bioassay data that were used to perform the fits.

Plutonium Type M
The solid lines in Figures D-36 through D-41 show the individual fits to the 50th-percentile excretion rates for type M plutonium material. Figure D-42 is the combined fit for all the intake periods. Figures D-43 through D-48 show the individual fits to the 84th-percentile excretion rates for type M plutonium material. Figure D-49 is the combined fit for all the intake periods. Table D-7 lists the 50th- and 84th-percentile intake rates along with the associated GSDs determined from the plutonium urinalysis.

Plutonium Type S
The solid lines in Figures D-50 through D-54 show the individual fits to the 50th-percentile excretion rates for type S plutonium material. Figure D-55 is the combined fit for all the intake periods. Figures D-56 through D-60 show the individual fits to the 84th-percentile excretion rates for type S plutonium material. Figure D-61 is the combined fit for all the intake periods.

The solid lines in Figures D-62 through D-64 show the individual fits to the 50th-percentile ^{241}Am lung burdens for type S material. Figure D-65 is the combined fit for all the intake periods. Figures D-66 through D-68 show the individual fits to the 84th-percentile ^{241}Am lung burdens for type S material. Figure D-69 is the combined fit for all the intake periods.

Table D-8 lists the 50th- and 84th-percentile intake rates along with the associated GSD determined from the plutonium urinalysis and ^{241}Am lung count data.

Plutonium Type SS
The solid lines in Figures D-70 through D-74 show the individual fits to the 50th-percentile excretion rates for type SS plutonium material. Figure D-75 is the combined fit for all the intake periods. Figures D-76 through D-80 show the individual fits to the 84th-percentile excretion rates for type SS plutonium material. Figure D-81 is the combined fit for all the intake periods.

The solid lines in Figures D-82 through D-84 show the individual fits to the 50th-percentile ^{241}Am lung burdens for type SS material. Figure D-85 is the combined fit for all the intake periods. Figures D-86 through D-88 show the individual fits to the 84th-percentile ^{241}Am lung burdens for type SS material. Figure D-89 is the combined fit for all the intake periods.

Table D-9 lists the 50th- and 84th-percentile intake rates along with the associated GSD determined from the plutonium urinalysis and ^{241}Am lung count data.

D.3 ASSIGNING INTAKES AND DOSES

This section describes the derived intake rates and provides guidance for assigning doses. For each intake period discussed below, the 50th- and 84th-percentile calculated intakes were used to determine the GSD of the data. The GSD along with the geometric mean were used to calculate the 95th-percentile intake rate. In 1993, the Secretary of Energy formally announced the end of nuclear production at RFP. Remediation was completed at the RFP in late 2005. Co-exposure intakes

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INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

should be assigned, when applicable, up through 2005. Only environmental intakes should be assigned after 2005.

D.3.1 Intake Rate Summary

Multiple intake periods were fit to the derived 50th- and 84th-percentile uranium excretion data. Tables D-4 through D-6 summarize the 95th-percentile uranium intake rates derived from the fits. Similarly, multiple intake periods were fit to the derived 50th- and 84th-percentile plutonium excretion and americium lung burden data for type M, S, and SS material. Tables D-7 through D-9 summarize the 95th-percentile plutonium intake rates derived from the fits.

D.3.2 Dose Assignment

Doses to be assigned to individuals are calculated from the 95th-percentile intake rates. Dose reconstructors should select the material type that is the most favorable to the claimant. The constant distribution is selected in IREP, with the calculated dose entered as Parameter 1.

D.4 CO-EXPOSURE DATA TABLES AND FIGURES

Table D-1. Summary of uranium urinary excretion rate analyses, 1953 to 2005.

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
7/1/1953	3.727E+00	1.001E+01
2/15/1954	3.866E+00	1.036E+01
5/15/1954	4.161E+00	1.147E+01
8/15/1954	3.732E+00	1.007E+01
11/15/1954	3.409E+00	9.389E+00
2/15/1955	3.225E+00	9.019E+00
5/15/1955	3.333E+00	9.487E+00
8/15/1955	3.434E+00	9.406E+00
11/15/1955	3.442E+00	9.875E+00
2/15/1956	3.310E+00	9.039E+00
5/15/1956	3.497E+00	9.843E+00
8/15/1956	3.635E+00	1.021E+01
11/15/1956	3.302E+00	9.121E+00
2/15/1957	3.460E+00	9.894E+00
5/15/1957	3.492E+00	1.017E+01
8/15/1957	3.655E+00	1.078E+01
11/15/1957	3.700E+00	1.100E+01
2/15/1958	4.089E+00	1.258E+01
5/15/1958	3.739E+00	1.059E+01
8/15/1958	3.907E+00	1.127E+01
11/15/1958	4.705E+00	1.432E+01
2/15/1959	4.381E+00	1.316E+01
5/15/1959	5.518E+00	1.791E+01
8/15/1959	5.544E+00	1.657E+01
11/15/1959	5.887E+00	1.913E+01
2/15/1960	8.806E+00	3.307E+01
5/15/1960	6.856E+00	2.223E+01
8/15/1960	7.476E+00	2.421E+01
11/15/1960	6.602E+00	2.367E+01
2/15/1961	5.944E+00	2.026E+01
5/15/1961	5.722E+00	1.863E+01
8/15/1961	5.574E+00	1.829E+01
11/15/1961	6.598E+00	2.267E+01
2/15/1962	5.862E+00	2.045E+01
5/15/1962	4.692E+00	1.538E+01
8/15/1962	5.654E+00	1.674E+01
11/15/1962	4.397E+00	1.383E+01
2/15/1963	4.166E+00	1.323E+01
5/15/1963	4.175E+00	1.315E+01
8/15/1963	3.841E+00	1.228E+01
11/15/1963	3.601E+00	1.151E+01
2/15/1964	6.354E+00	1.851E+01
5/15/1964	8.368E+00	2.339E+01
8/15/1964	8.161E+00	2.217E+01
11/15/1964	8.297E+00	2.354E+01
7/1/1965	7.823E+00	2.079E+01
7/1/1966	7.432E+00	1.836E+01
7/1/1967	7.445E+00	1.844E+01
7/1/1968	7.430E+00	1.846E+01
7/1/1969	7.509E+00	1.852E+01
7/1/1970	7.440E+00	1.828E+01
7/1/1971	7.421E+00	1.813E+01
7/1/1972	7.316E+00	1.818E+01
7/1/1973	7.403E+00	1.806E+01
7/1/1974	7.388E+00	1.808E+01
7/1/1975	7.378E+00	1.810E+01
7/1/1976	7.418E+00	1.804E+01
7/1/1977	1.720E-01	5.380E-01
7/1/1978	8.930E-01	2.355E+00
7/1/1979	4.440E-01	2.037E+00
7/1/1980	2.410E-01	1.049E+00
7/1/1981	1.780E-01	1.109E+00
2/15/1982	2.370E-01	1.152E+00
5/15/1982	6.200E-02	6.770E-01
8/15/1982	1.600E-02	2.110E-01
11/15/1982	1.120E-01	7.410E-01

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INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
2/15/1983	2.210E-01	1.062E+00
5/15/1983	4.320E-01	1.330E+00
8/15/1983	3.270E-01	1.576E+00
11/15/1983	7.200E-02	6.460E-01
2/15/1984	2.730E-01	1.400E+00
5/15/1984	2.210E-01	1.330E+00
8/15/1984	1.330E-01	9.970E-01
11/15/1984	6.500E-02	4.640E-01
2/15/1985	3.400E-02	4.100E-01
5/15/1985	3.000E-02	2.810E-01
8/15/1985	4.000E-02	5.110E-01
11/15/1985	3.700E-02	4.150E-01
2/15/1986	2.900E-02	3.570E-01
5/15/1986	3.300E-02	3.390E-01
8/15/1986	1.800E-02	2.070E-01
11/15/1986	2.200E-02	3.160E-01
7/1/1987	5.700E-02	4.670E-01
7/1/1988	5.900E-02	4.120E-01
2/15/1989	5.400E-02	7.260E-01
5/15/1989	4.200E-02	3.370E-01
8/15/1989	3.500E-02	2.580E-01
11/15/1989	5.300E-02	2.900E-01
7/1/1990	3.300E-02	2.120E-01
7/1/1991	3.700E-02	1.870E-01
7/1/1992	6.700E-02	6.780E-01
7/1/1993	4.800E-02	2.780E-01
2/15/1994	8.000E-03	7.100E-02
5/15/1994	2.700E-02	1.820E-01
8/15/1994	1.000E-02	1.000E-01
11/15/1994	2.100E-02	1.260E-01
7/1/1995	2.300E-02	1.240E-01
2/15/1996	6.000E-03	7.500E-02
5/15/1996	3.000E-03	4.700E-02
8/15/1996	4.000E-03	4.200E-02
11/15/1996	7.000E-03	4.700E-02
2/15/1997	1.000E-02	6.700E-02
5/15/1997	1.400E-02	7.600E-02
8/15/1997	1.400E-02	6.000E-02
11/15/1997	1.700E-02	6.000E-02
7/1/1998	4.000E-03	2.400E-02
7/1/1999	8.000E-03	4.600E-02
2/15/2000	7.000E-03	4.600E-02
5/15/2000	4.000E-03	8.900E-02
8/15/2000	4.000E-03	3.600E-02
11/15/2000	5.000E-03	3.800E-02
2/15/2001	1.000E-02	4.100E-02
5/15/2001	1.100E-02	6.600E-02
8/15/2001	7.000E-03	4.900E-02
11/15/2001	6.000E-03	6.800E-02
2/15/2002	9.000E-03	7.200E-02
5/15/2002	1.600E-02	1.420E-01
8/15/2002	1.200E-02	6.550E-01
11/15/2002	9.000E-03	1.690E-01
2/15/2003	6.000E-03	1.100E-01
5/15/2003	1.200E-02	7.400E-02
8/15/2003	3.000E-03	3.200E-02
11/15/2003	1.400E-02	5.400E-02
2/15/2004	1.800E-02	8.900E-02
5/15/2004	1.500E-02	7.900E-02
8/15/2004	8.000E-03	5.500E-02
11/15/2004	7.000E-03	4.940E-01
7/1/2005	1.200E-02	4.800E-02

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

Table D-2. Summary of plutonium urinary excretion rate analyses, 1952 to 2005.[a,b]

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
7/1/1952	2.514E+00	8.198E+00
7/1/1953	7.160E-01	1.046E+00
7/1/1954	5.750E-01	1.053E+00
7/1/1955	4.690E-01	9.190E-01
7/1/1956	6.150E-01	1.264E+00
7/1/1957	2.610E+00	1.201E+01
2/15/1958	2.173E+00	1.004E+01
5/15/1958	1.037E+00	2.872E+00
8/15/1958	1.295E+00	3.801E+00
11/15/1958	9.190E-01	2.581E+00
2/15/1959	7.090E-01	1.542E+00
5/15/1959	9.420E-01	2.276E+00
8/15/1959	9.450E-01	2.482E+00
11/15/1959	5.600E-01	1.211E+00
2/15/1960	6.140E-01	1.353E+00
5/15/1960	5.960E-01	1.221E+00
8/15/1960	4.530E-01	9.550E-01
11/15/1960	5.730E-01	1.528E+00
2/15/1961	7.280E-01	1.625E+00
5/15/1961	6.910E-01	1.377E+00
8/15/1961	7.540E-01	2.035E+00
11/15/1961	6.560E-01	1.645E+00
2/15/1962	3.370E-01	8.090E-01
5/15/1962	3.260E-01	7.350E-01
8/15/1962	2.710E-01	5.890E-01
11/15/1962	2.200E-01	4.310E-01
2/15/1963	2.500E-01	4.670E-01
5/15/1963	2.480E-01	4.960E-01
8/15/1963	2.380E-01	4.320E-01
11/15/1963	2.520E-01	5.620E-01
2/15/1964	2.960E-01	8.100E-01
5/15/1964	2.490E-01	4.830E-01
8/15/1964	3.790E-01	1.668E+00
11/15/1964	3.340E-01	1.066E+00
2/15/1965	2.830E-01	7.570E-01
5/15/1965	3.480E-01	1.085E+00
8/15/1965	2.210E-01	4.170E-01
11/15/1965	2.660E-01	6.460E-01
2/15/1966	2.930E-01	8.210E-01
5/15/1966	2.370E-01	5.540E-01
8/15/1966	2.130E-01	4.300E-01
11/15/1966	2.520E-01	6.250E-01
2/15/1967	2.510E-01	6.220E-01
5/15/1967	2.400E-01	5.650E-01
8/15/1967	1.990E-01	4.130E-01
11/15/1967	2.360E-01	5.350E-01
2/15/1968	2.280E-01	5.260E-01
5/15/1968	2.050E-01	4.610E-01
8/15/1968	2.520E-01	5.850E-01
11/15/1968	2.780E-01	7.240E-01
2/15/1969	2.920E-01	6.920E-01
5/15/1969	2.660E-01	6.060E-01
8/15/1969	2.400E-01	5.190E-01
11/15/1969	2.640E-01	5.580E-01
2/15/1970	2.420E-01	5.150E-01
5/15/1970	1.650E-01	6.230E-01
8/15/1970	1.000E-01	4.230E-01
11/15/1970	1.200E-01	4.700E-01
2/15/1971	9.100E-02	3.660E-01
5/15/1971	5.500E-02	2.090E-01
8/15/1971	7.300E-02	2.930E-01
11/15/1971	6.100E-02	2.490E-01
2/15/1972	4.600E-02	3.980E-01
5/15/1972	4.600E-02	4.420E-01
8/15/1972	2.900E-02	1.990E-01
11/15/1972	2.800E-02	1.680E-01
2/15/1973	2.400E-02	1.450E-01
5/15/1973	3.300E-02	1.800E-01
8/15/1973	6.700E-02	3.050E-01
11/15/1973	6.100E-02	2.680E-01
2/15/1974	6.000E-02	2.240E-01
5/15/1974	4.900E-02	1.890E-01

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
8/15/1974	3.300E-02	1.440E-01
11/15/1974	1.600E-02	1.090E-01
2/15/1975	2.100E-02	1.040E-01
5/15/1975	1.900E-02	9.500E-02
8/15/1975	2.200E-02	2.000E-01
11/15/1975	1.500E-02	9.700E-02
2/15/1976	1.600E-02	1.440E-01
5/15/1976	2.100E-02	1.020E-01
8/15/1976	1.500E-02	1.040E-01
11/15/1976	4.300E-02	1.840E-01
2/15/1977	8.300E-02	2.620E-01
5/15/1977	9.200E-02	2.450E-01
8/15/1977	7.200E-02	1.900E-01
11/15/1977	6.200E-02	1.880E-01
2/15/1978	9.500E-02	3.070E-01
5/15/1978	6.000E-02	1.990E-01
8/15/1978	5.600E-02	2.010E-01
11/15/1978	3.300E-02	1.340E-01
2/15/1979	6.200E-02	2.370E-01
5/15/1979	1.300E-02	1.000E-01
8/15/1979	1.300E-02	8.700E-02
11/15/1979	2.900E-02	1.390E-01
2/15/1980	1.700E-02	1.060E-01
5/15/1980	1.700E-02	6.400E-02
8/15/1980	1.300E-02	6.100E-02
11/15/1980	4.000E-03	3.500E-02
2/15/1981	6.000E-03	3.700E-02
8/15/1981	5.000E-03	3.600E-02
11/15/1981	8.000E-03	5.600E-02
2/15/1983	1.000E-03	1.700E-02
8/15/1983	2.000E-03	1.600E-02
11/15/1983	4.000E-03	2.900E-02
2/15/1984	8.000E-03	5.000E-02
5/15/1984	5.300E-02	2.220E-01
8/15/1984	1.100E-02	7.100E-02
11/15/1984	5.400E-02	1.960E-01
2/15/1985	1.000E-02	8.000E-02

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
5/15/1985	2.500E-02	1.000E-01
8/15/1985	1.400E-02	8.100E-02
11/15/1985	1.700E-02	1.000E-01
2/15/1986	5.000E-03	3.300E-02
5/15/1986	4.000E-03	3.800E-02
8/15/1986	7.000E-03	3.800E-02
11/15/1986	8.000E-03	4.200E-02
2/15/1987	4.000E-03	3.000E-02
5/15/1987	5.000E-03	3.600E-02
8/15/1987	8.000E-03	5.100E-02
11/15/1987	8.000E-03	5.000E-02
2/15/1988	3.000E-03	3.200E-02
5/15/1988	2.000E-03	3.300E-02
8/15/1988	5.000E-03	3.400E-02
11/15/1988	6.000E-03	3.800E-02
2/15/1989	4.230E-03	3.460E-02
5/15/1989	1.060E-02	7.150E-02
8/15/1989	2.880E-03	6.550E-02
11/15/1989	6.970E-03	8.480E-02
2/15/1990	2.280E-03	2.830E-02
5/15/1990	3.970E-03	3.090E-02
8/15/1990	2.300E-03	3.070E-02
11/15/1990	3.970E-03	3.760E-02
2/15/1991	4.120E-03	2.560E-02
5/15/1991	1.180E-03	6.270E-02
8/15/1991	7.960E-03	9.720E-02
11/15/1991	6.950E-04	3.900E-02
2/15/1992	5.110E-04	1.910E-02
5/15/1992	2.780E-03	2.020E-02
8/15/1992	5.150E-03	3.630E-02
11/15/1992	9.490E-03	1.070E-01
7/1/1993	4.130E-03	6.460E-02
2/15/1994	4.500E-04	5.460E-03
5/15/1994	5.760E-04	5.650E-03
8/15/1994	6.550E-04	7.430E-03
11/15/1994	8.510E-04	6.880E-03
7/1/1995	8.570E-04	8.150E-03

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
2/15/1996	1.050E-03	9.700E-03
5/15/1996	8.740E-04	7.070E-03
8/15/1996	4.900E-04	4.330E-03
11/15/1996	3.640E-04	5.270E-03
2/15/1997	6.430E-04	7.300E-03
5/15/1997	5.280E-04	6.210E-03
8/15/1997	4.840E-04	6.610E-03
11/15/1997	2.300E-04	1.210E-03
7/1/1998	3.940E-04	3.180E-03
7/1/1999	3.580E-04	4.280E-03
2/15/2000	3.320E-04	2.790E-03
5/15/2000	5.430E-04	3.320E-03
8/15/2000	5.230E-04	3.410E-03
11/15/2000	4.300E-04	2.950E-03
2/15/2001	3.250E-04	3.030E-03
5/15/2001	2.520E-04	4.150E-03

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
8/15/2001	3.940E-04	4.340E-03
11/15/2001	4.190E-04	3.500E-03
2/15/2002	3.860E-04	3.560E-03
5/15/2002	4.050E-04	3.570E-03
8/15/2002	2.420E-04	2.700E-03
11/15/2002	3.630E-04	2.950E-03
2/15/2003	2.320E-04	2.140E-03
5/15/2003	4.720E-04	2.970E-03
8/15/2003	3.150E-04	2.630E-03
11/15/2003	3.050E-04	2.620E-03
2/15/2004	2.840E-04	6.720E-03
5/15/2004	2.690E-04	2.550E-03
8/15/2004	3.620E-04	2.680E-03
11/15/2004	1.790E-04	2.000E-03
7/1/2005	2.040E-04	3.090E-03

a. Very large results for badge 395943 excluded from 1964-1965; badges 164455 and 184168 excluded from quarter 3, 1971; 164455 and 184169 excluded from quarter 4, 1971; badge 184106 excluded from quarter 2, 1976.
b. Results for quarter 2, 1981, all of 1982, and quarter 2, 1983 were not used in calculations because there are too few results.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

Table D-3. Americium-241 lung count bioassay data.[a]

Effective sample date	50th percentile (dpm)	84th percentile (dpm)
7/1/1972	0.15	6.66
2/15/1973	11.10	130.98
5/15/1973	22.20	237.54
8/15/1973	55.50	417.36
11/15/1973	19.98	210.90
2/15/1974	11.10	148.74
5/15/1974	15.54	177.60
8/15/1974	15.54	175.38
11/15/1974	15.54	175.38
2/15/1975	38.41	333.00
5/15/1975	59.94	399.60
8/15/1975	87.02	541.68
11/15/1975	105.89	617.16
2/15/1976	95.46	579.42
5/15/1976	97.68	563.88
8/15/1976	37.74	295.26
11/15/1976	26.64	246.42
2/15/1977	22.20	215.34
5/15/1977	17.76	182.04
8/15/1977	15.54	135.42
11/15/1977	8.88	113.22
2/15/1978	17.76	184.26
5/15/1978	15.54	155.40
8/15/1978	15.54	146.52
11/15/1978	8.88	99.90
7/1/1979	26.64	239.76
2/15/1980	57.72	432.90
5/15/1980	44.40	352.98
8/15/1980	46.62	379.62
11/15/1980	59.94	459.54
2/15/1981	39.96	335.22
5/15/1981	35.52	310.80
8/15/1981	35.52	306.36
11/15/1981	35.52	301.92
2/15/1982	28.86	279.72
5/15/1982	24.42	246.42
8/15/1982	22.20	226.44
11/15/1982	19.98	179.82
2/15/1983	13.32	146.52
5/15/1983	4.44	68.82
8/15/1983	11.10	122.10
11/15/1983	17.76	139.86
2/15/1984	11.10	128.76
5/15/1984	13.32	128.76
8/15/1984	11.10	119.88
11/15/1984	17.76	148.74
2/15/1985	8.88	93.24
5/15/1985	11.10	113.22
8/15/1985	6.66	77.70
11/15/1985	6.66	82.14
2/15/1986	8.88	108.78
5/15/1986	15.54	119.88
8/15/1986	11.10	126.54
11/15/1986	8.88	95.46
2/15/1987	17.76	159.84
5/15/1987	11.10	113.22
8/15/1987	19.98	202.02
11/15/1987	19.98	159.84
2/15/1988	13.32	135.42
5/15/1988	17.76	162.06
8/15/1988	11.10	95.46
11/15/1988	8.88	93.24
2/15/1989	31.08	208.68

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

Table D-4. Derived uranium intake rates (95th-percentile), dpm/day, type F material, 1953 to 2005.

Period	50th percentile	84th percentile	GSD	95th percentile
1953–1958	13.37	37.91	2.84	74.2
1959	19.7	61.99	3.15	130
1960	27.23	94.74	3.48	212
1961	21.62	65.97	3.05	135
1962	16.27	65.97	4.05	163
1963	16.27	44.36	2.73	84.7
1964	27.26	80.39	2.95	161
1965–1976	27.26	66.26	2.43	118
1977–1988	0.597	3.003	5.03	8.52
1989–1993	0.171	1.365	7.98	5.21
1994–2005	0.038	0.370	9.86	1.64

Table D-5. Derived uranium intake rates (95th-percentile), dpm/day, type M material, 1953 to 2005.

Period	50th percentile	84th percentile	GSD	95th percentile
1953–1958	54.75	154.8	2.83	303
1959–1960	102.7	347.6	3.38	763
1961–1963	71.85	234.2	3.26	502
1964–1976	112.8	284.2	2.52	516
1977–1988	2.443	6.252	2.56	11.5
1989–1993	0.753	5.821	7.73	21.8
1994–2005	0.154	1.532	9.94	6.74

Table D-6. Derived uranium intake rates (95th-percentile), dpm/day, type S material, 1953 to 2005.

Period	50th percentile	84th percentile	GSD	95th percentile
1953–1958	936.9	2,676	2.86	5,266
1959–1960	2,768	9,300	3.36	20,322
1961–1963	1,680	5,439	3.24	11,604
1964–1976	1,630	4,086	2.51	7,391
1977–1988	29.12	157	5.39	465
1989–1993	15.11	115.00	7.61	426
1994–2005	2.06	21.96	10.65	101

Table D-7. Derived type M plutonium intake rates (95th-percentile), dpm/day, 1952 to 2005.

Period	50th percentile	84th percentile	GSD	95th percentile
1952–1961	121	357.2	2.95	718
1962–1969	43.5	106.5	2.45	190
1970–1979	7.05	29.81	4.23	75.6
1980–1988	1.622	8.908	5.49	26.7
1989–1993	0.9542	11.66	12.22	58.6
1994–2005	0.05418	0.526	9.71	2.28

Table D-8. Derived type S ^{239}Pu intake rates (95th percentile), dpm/day, 1952 to 2005.

Period	50th percentile	84th percentile	GSD	95th percentile
1952–1961	1,925	5,628	2.92	11,243
1962–1971	518.3	1,362	2.63	2,540
1972–1976	20.59	144.2	7.00	506
1977–1982	10.88	94.56	8.69	381
1983–1988	4.531	43.85	9.68	190
1989–1993	13.7	104.0	7.59	384
1994–2005	0.7966	7.745	9.72	33.6

Table D-9. Derived type SS ^{239}Pu intake rates (95th percentile), dpm/day, 1952 to 2005.

Period	50th percentile	84th percentile	GSD	95th percentile[a]
1952–1961	16,400	48,500	2.96	99,933
1962–1971	4,470	11,600	2.60	27,238
1972–1976	67.2	465	6.92	1,619
1977–1982	29.9	261	8.74	1,057
1983–1988	12.8	123	9.60	527
1989–1993	130	933	7.18	3326
1994–2005	7.32	71.1	9.71	308

a. A minimum GSD of 3.00 was used to calculated the 95th percentile.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

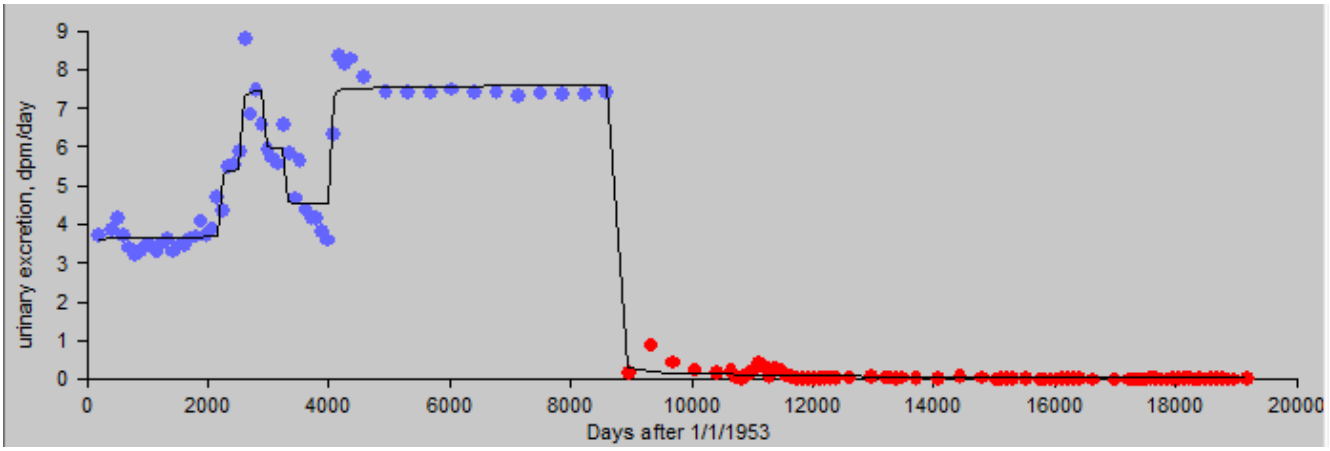

Figure D-1. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1976, 50th-percentile, type F.

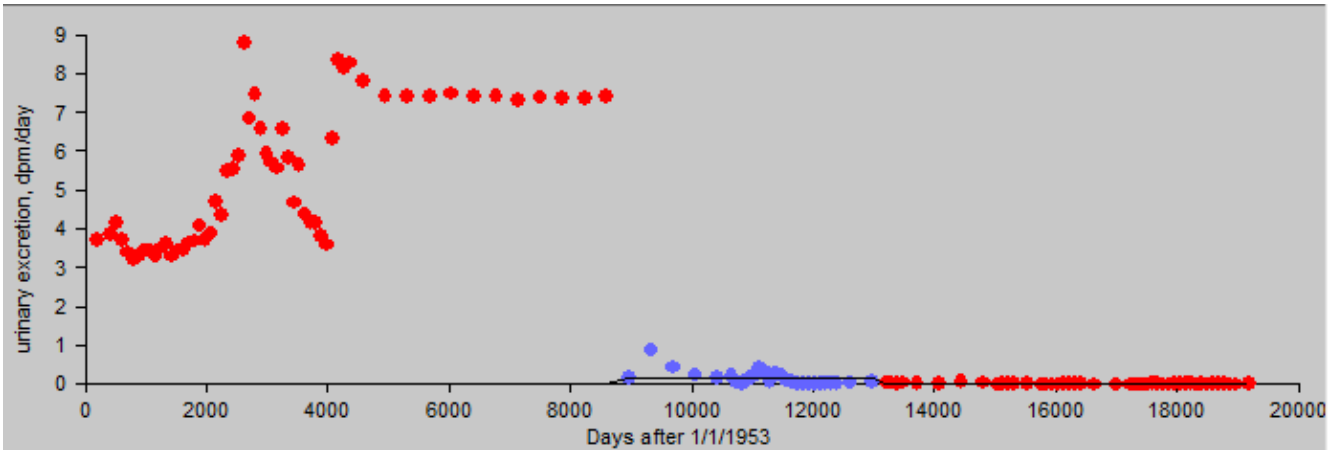

Figure D-2. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1977 to 12/31/1988, 50th-percentile, type F.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

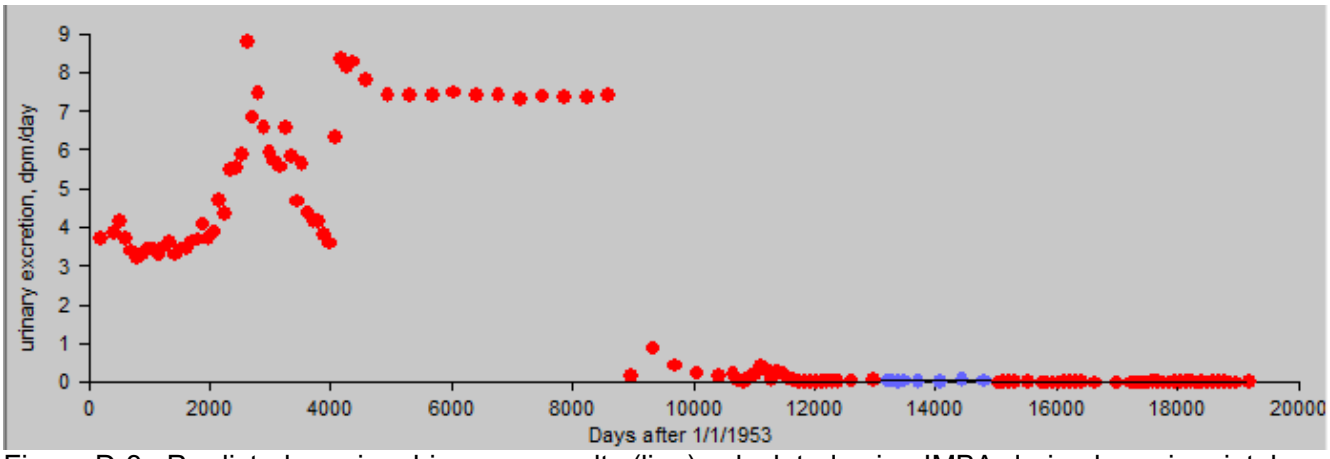

Figure D-3. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1989 to 12/31/1993, 50th-percentile, type F.

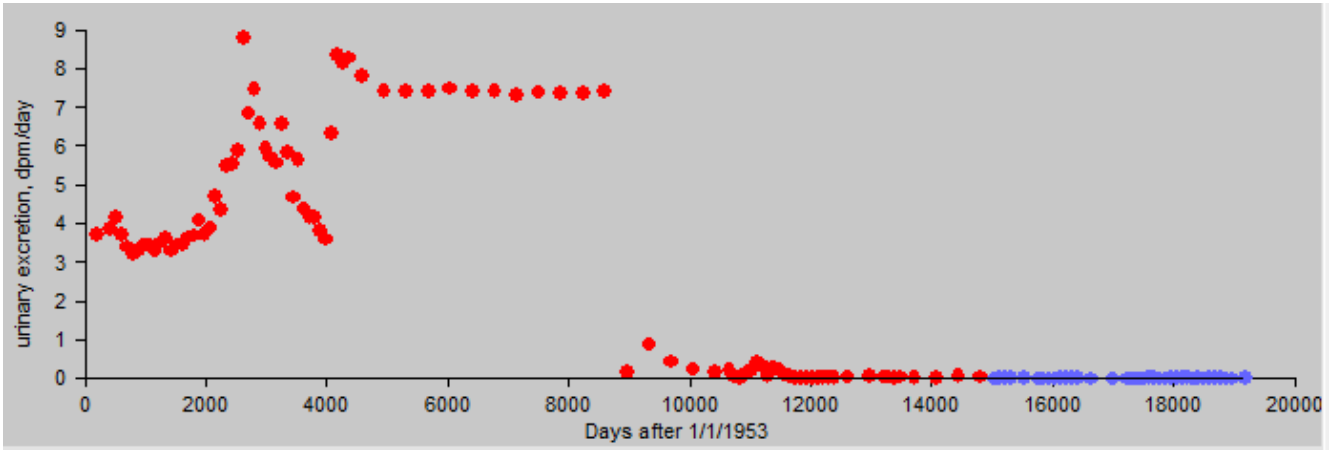

Figure D-4. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1994 to 12/31/2005, 50th-percentile, type F.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

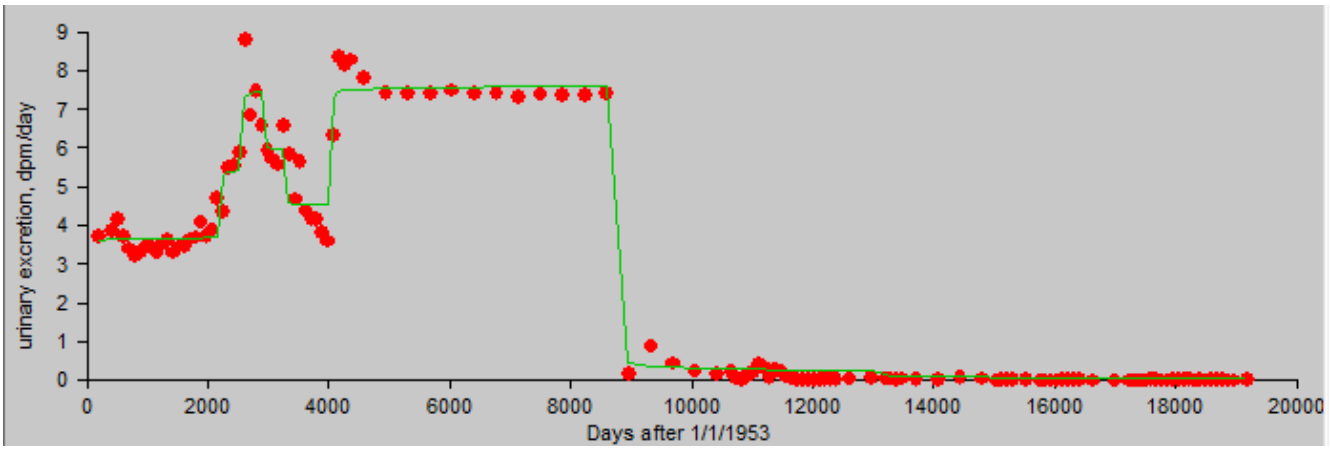

Figure D-5. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots) from all intakes 1/1/1953 to 12/31/2005, 50th-percentile, type F.

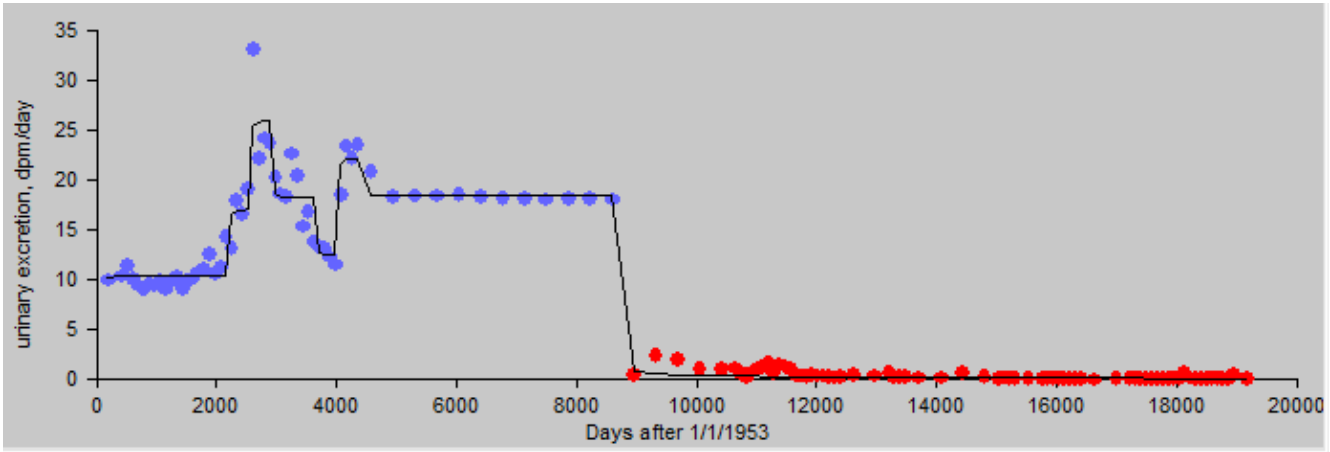

Figure D-6. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1976, 84th-percentile, type F.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

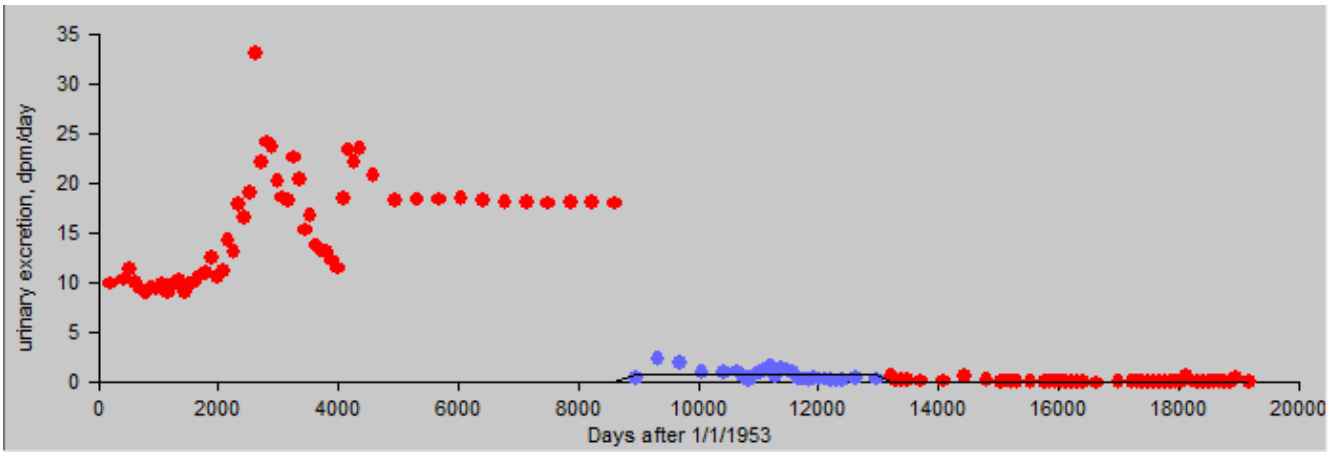

Figure D-7. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1977 to 12/31/1988, 84th-percentile, type F.

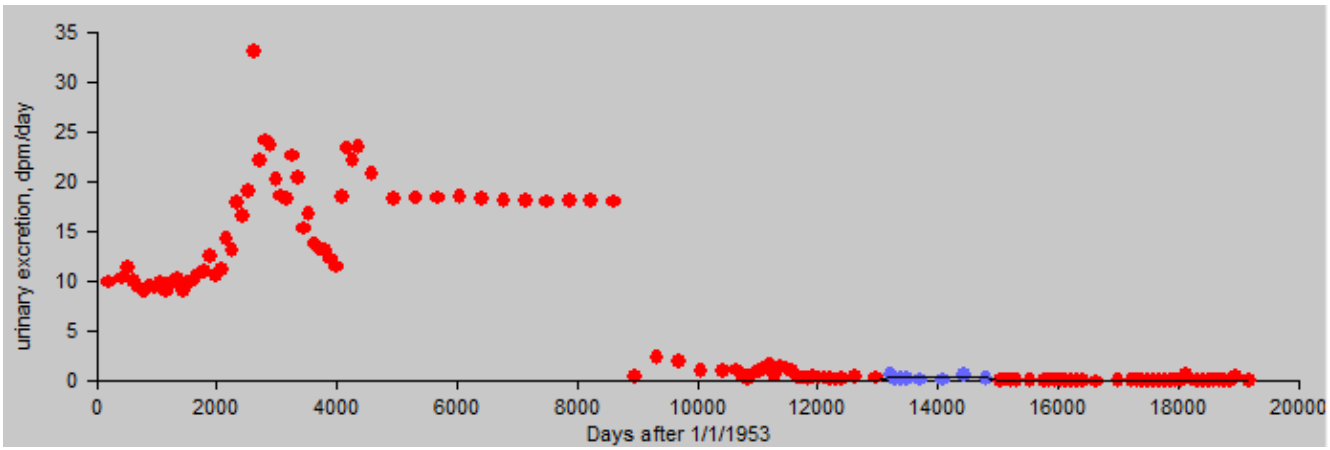

Figure D-8. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1989 to 12/31/1993, 84th-percentile, type F.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

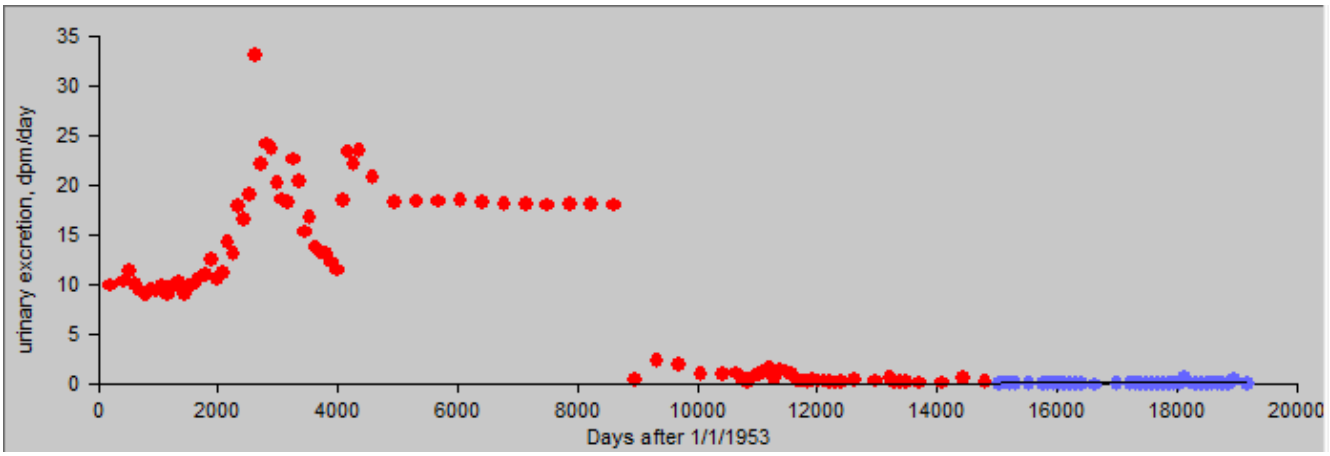

Figure D-9. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1994 to 12/31/2005, 84th-percentile, type F.

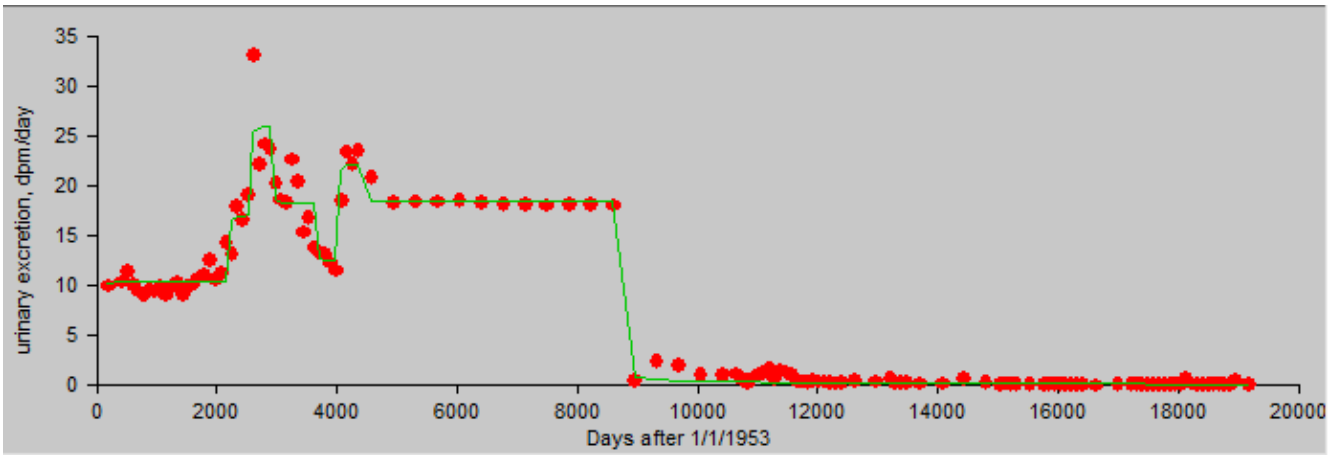

Figure D-10. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots) from all intakes 1/1/1953 to 12/31/2005, 84th-percentile, type F.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

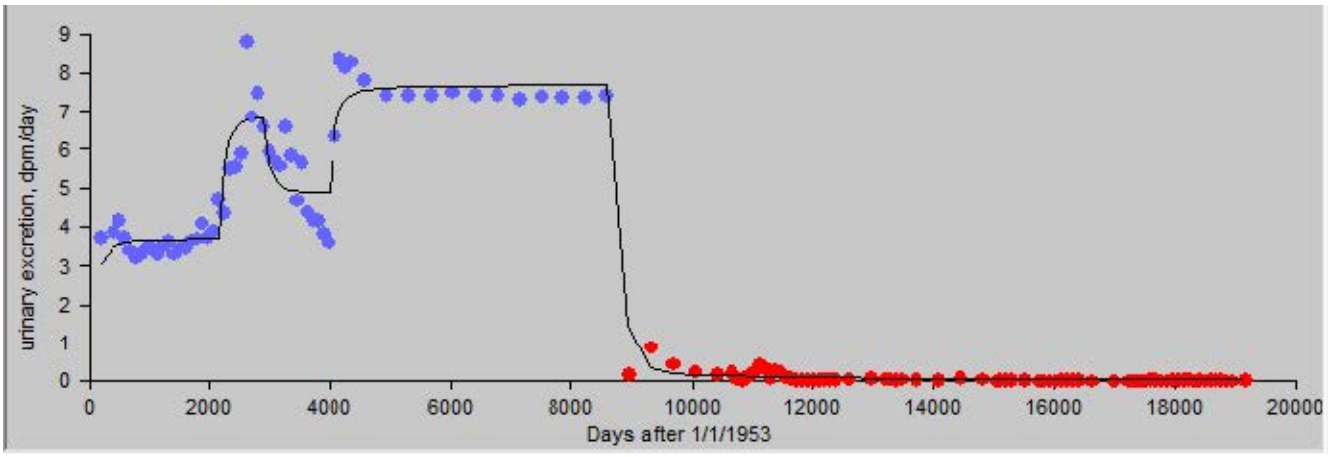

Figure D-11. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1976, 50th-percentile, type M.

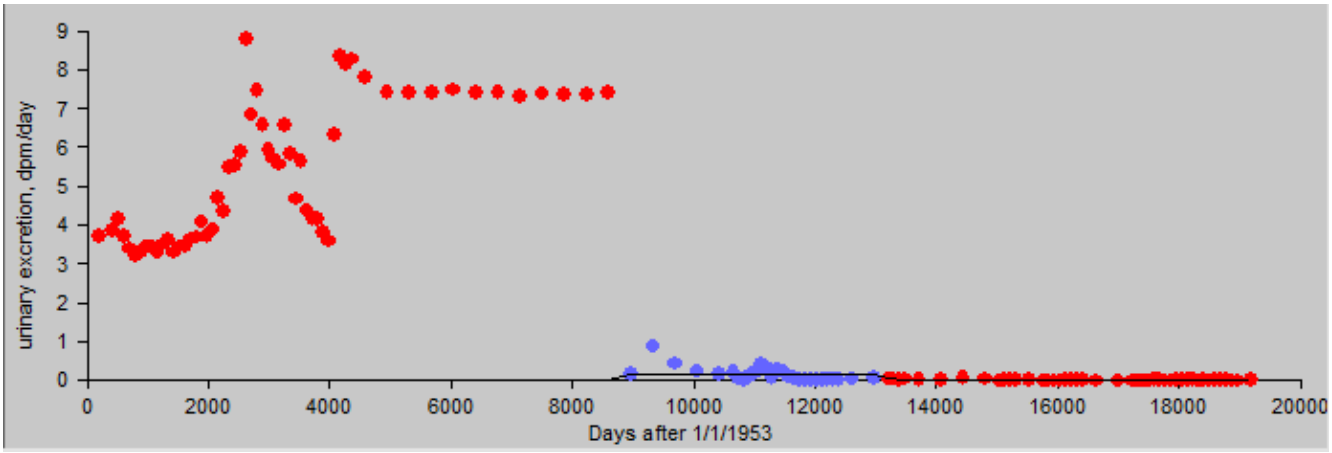

Figure D-12. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1977 to 12/31/1988, 50th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

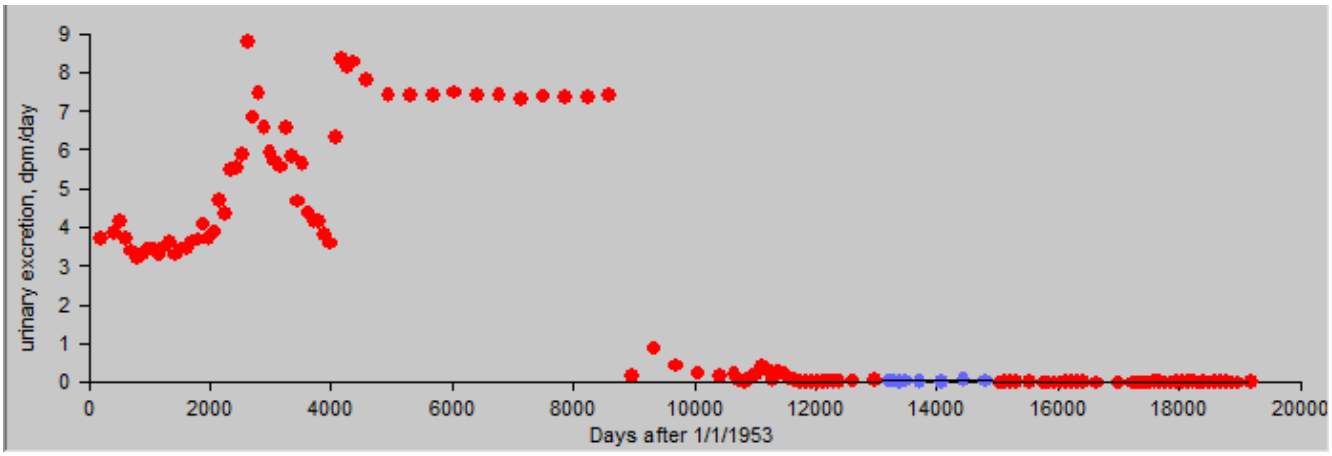

Figure D-13. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1989 to 12/31/1993, 50th-percentile, type M.

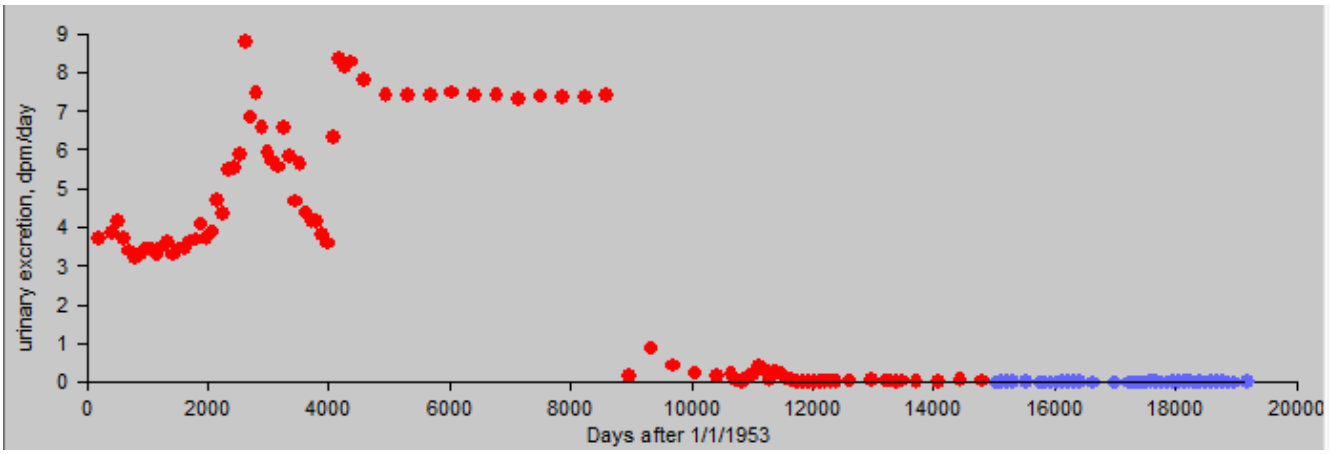

Figure D-14. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1994 to 12/31/2005, 50th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

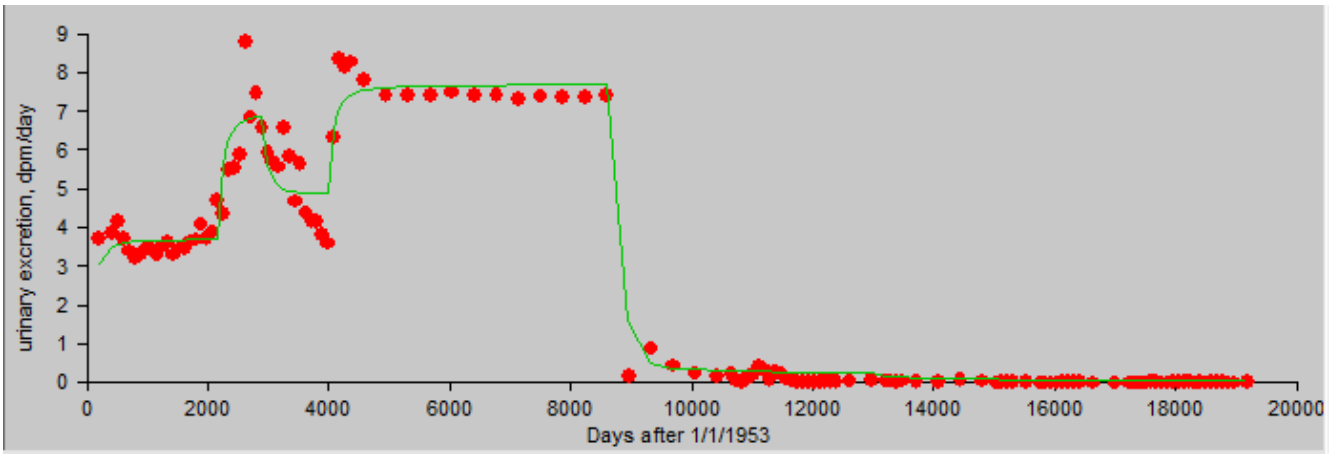

Figure D-15. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots) from all intakes 1/1/1953 to 12/31/2005, 50th-percentile, type M.

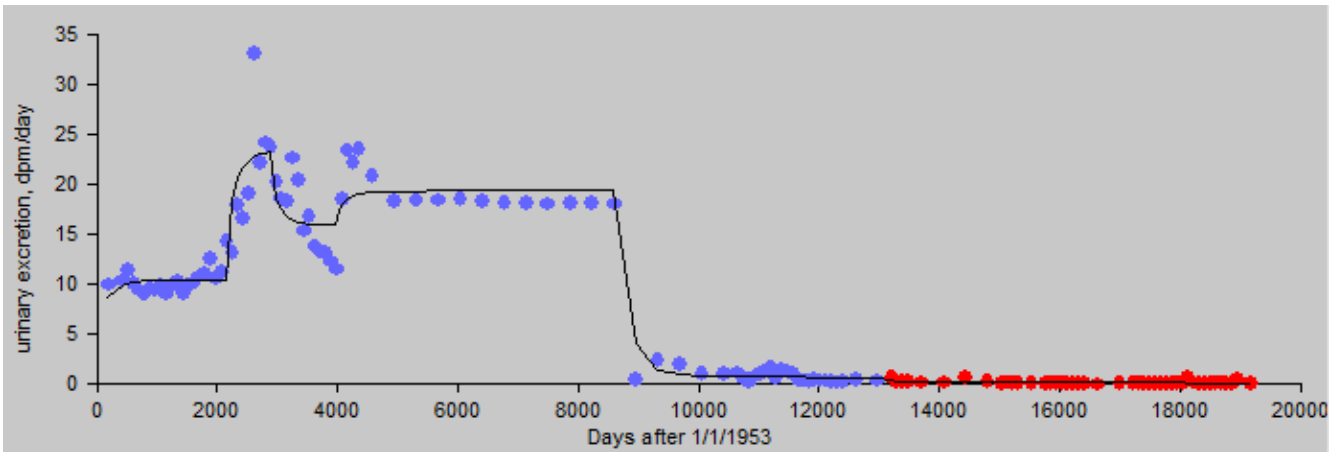

Figure D-16. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1988, 84th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

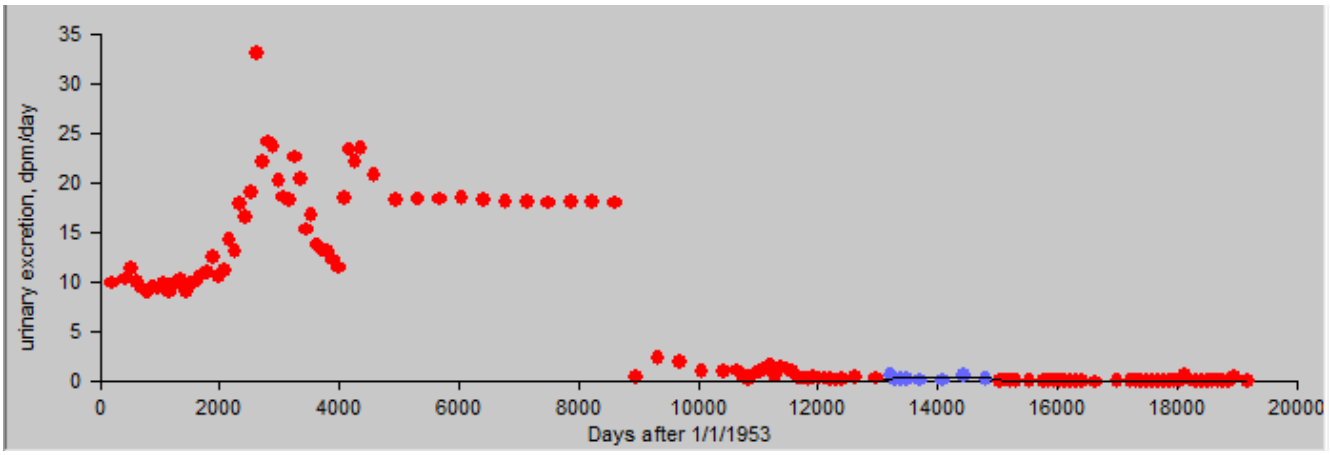

Figure D-17. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1989 to 12/31/1993, 84th-percentile, type M.

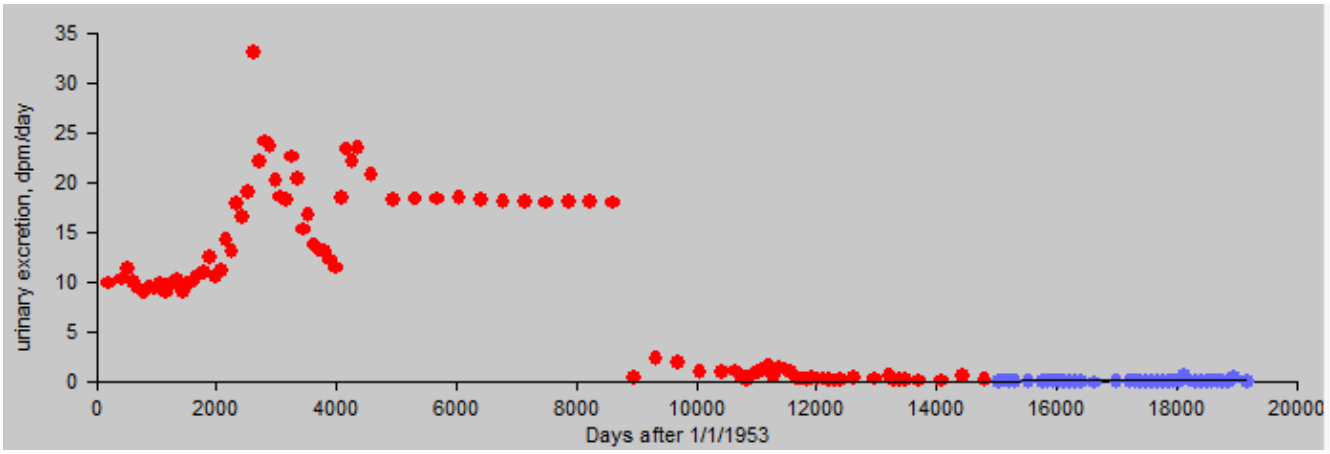

Figure D-18. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1994 to 12/31/2005, 84th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

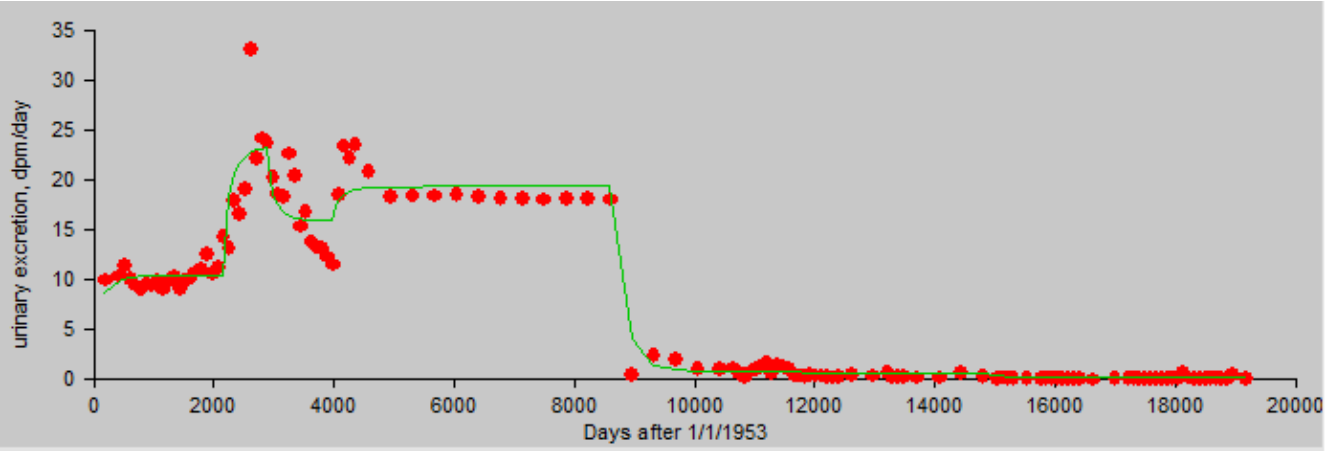

Figure D-19. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots) from all intakes 1/1/1953 to 12/31/2005, 84th-percentile, type M.

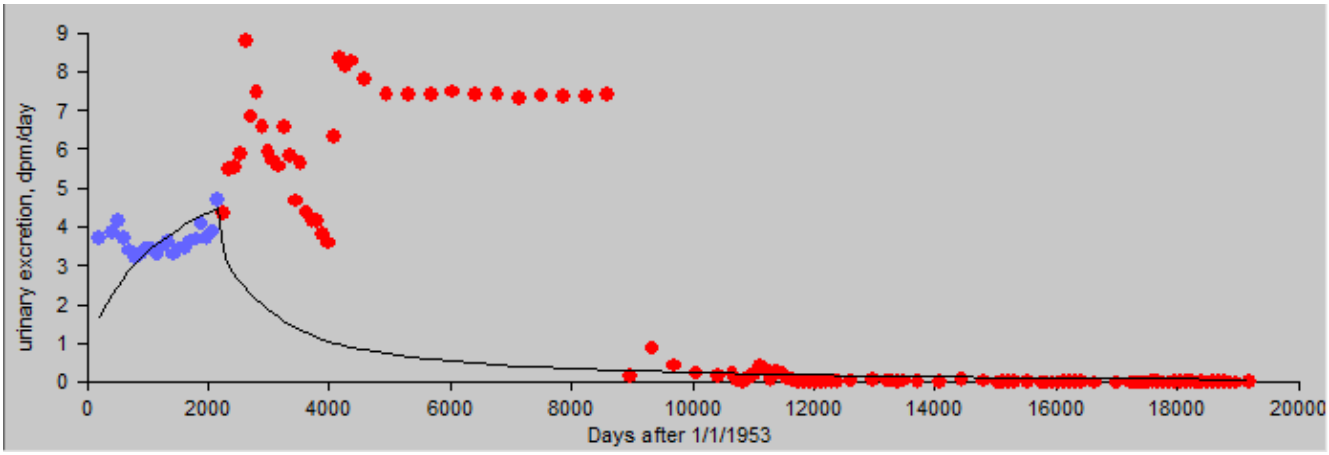

Figure D-20. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1958, 50th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

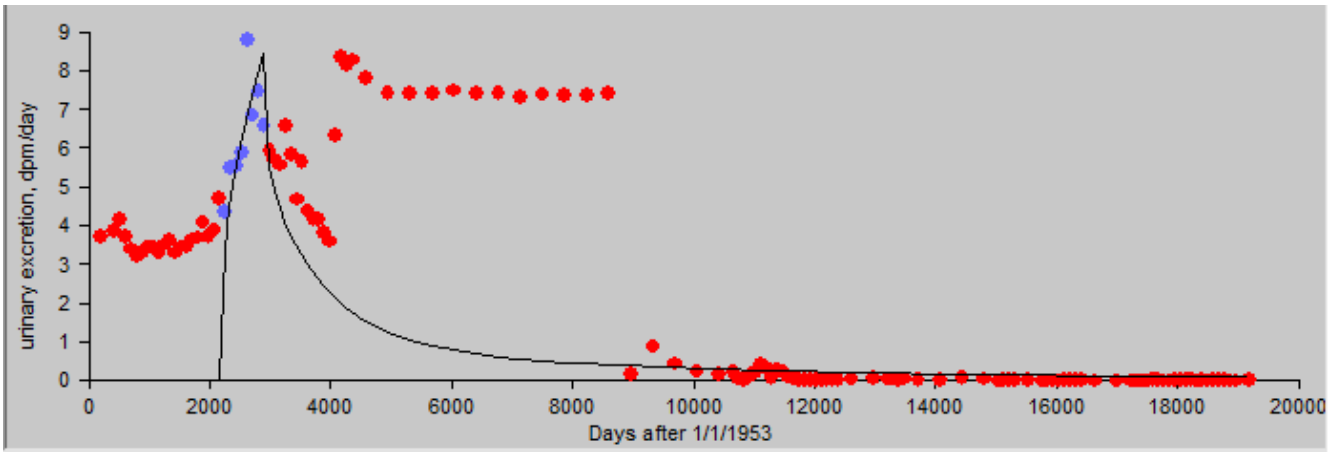

Figure D-21. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1959 to 12/31/1960, 50th-percentile, type S.

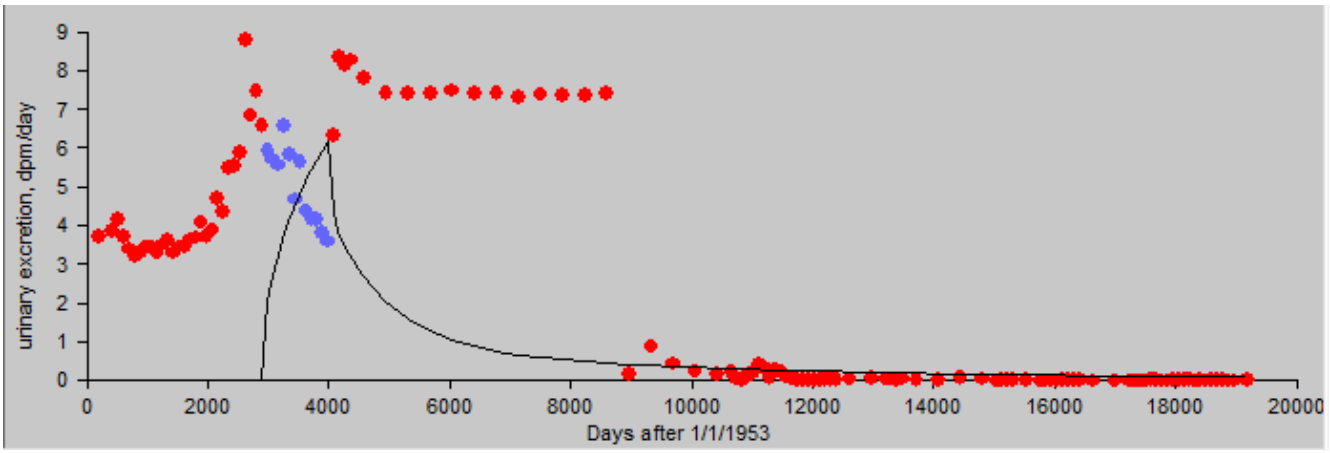

Figure D-22. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1961 to 12/31/1963, 50th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

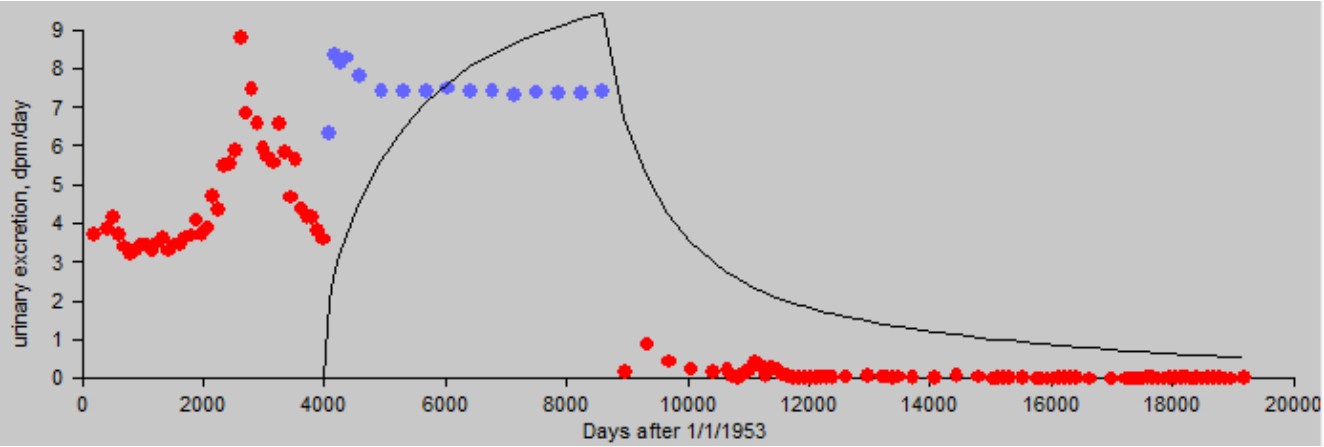

Figure D-23. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1964 to 12/31/1976, 50th-percentile, type S.

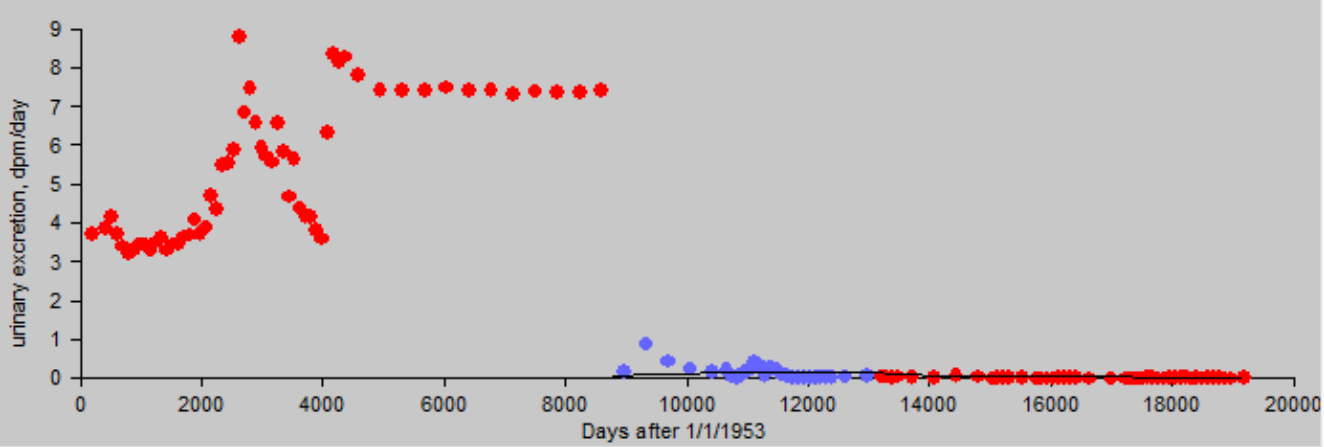

Figure D-24. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1977 to 12/31/1988, 50th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

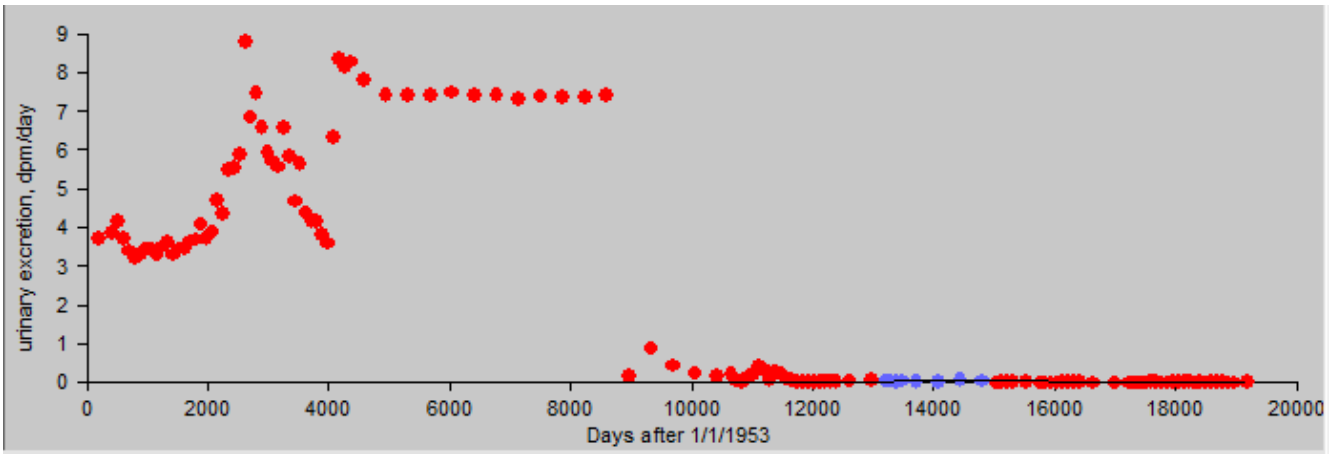

Figure D-25. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1989 to 12/31/1993, 50th-percentile, type S.

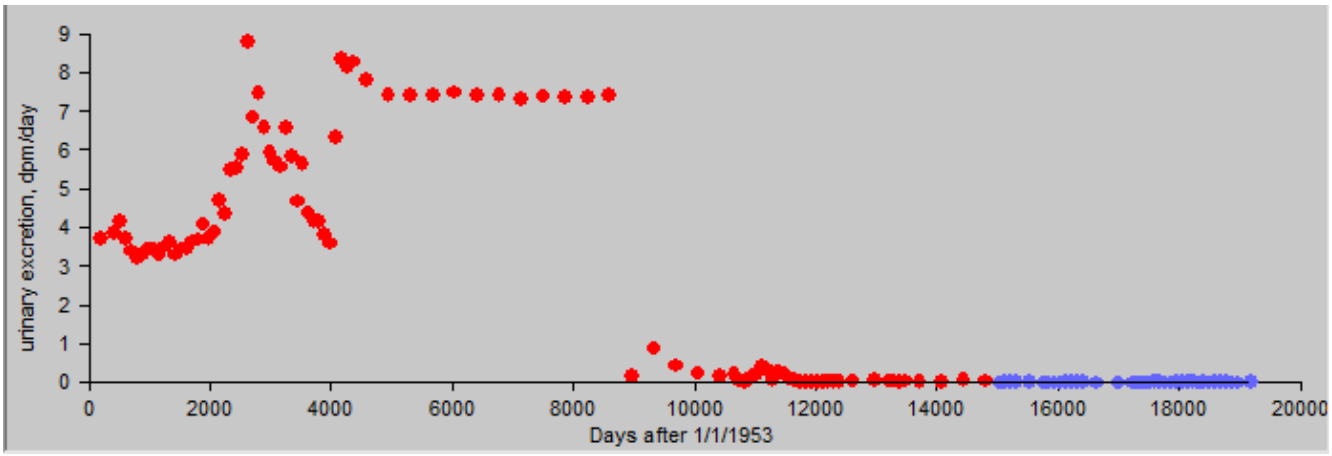

Figure D-26. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1994 to 12/31/2005, 50th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

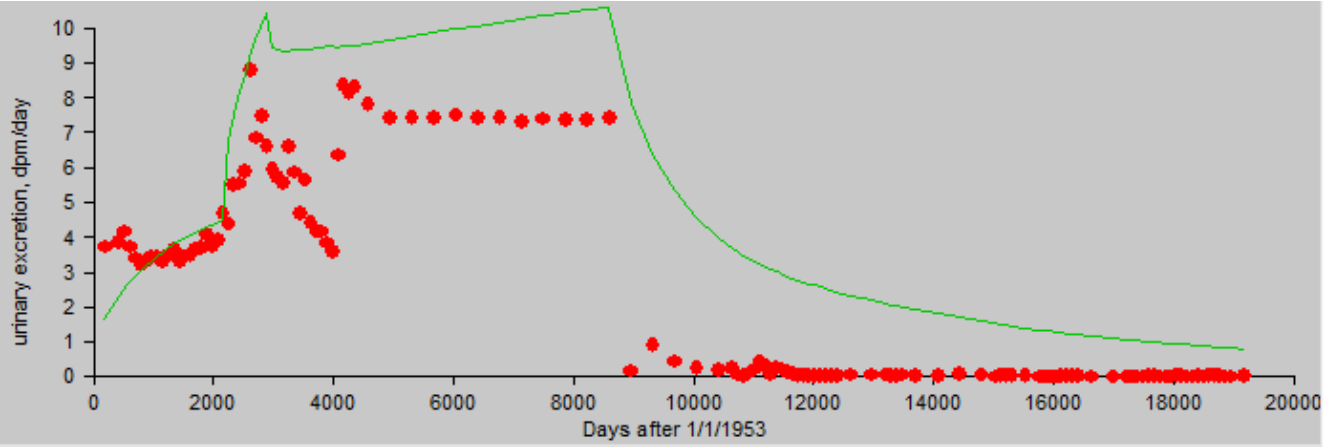

Figure D-27. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots) from all intakes 1/1/1953 to 12/31/2005, 50th-percentile, type S.

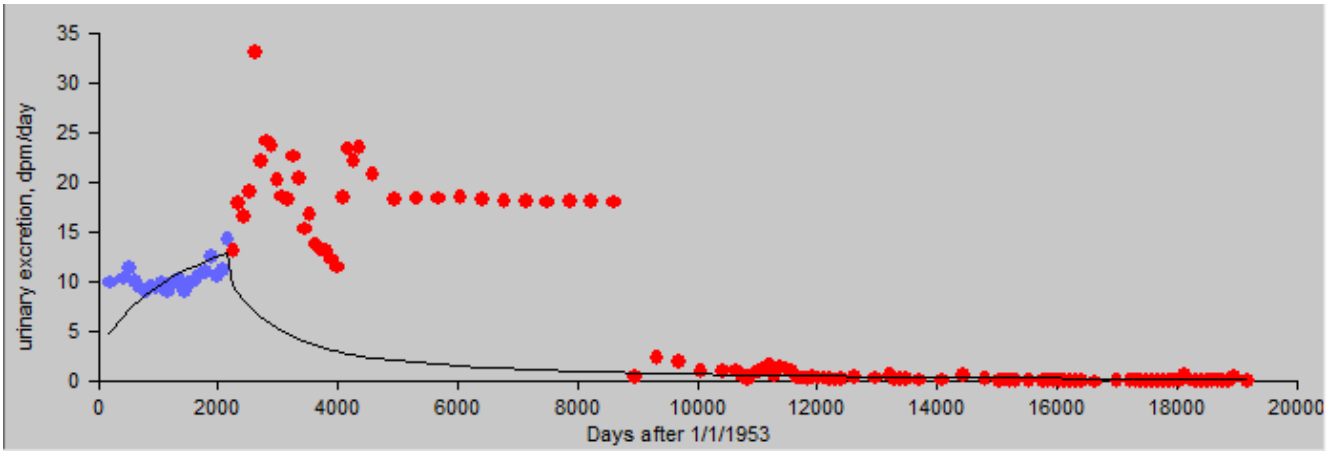

Figure D-28. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1958, 84th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

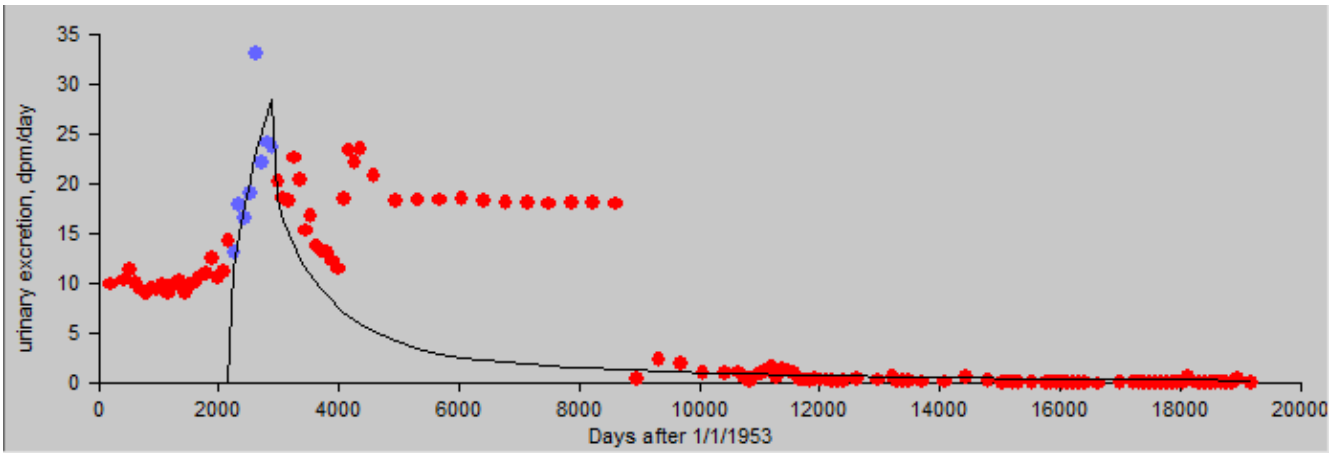

Figure D-29. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1959 to 12/31/1960, 84th-percentile, type S.

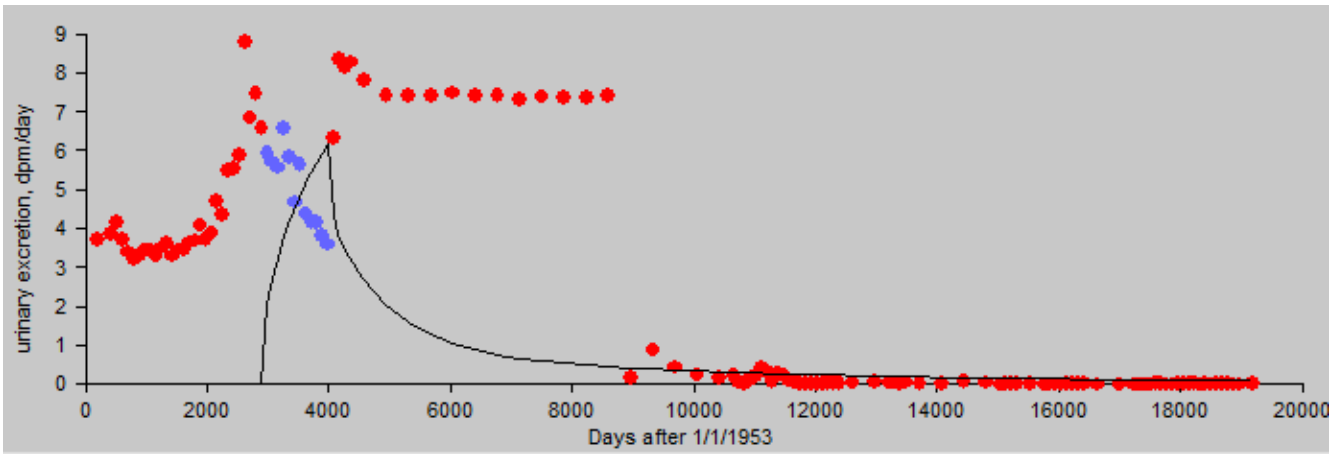

Figure D-30. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1961 to 12/31/1963, 84th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

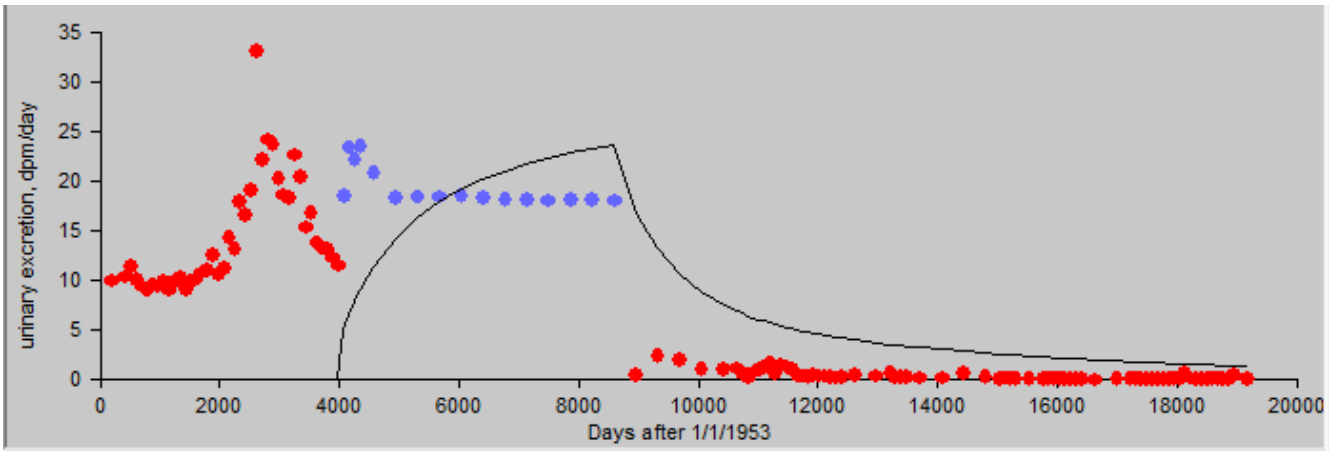

Figure D-31. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1964 to 12/31/1976, 84th-percentile, type S.

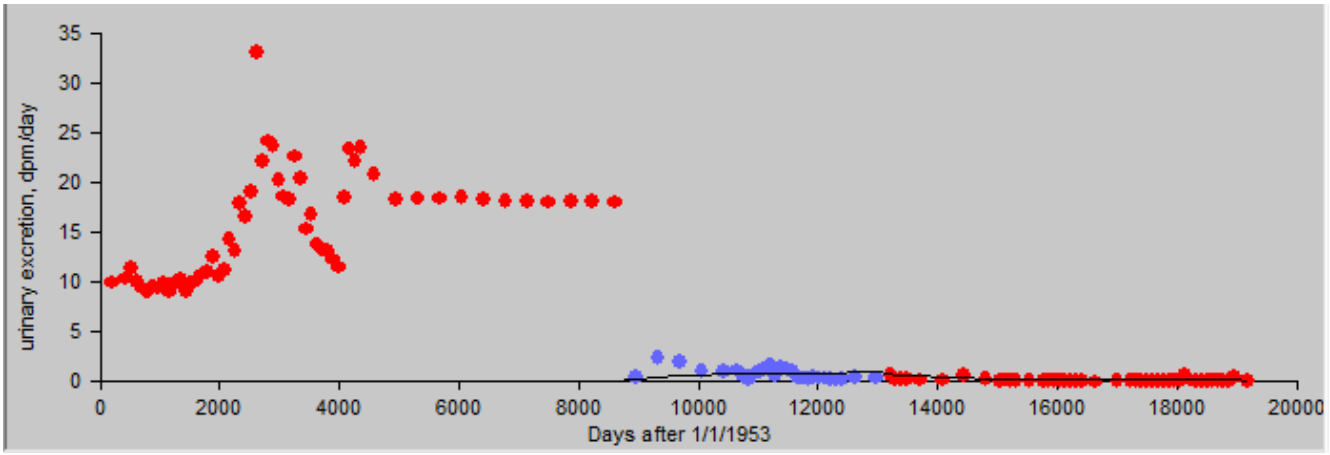

Figure D-12. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1977 to 12/31/1988, 50th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

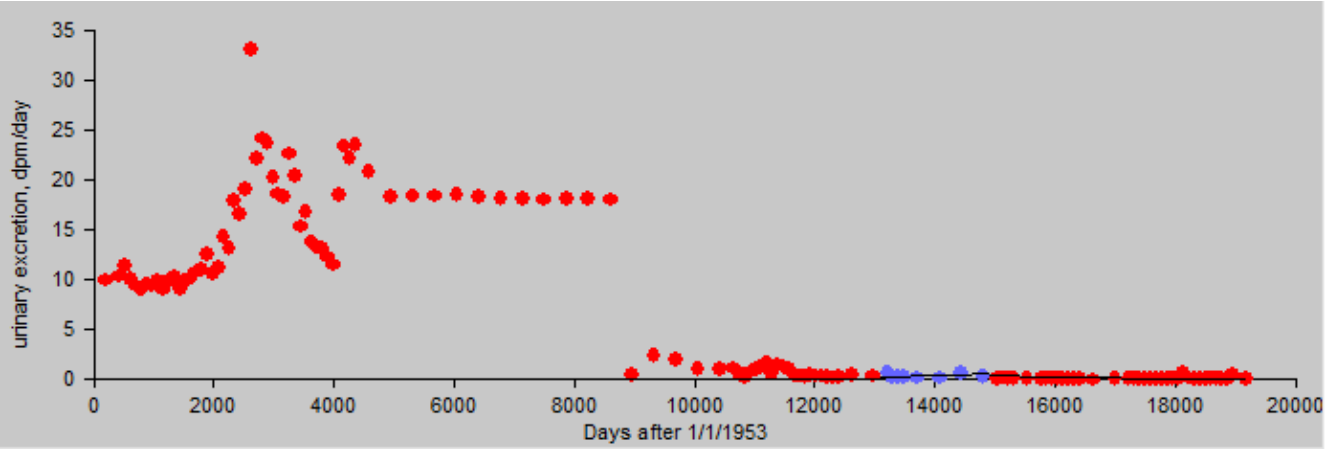

Figure D-33. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1989 to 12/31/1993, 84th-percentile, type S.

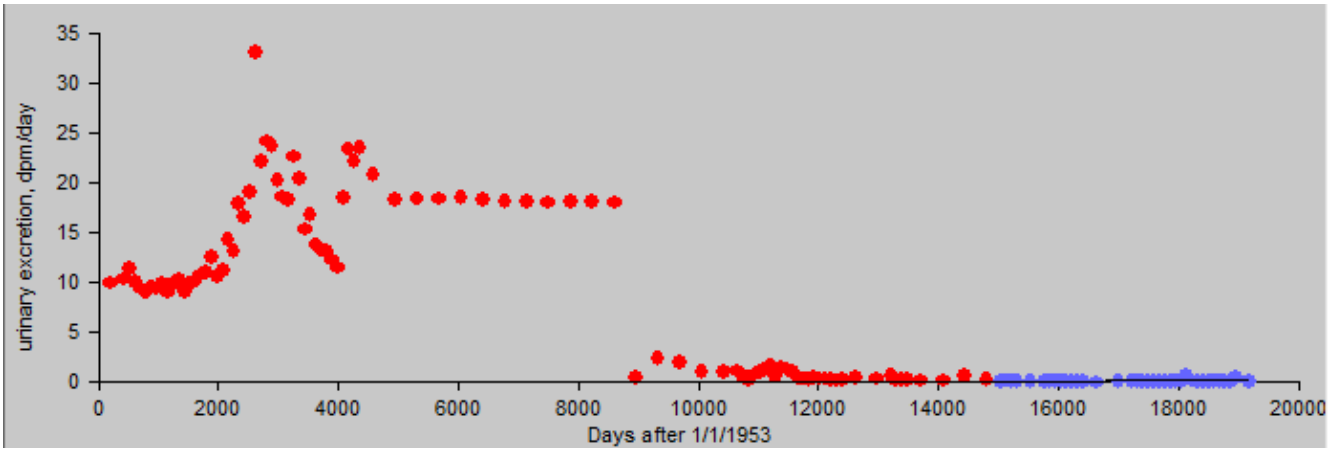

Figure D-34. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1994 to 12/31/2005, 84th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

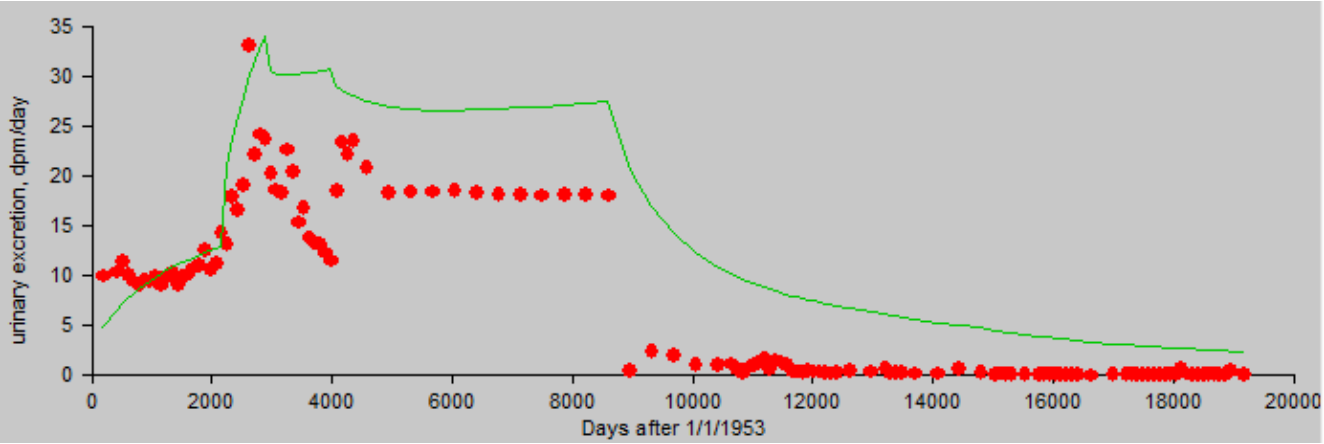

Figure D-35. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots) from all intakes 1/1/1953 to 12/31/2005, 84th-percentile, type S.

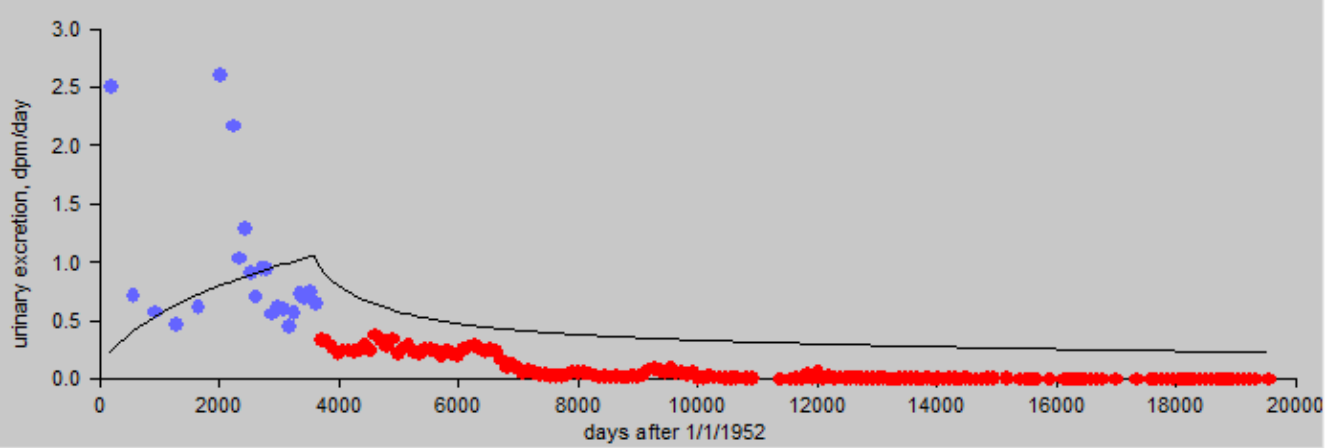

Figure D-36. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/1961, 50th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

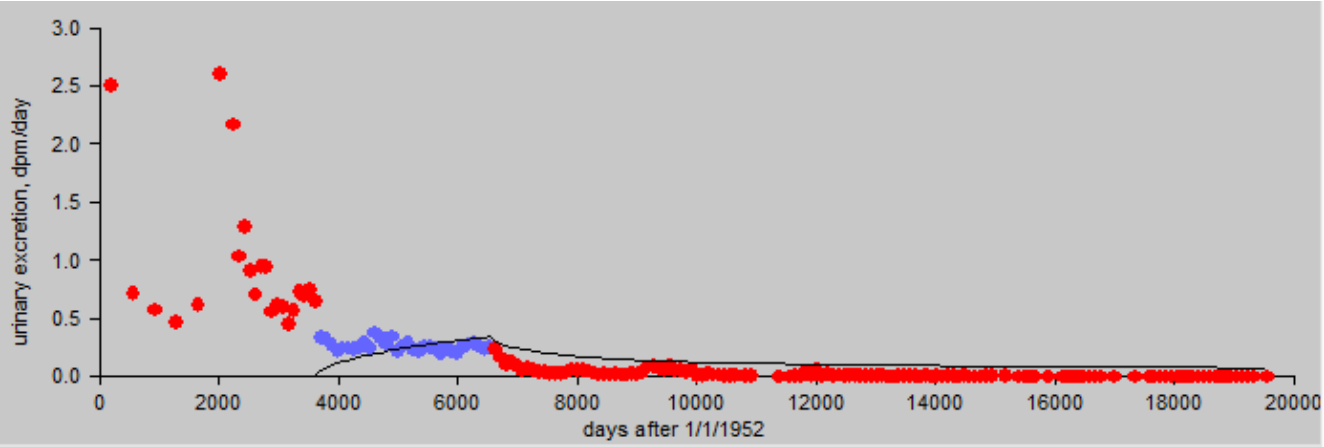

Figure D-37. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1962 to 12/31/1969, 50th-percentile, type M.

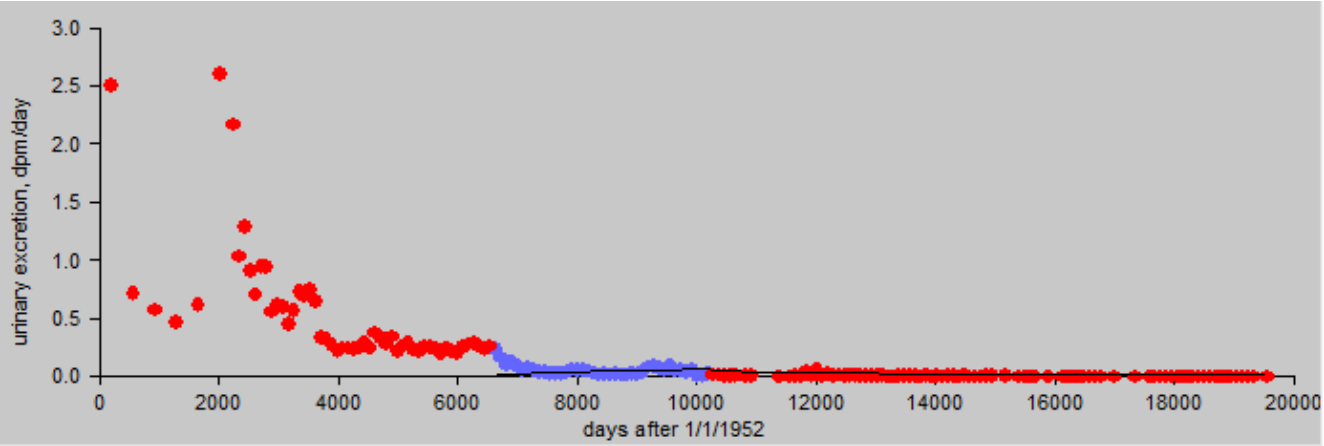

Figure D-38. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1970 to 12/31/1979, 50th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

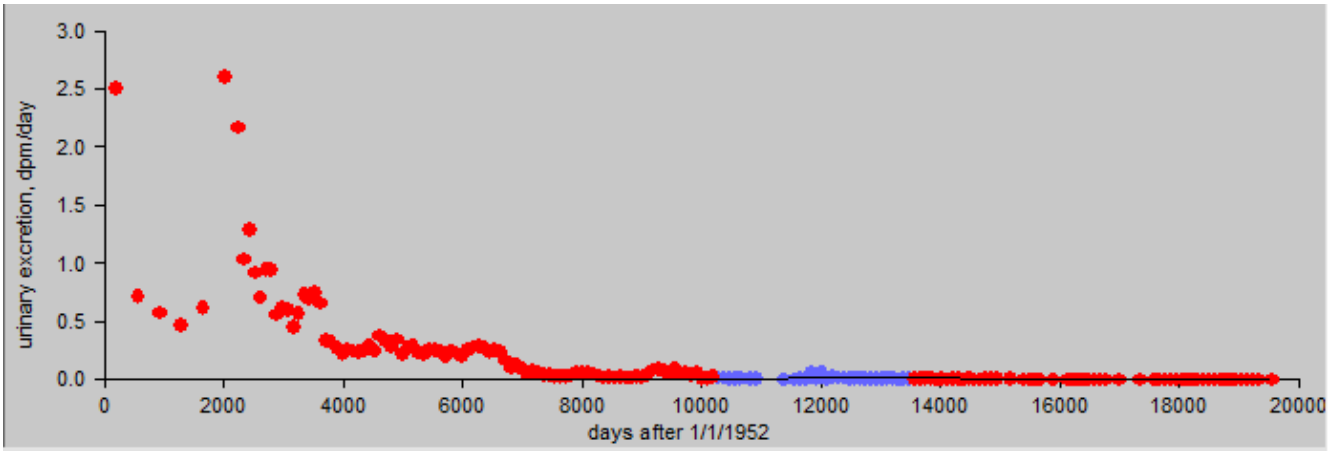

Figure D-39. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1980 to 12/31/1988, 50th-percentile, type M.

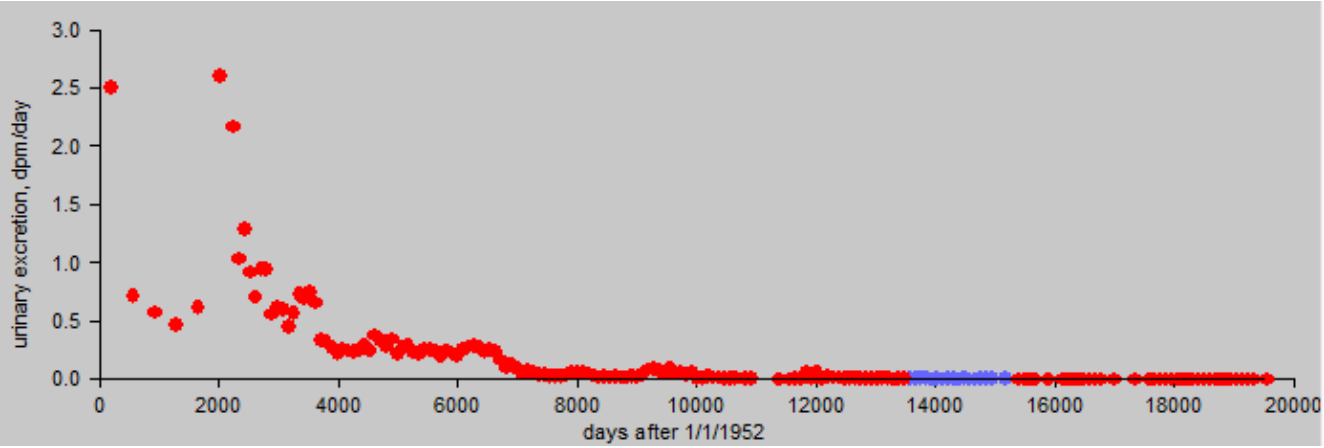

Figure D-40. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1989 to 12/31/1993, 50th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

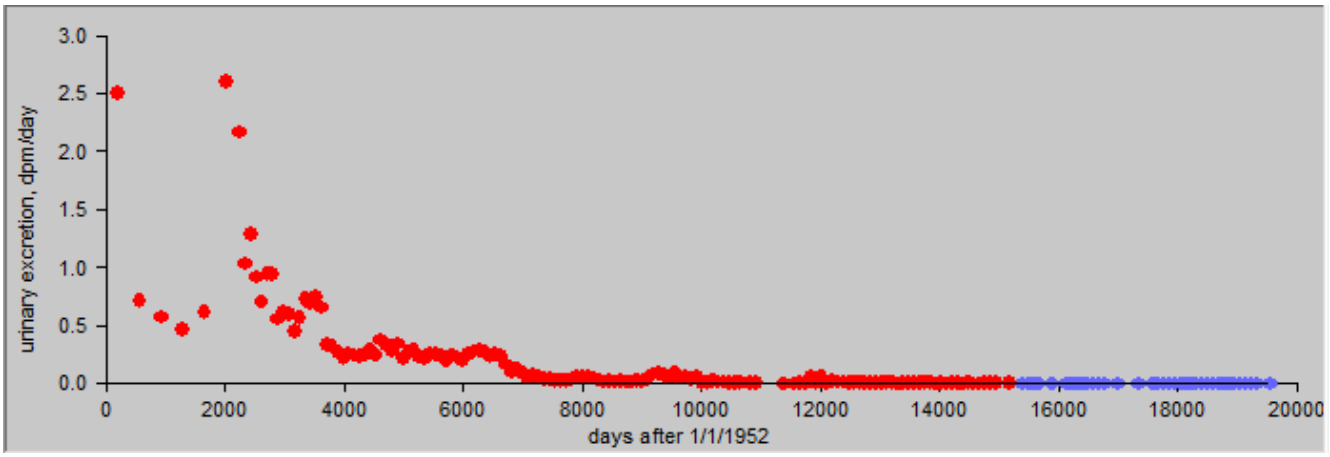

Figure D-41. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1994 to 12/31/2005, 50th-percentile, type M.

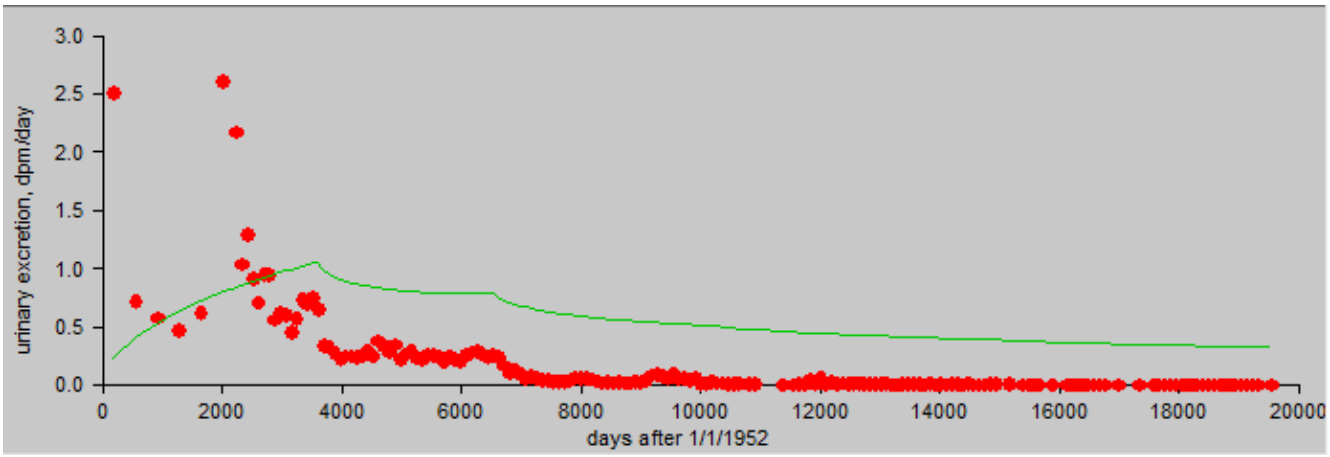

Figure D-42. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), from all intakes 1/1/1952 to 12/31/2005, 50th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

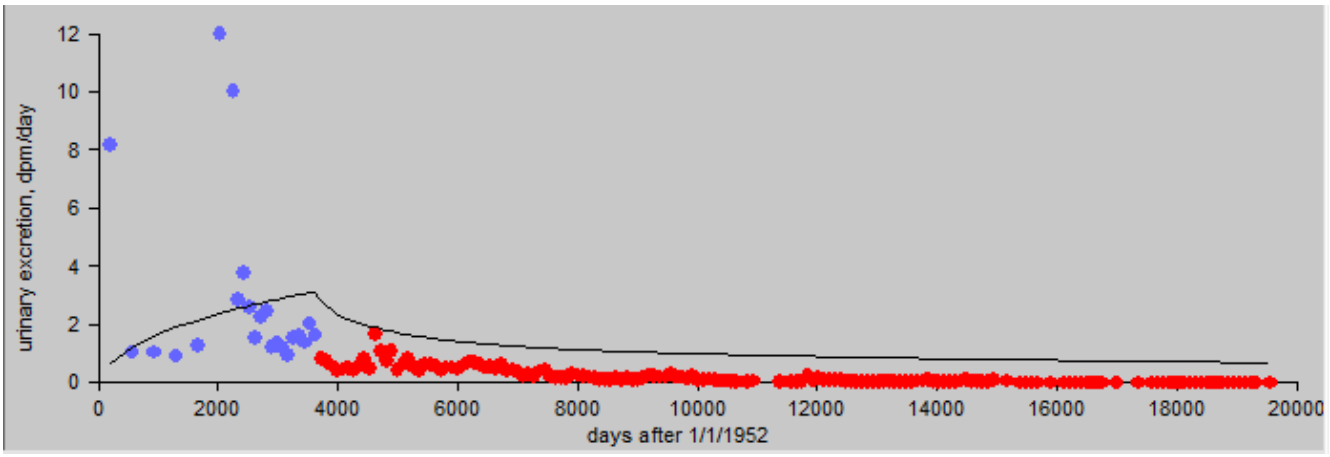

Figure D-43. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/1961, 84th-percentile, type M.

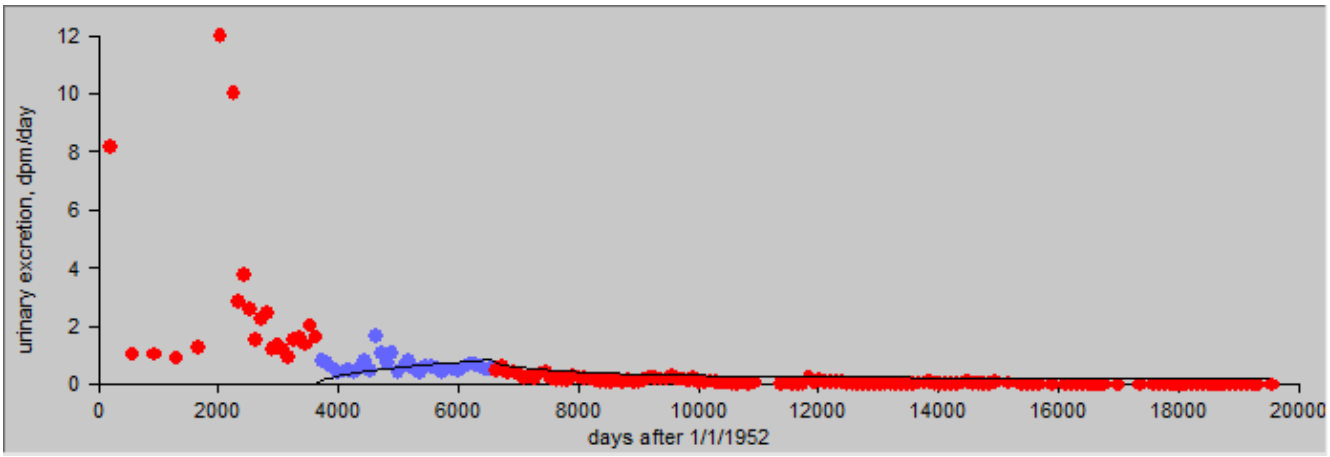

Figure D-44. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1962 to 12/31/1969, 84th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

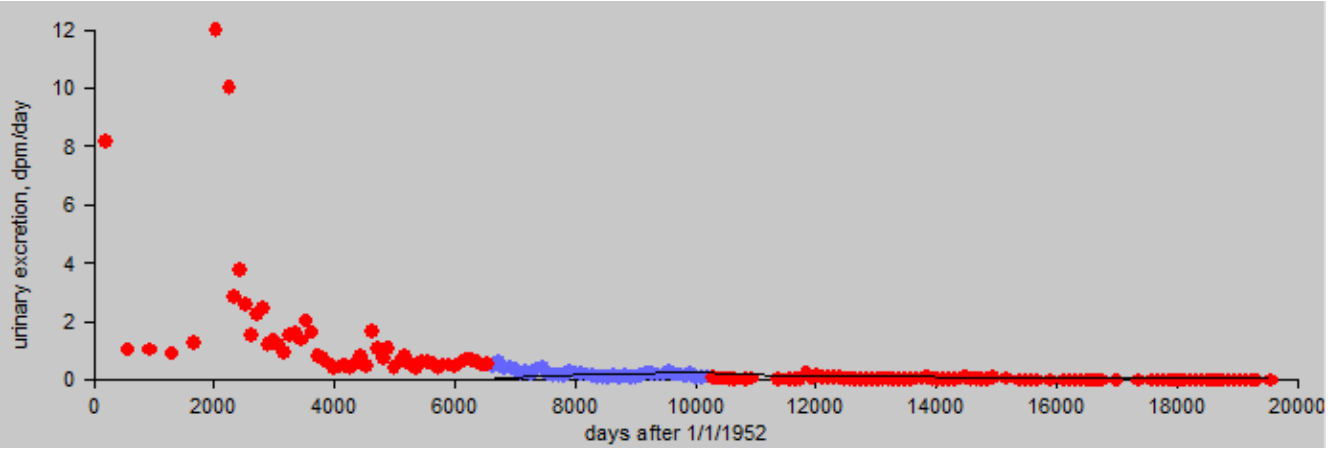

Figure D-45. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1970 to 12/31/1979, 84th-percentile, type M.

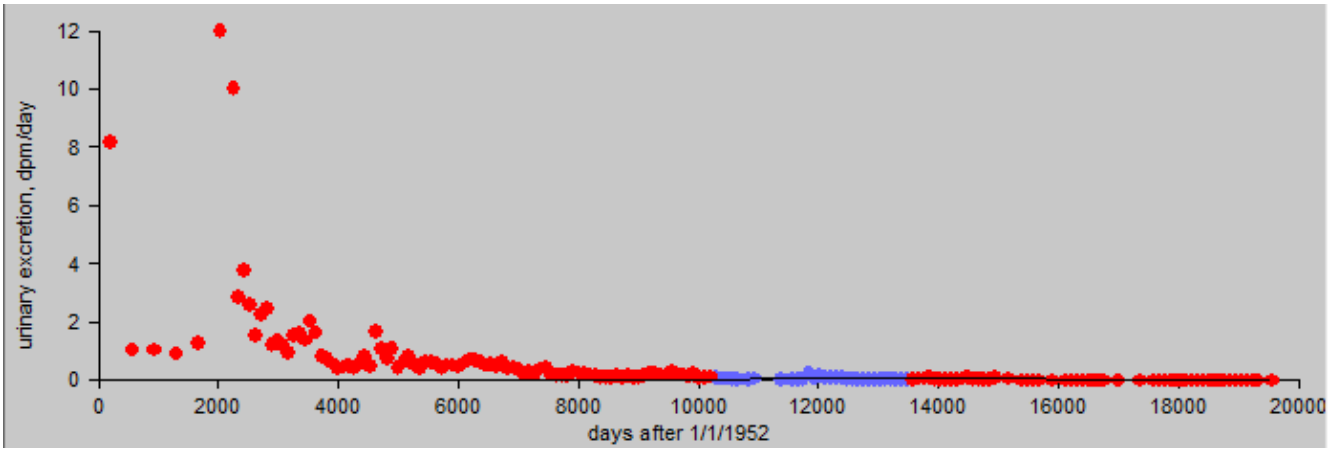

Figure D-46. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1980 to 12/31/1988, 84th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

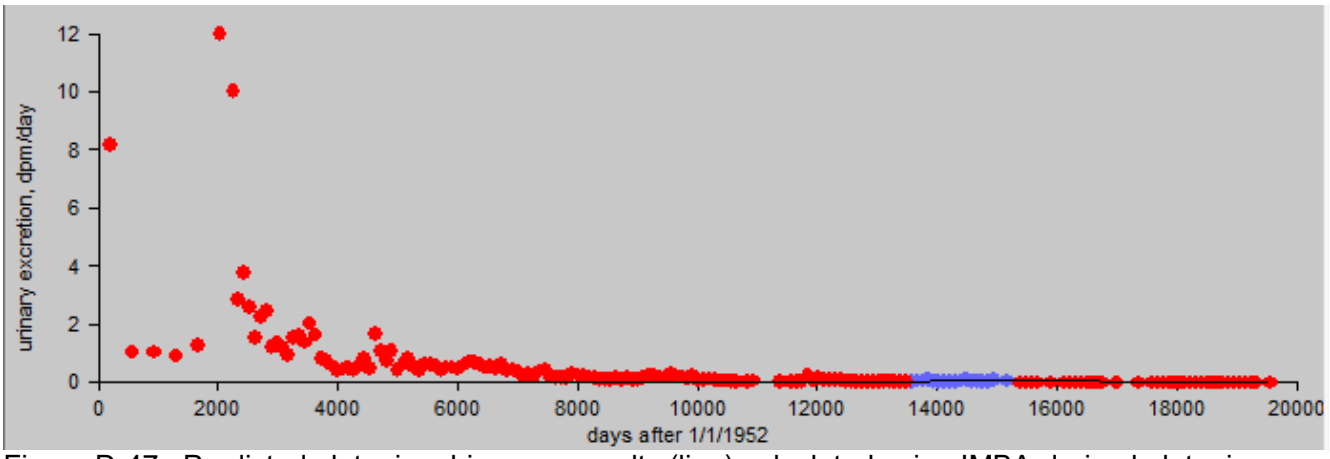

Figure D-47. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 01/01/1989 to 12/31/1993, 84th-percentile, type M.

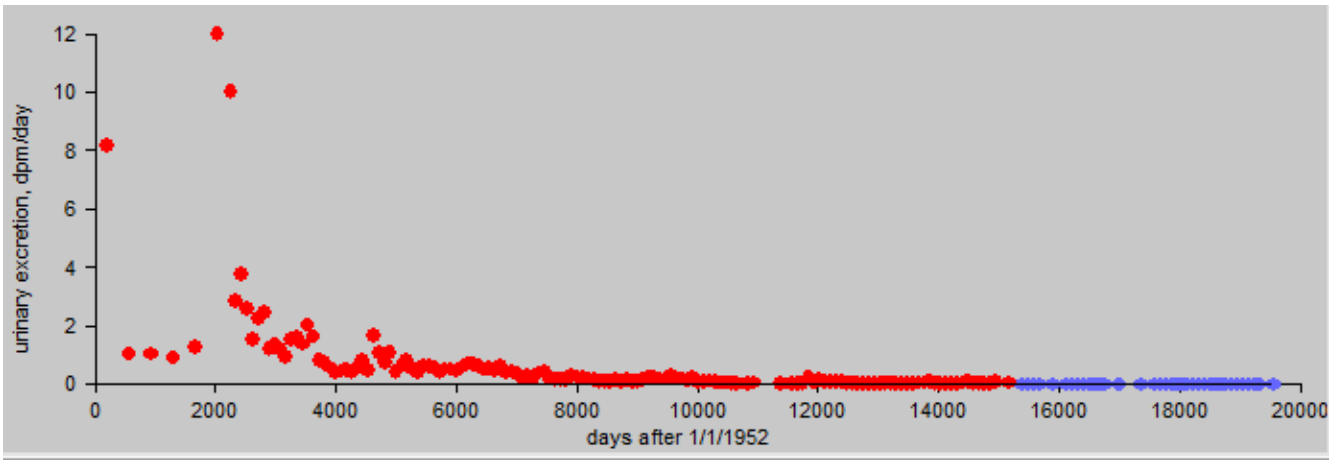

Figure D-48. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 01/01/1994 to 12/31/2005, 84th-percentile, type M.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

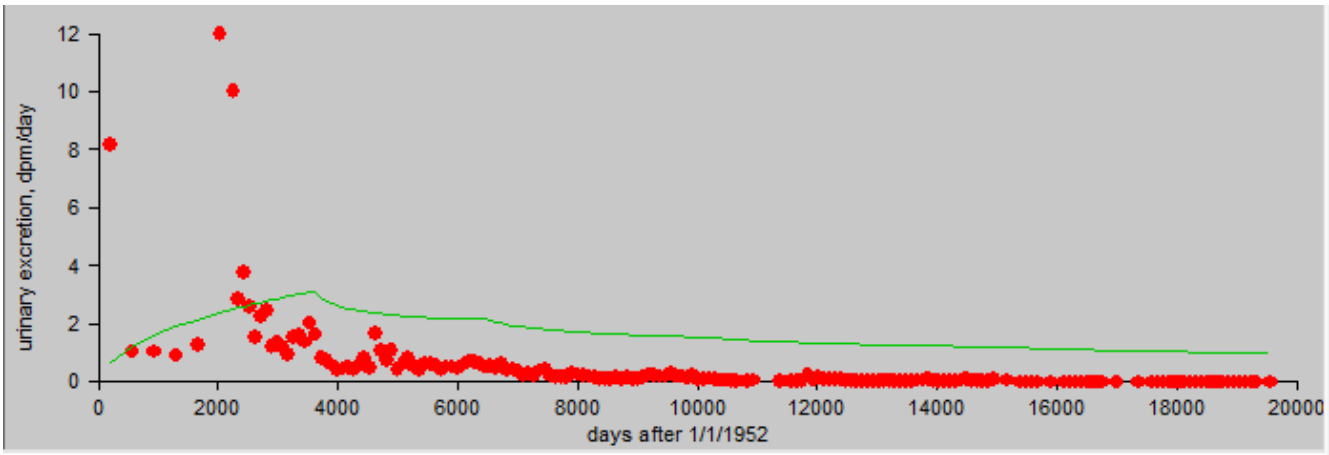

Figure D-49. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), from all intakes 1/1/1952 to 12/31/2005, 84th-percentile, type M.

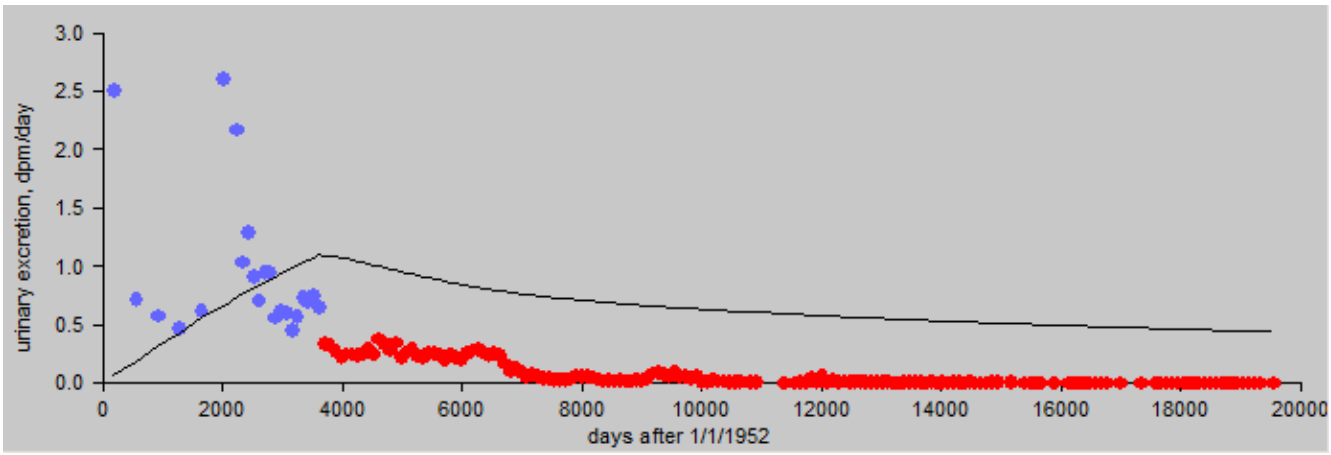

Figure D-50. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/1961, 50th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

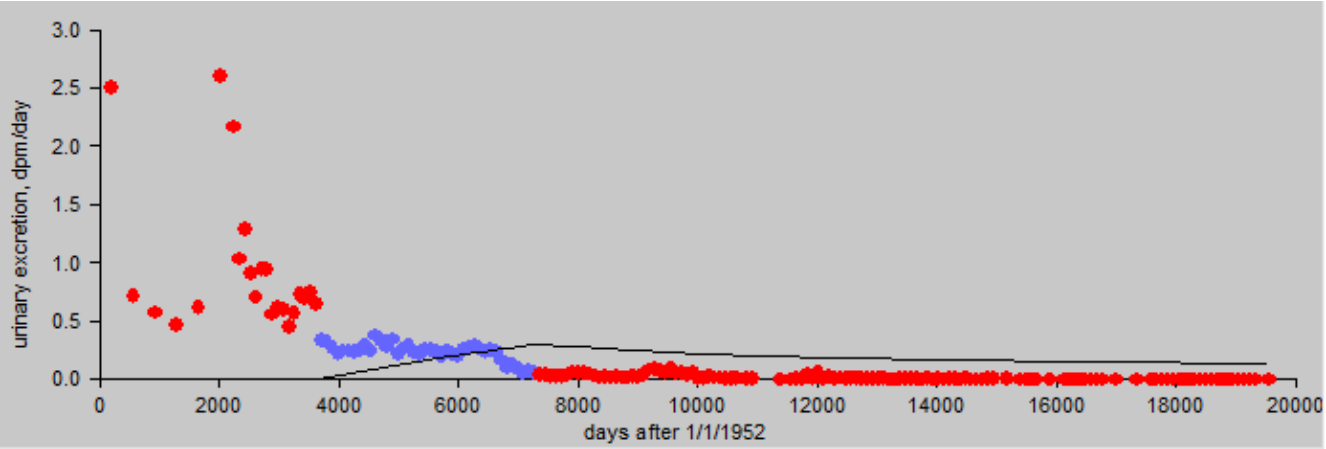

Figure D-51. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1962 to 12/31/1971, 50th-percentile, type S.

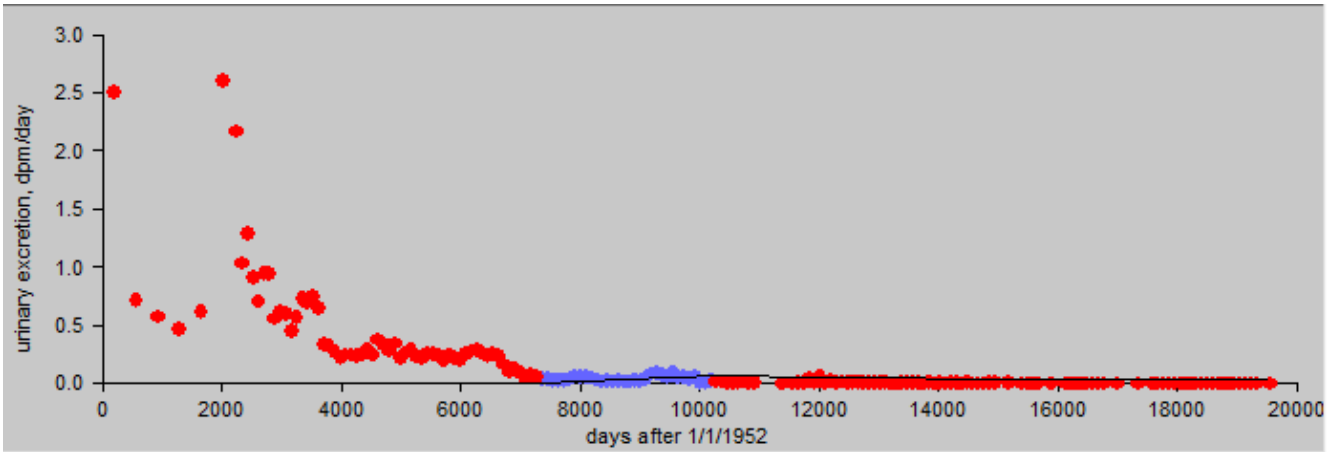

Figure D-52. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1972 to 12/31/1979, 50th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

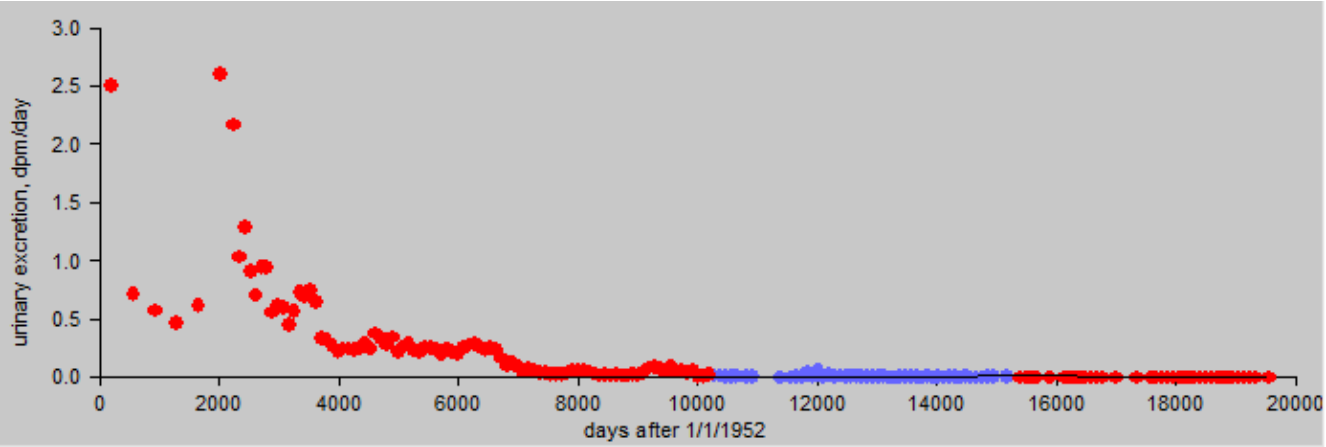

Figure D-53. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1980 to 12/31/1993, 50th-percentile, type S.

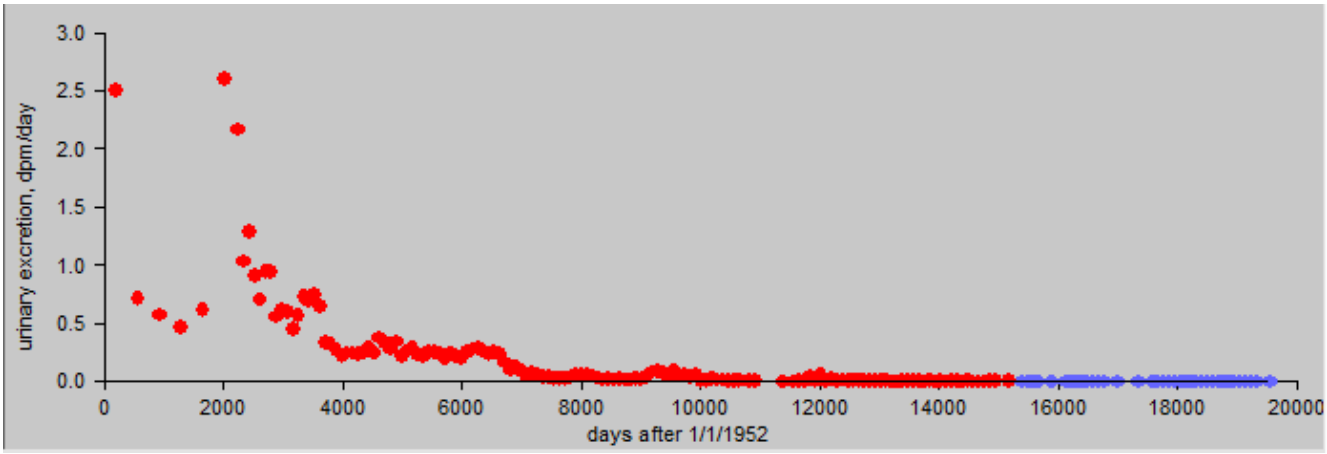

Figure D-54. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1994 to 12/31/2005, 50th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

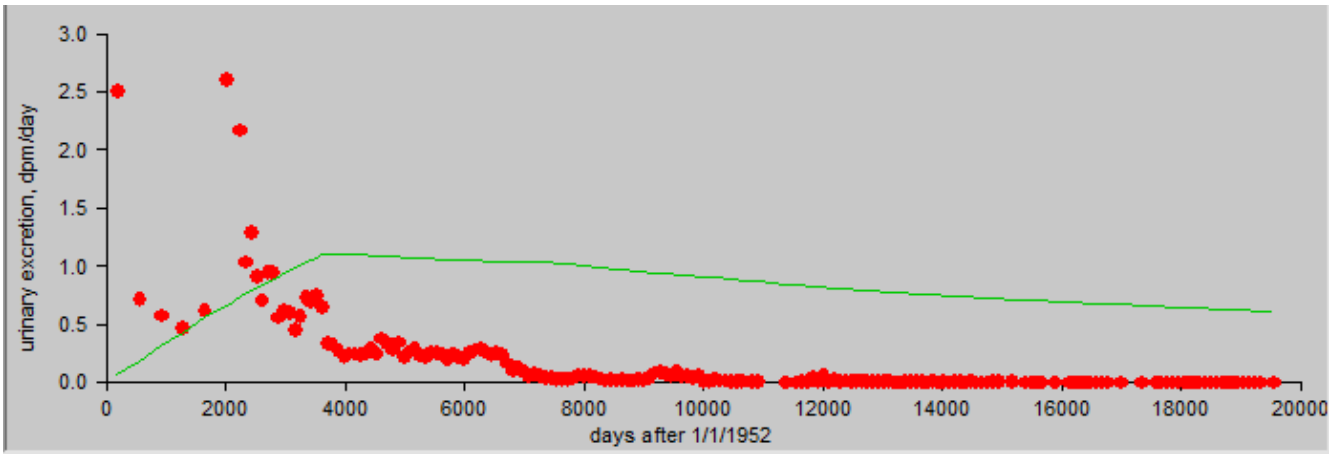

Figure D-55. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/2005, 50th-percentile, type S.

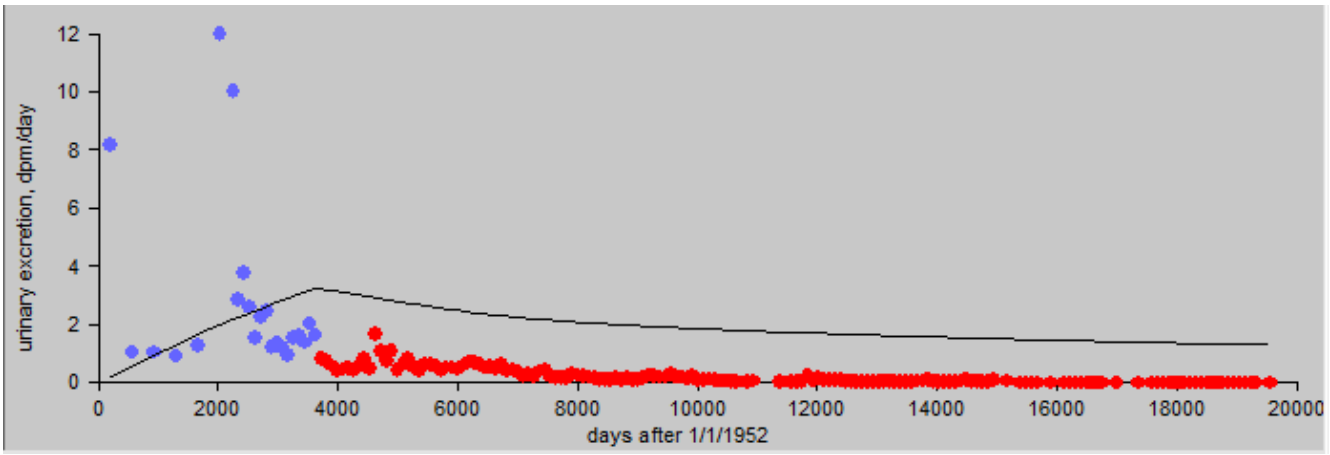

Figure D-56. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/1961, 84th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

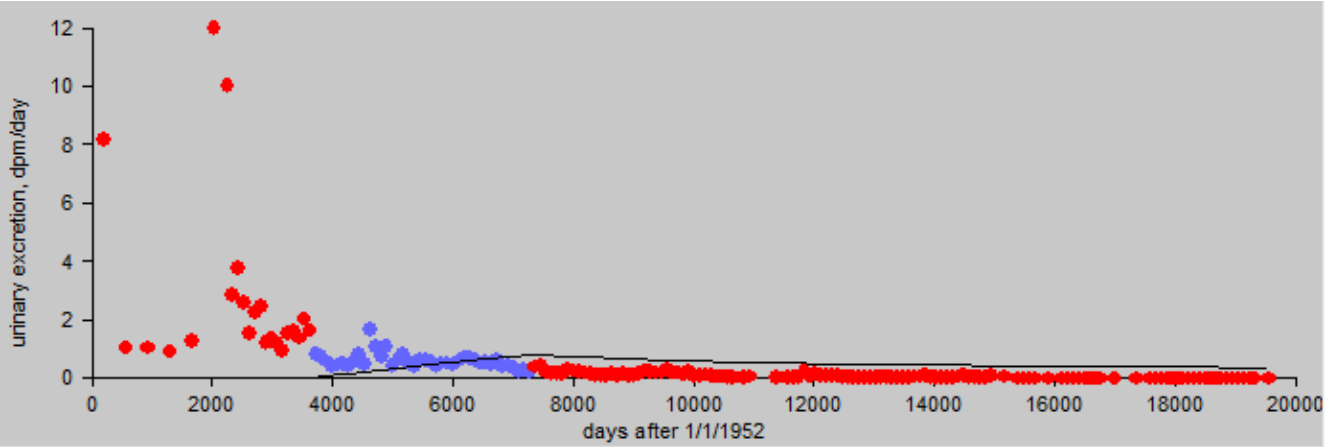

Figure D-57. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1962 to 12/31/1971, 84th-percentile, type S.

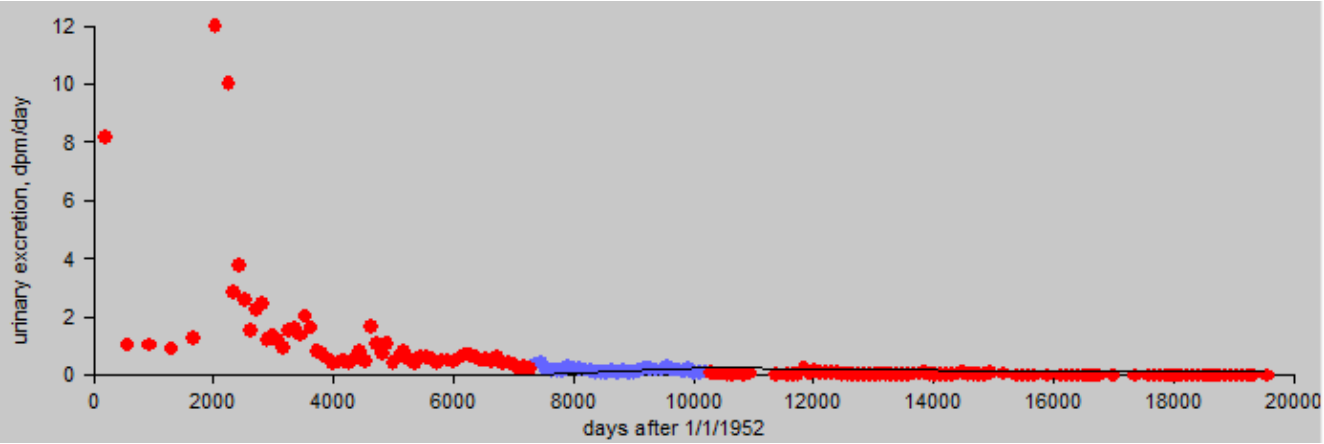

Figure D-58. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1972 to 12/31/1979, 84th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

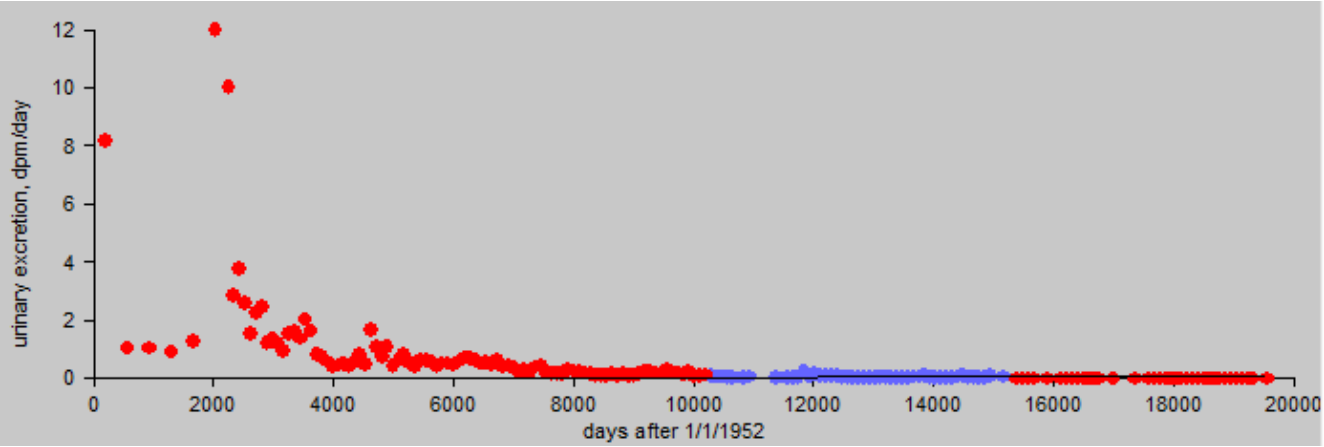

Figure D-59. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1980 to 12/31/1993, 84th-percentile, type S.

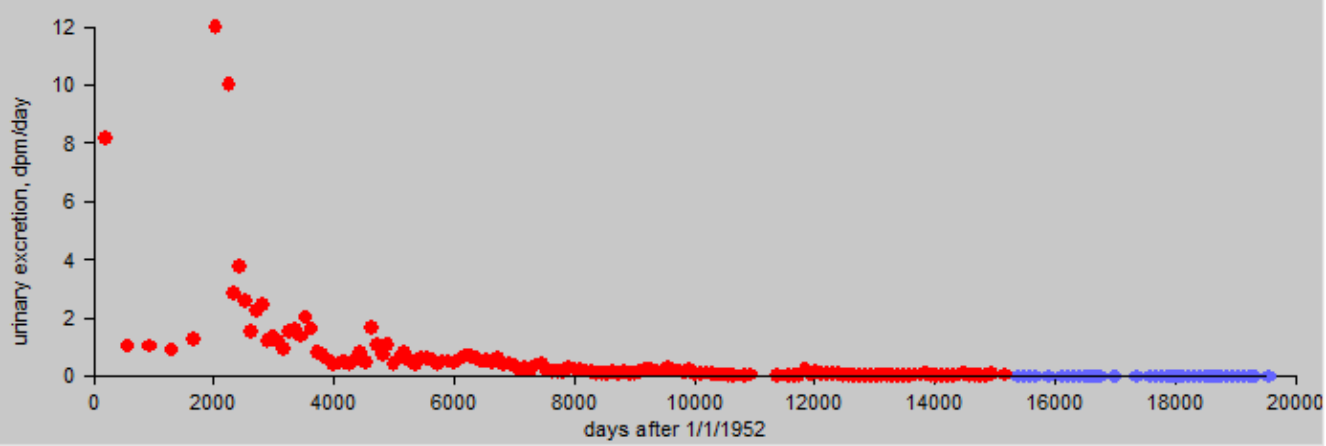

Figure D-60. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1994 to 12/31/2005, 84th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

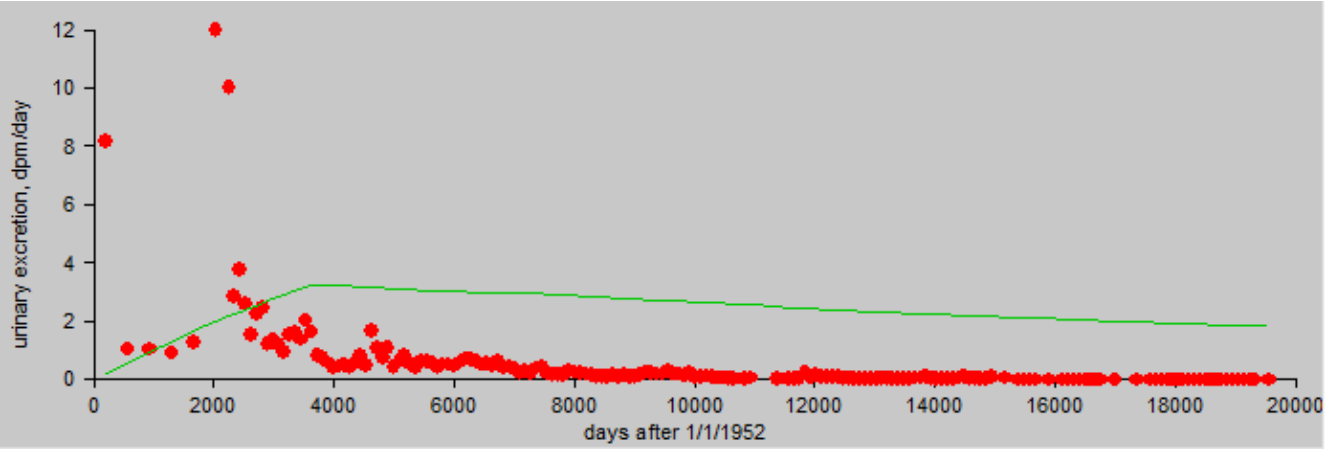

Figure D-61. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/2005, 84th-percentile, type S.

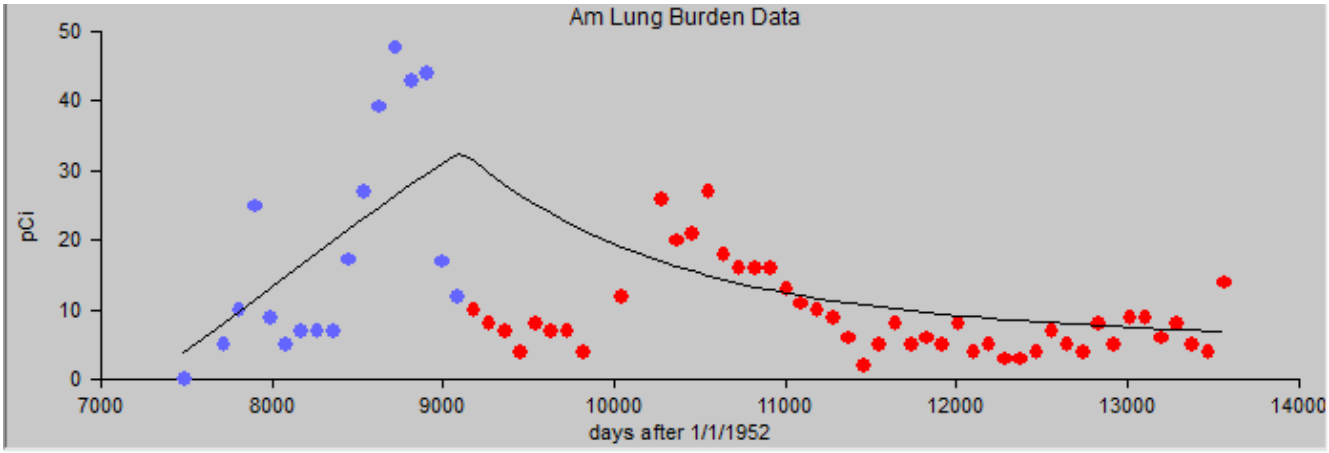

Figure D-62. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1972 to 12/31/1976, 50th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

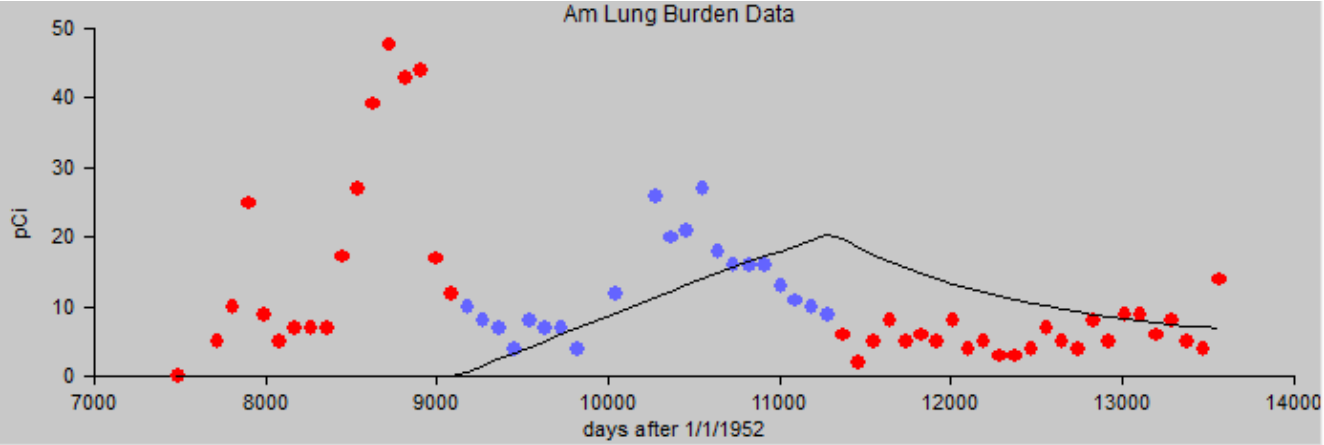

Figure D-63. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1977 to 12/31/1982, 50th-percentile, type S.

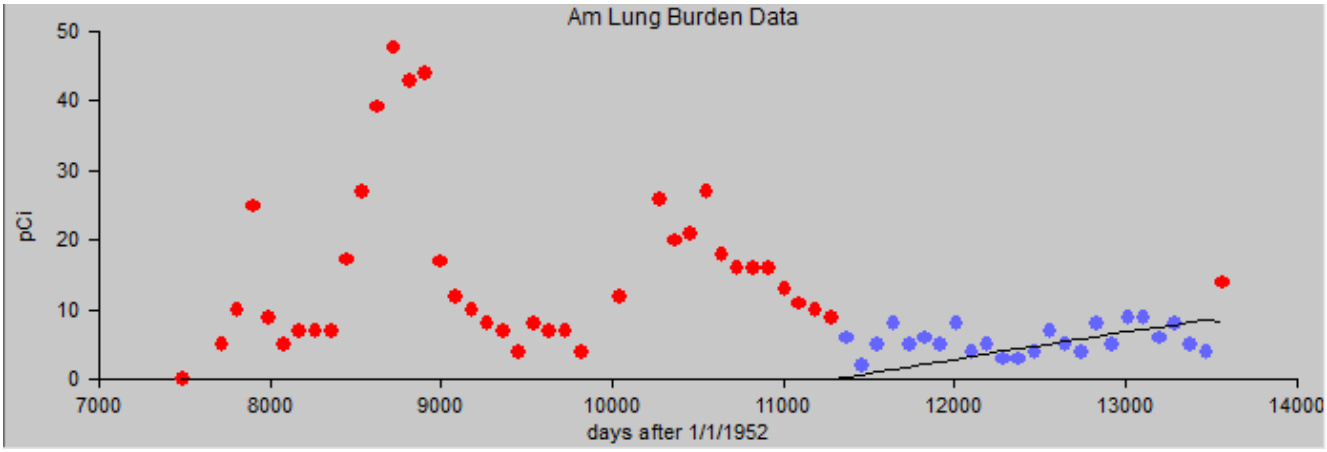

Figure D-64. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1983 to 12/31/1988, 50th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

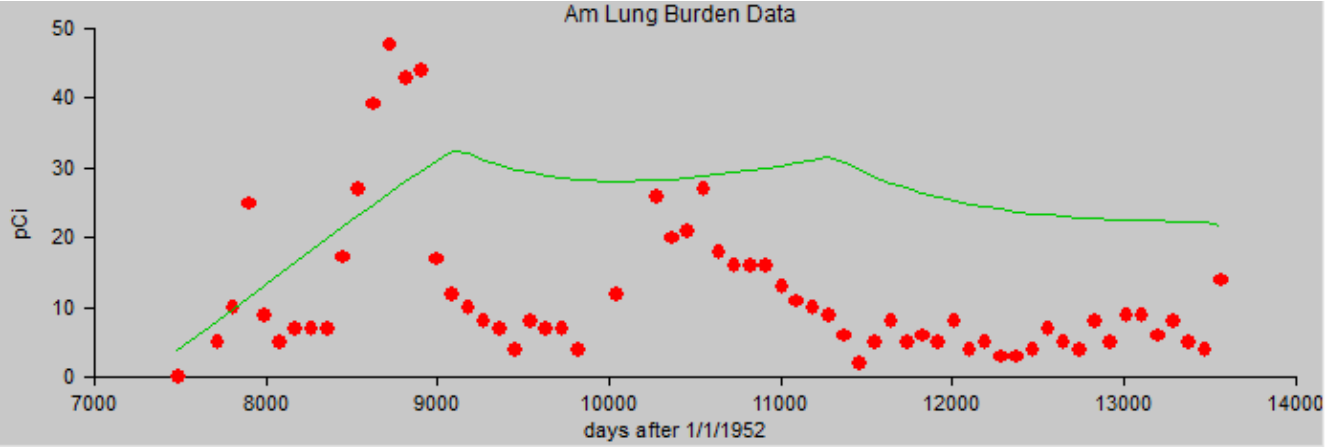

Figure D-65. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots) from all intakes 1/1/1972 to 12/31/1988, 50th-percentile, type S.

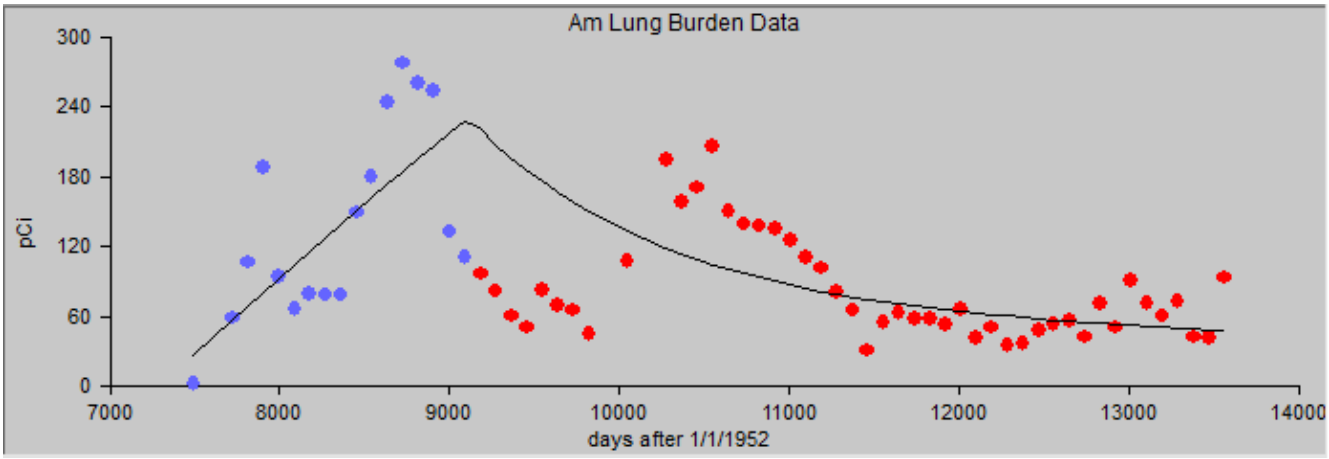

Figure D-66. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1972 to 12/31/1976, 84th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

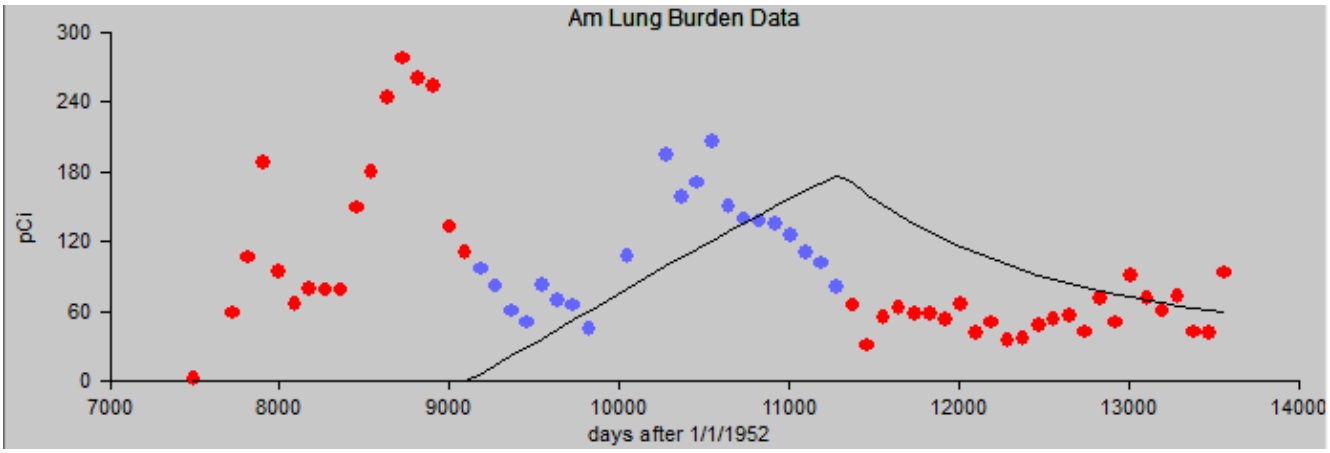

Figure D-67. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1977 to 12/31/1982, 84th-percentile, type S.

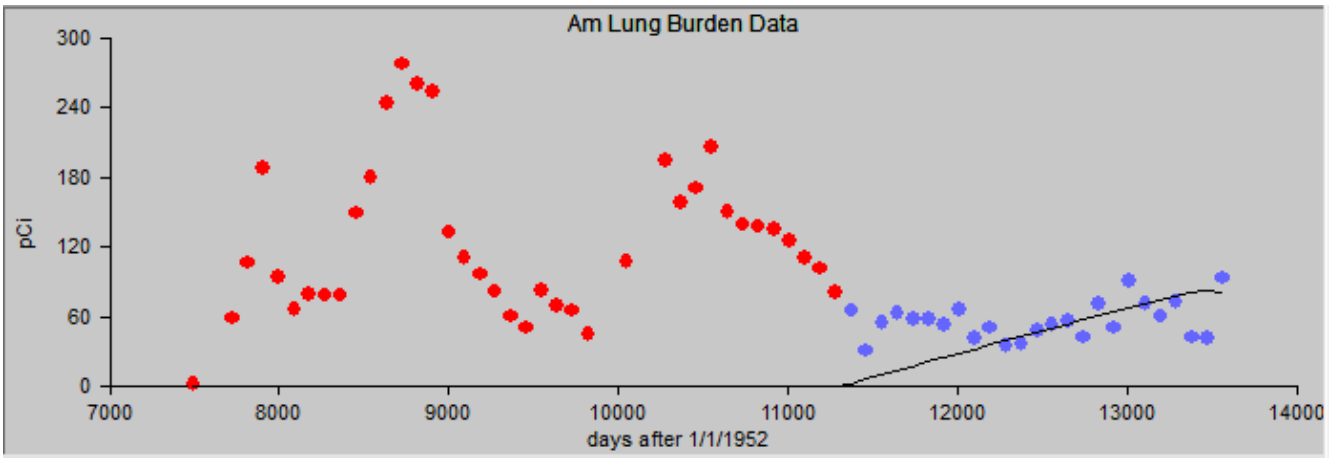

Figure D-68. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1983 to 12/31/1988, 84th-percentile, type S.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

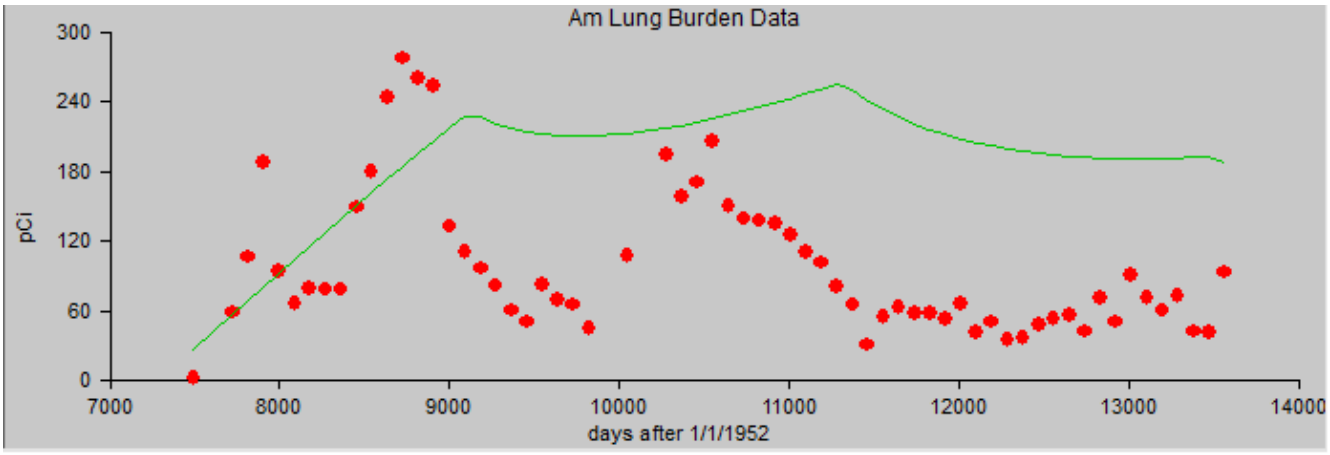

Figure D-69. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots) from all intakes 1/1/1972 to 12/31/1988, 84th-percentile, type S.

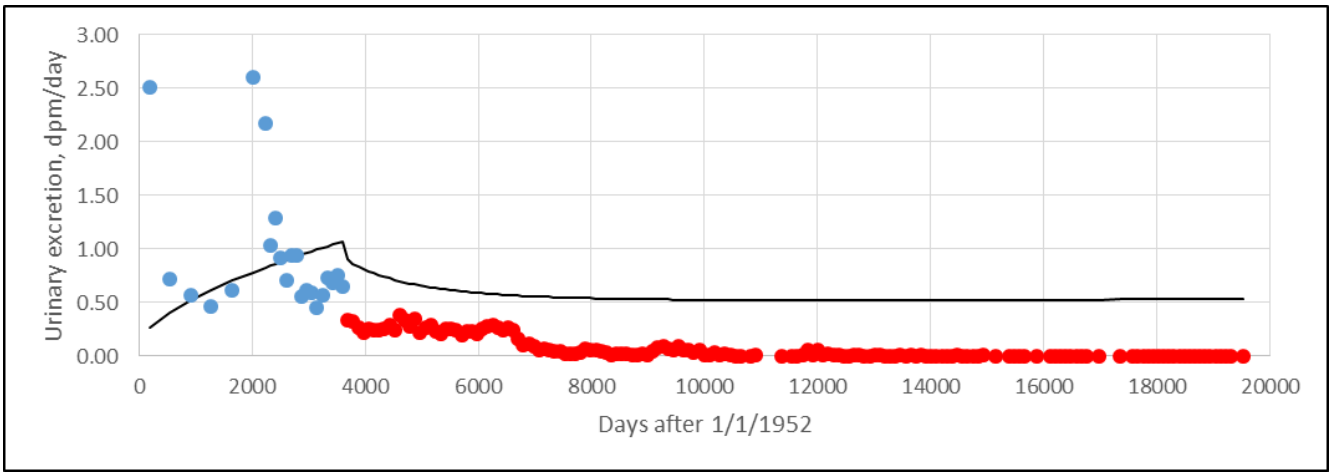

Figure D-70. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/1961, 50th-percentile, type SS.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

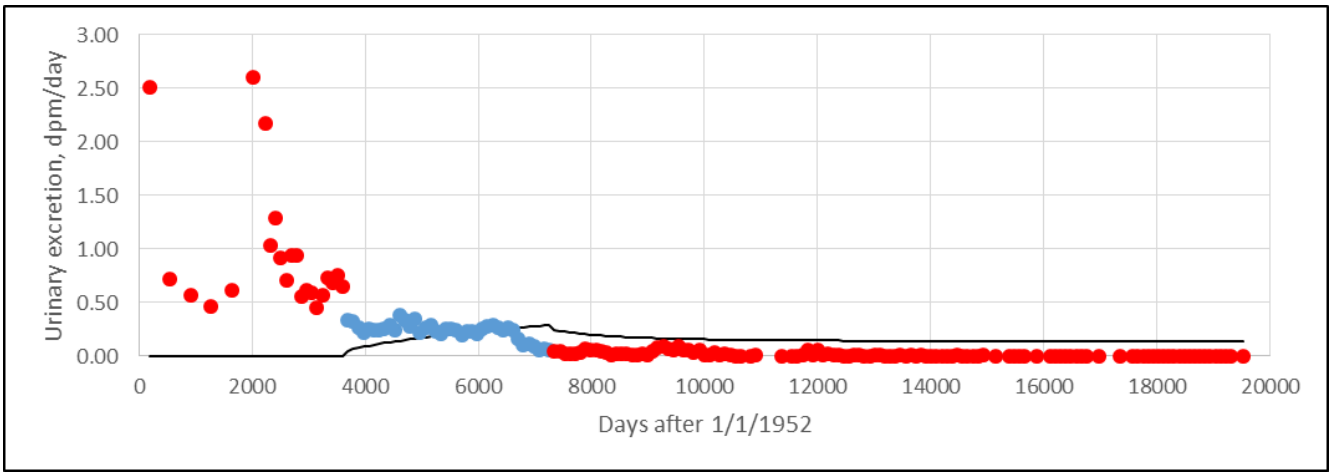

Figure D-71. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1962 to 12/31/1971, 50th-percentile, type SS.

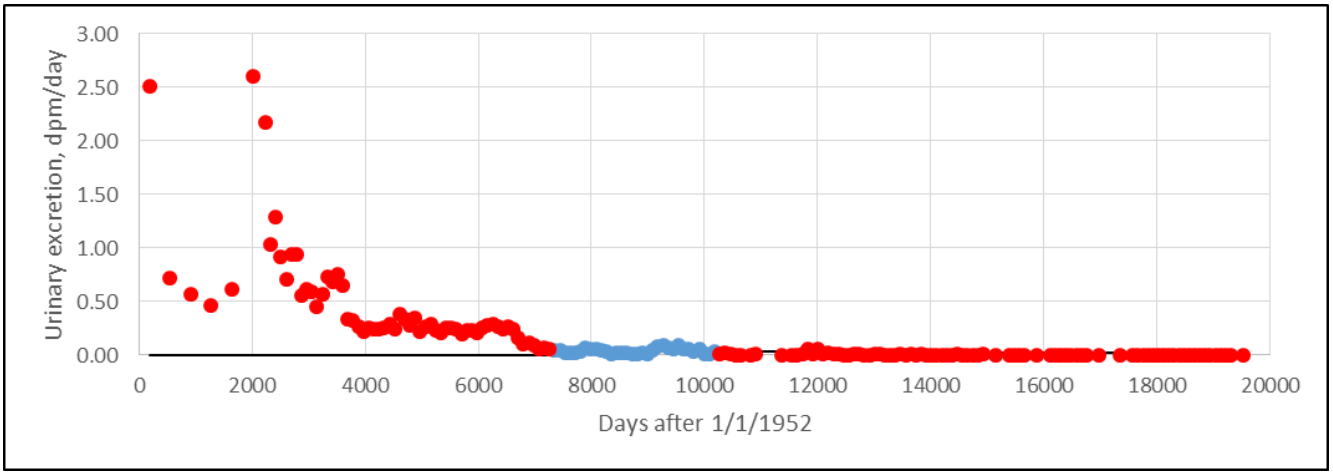

Figure D-72. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1972 to 12/31/1979, 50th-percentile, type SS.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

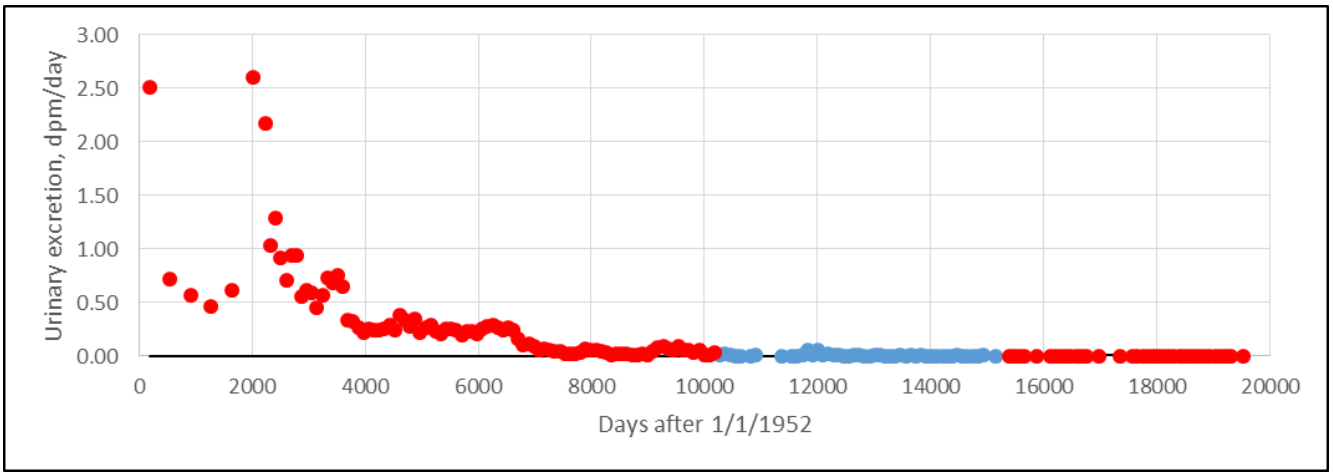

Figure D-73. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1980 to 12/31/1993, 50th-percentile, type SS.

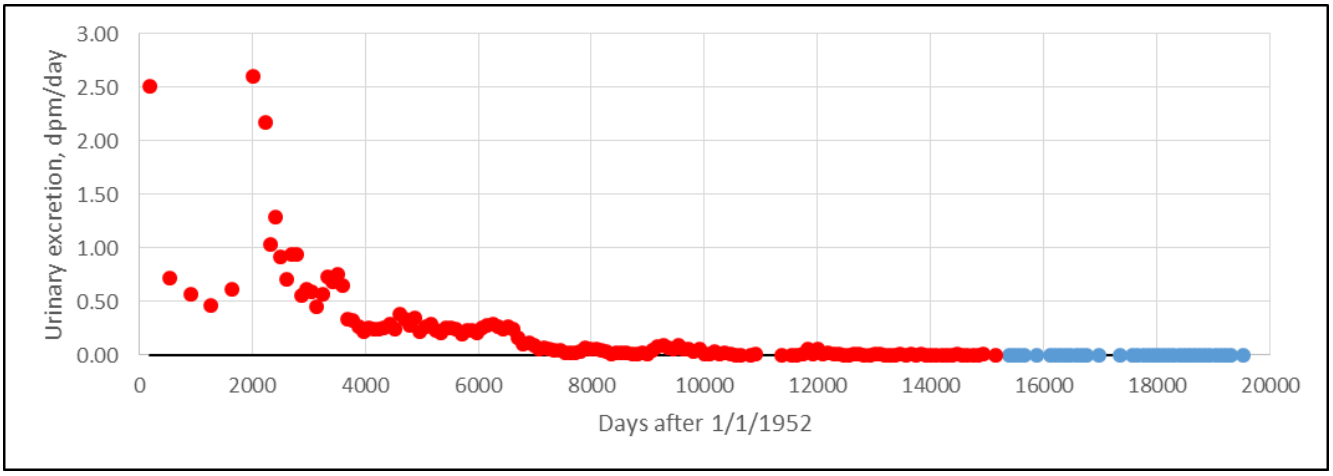

Figure D-74. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1994 to 12/31/2005, 50th-percentile, type SS.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

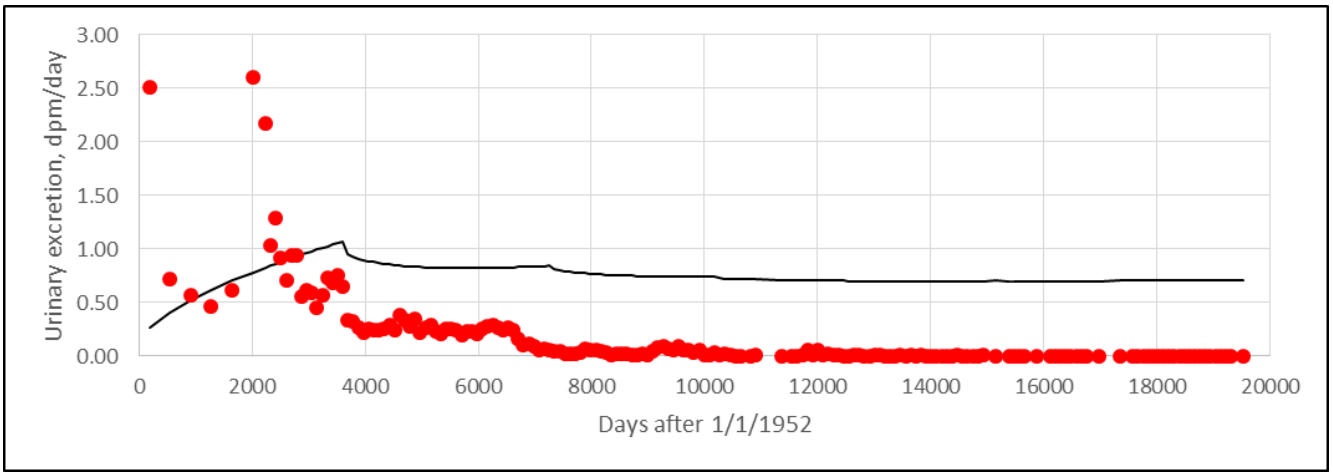

Figure D-75. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/2005, 50th-percentile, type SS.

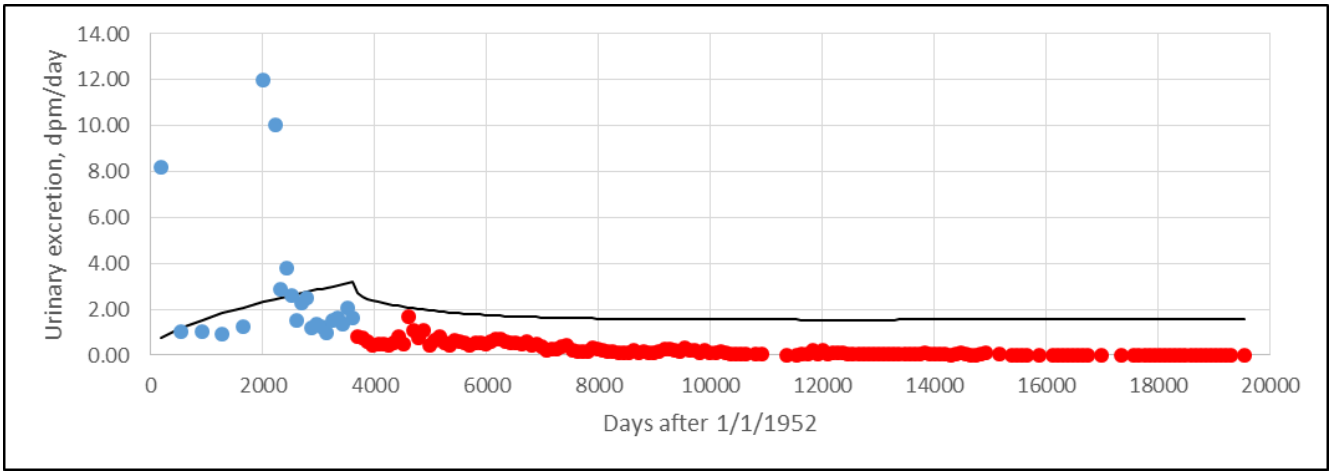

Figure D-76. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/1961, 84th-percentile, type SS.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

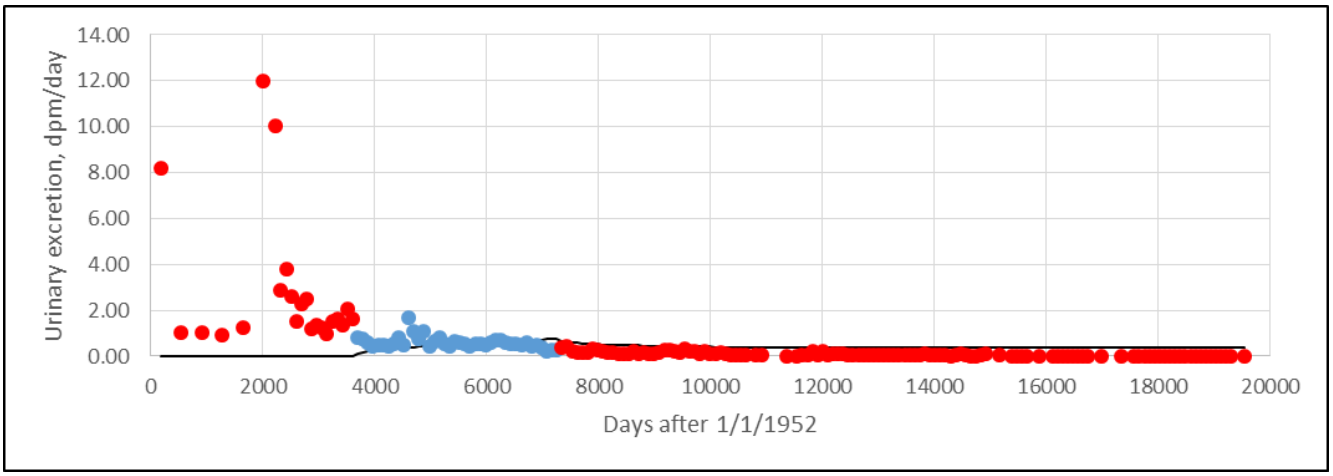

Figure D-77. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1962 to 12/31/1971, 84th-percentile, type SS.

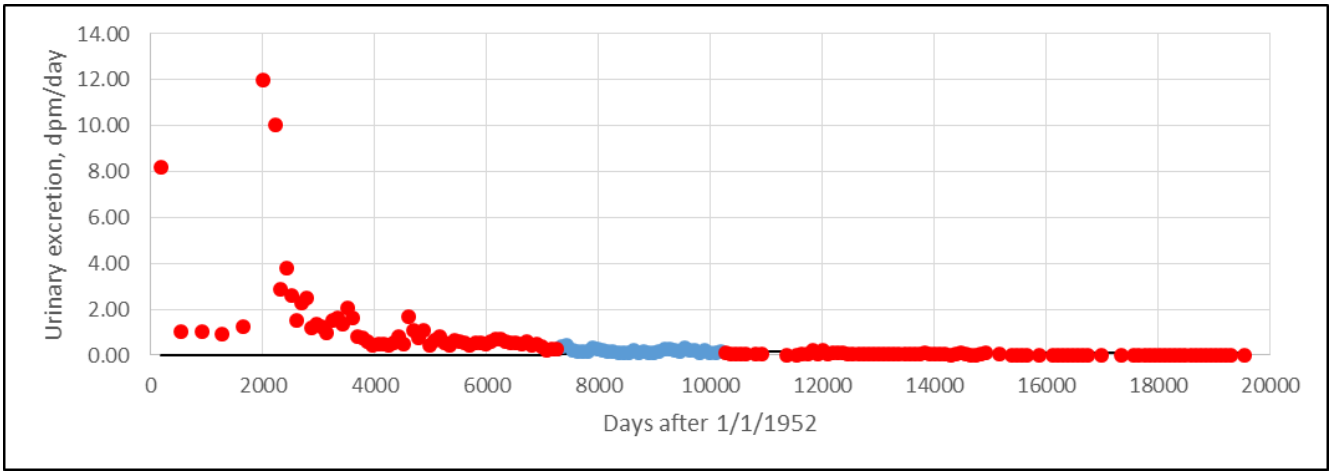

Figure D-78. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1972 to 12/31/1979, 84th-percentile, type SS.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

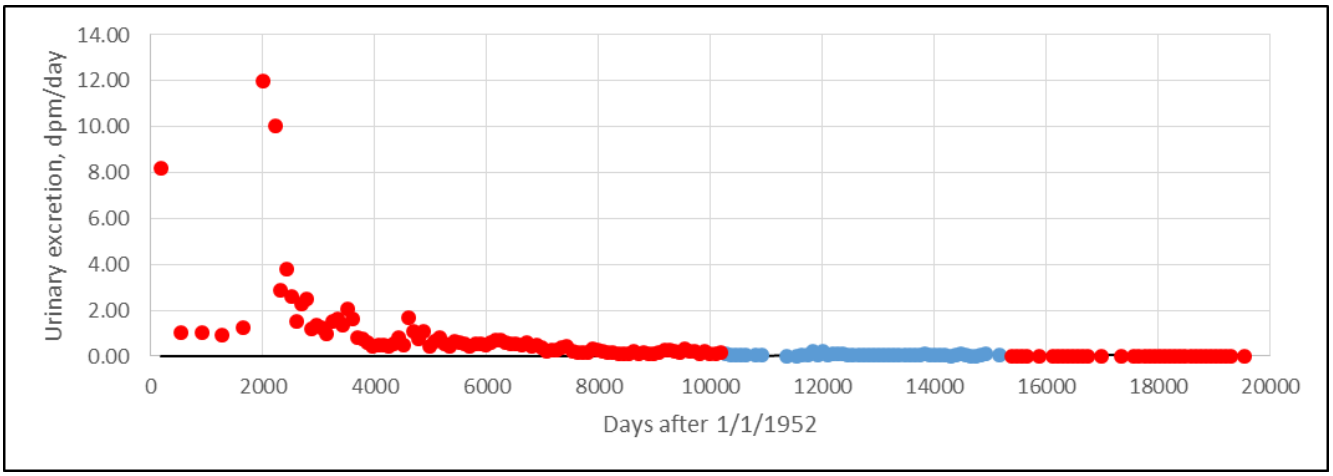

Figure D-79. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1980 to 12/31/1993, 84th-percentile, type SS.

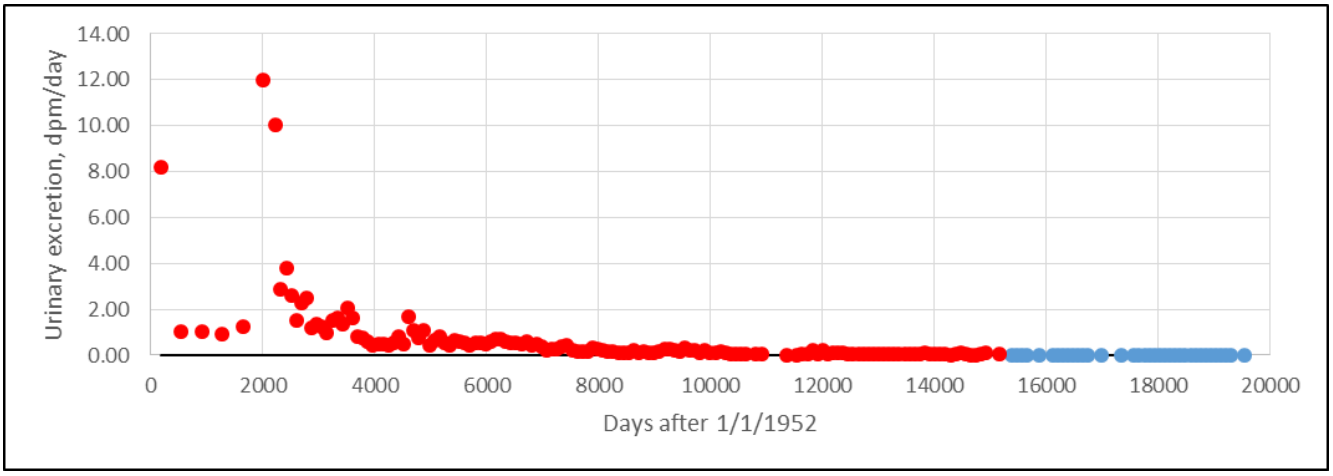

Figure D-80. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1994 to 12/31/2005, 84th-percentile, type SS.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

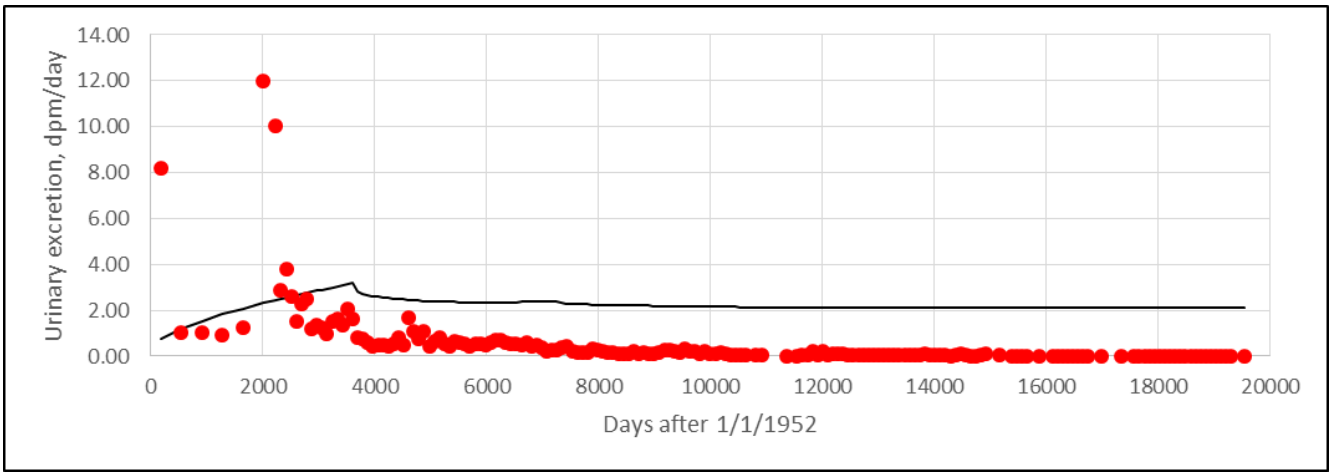

Figure D-81. Predicted plutonium bioassay results (line) calculated using IDOT-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/2005, 84th-percentile, type SS.

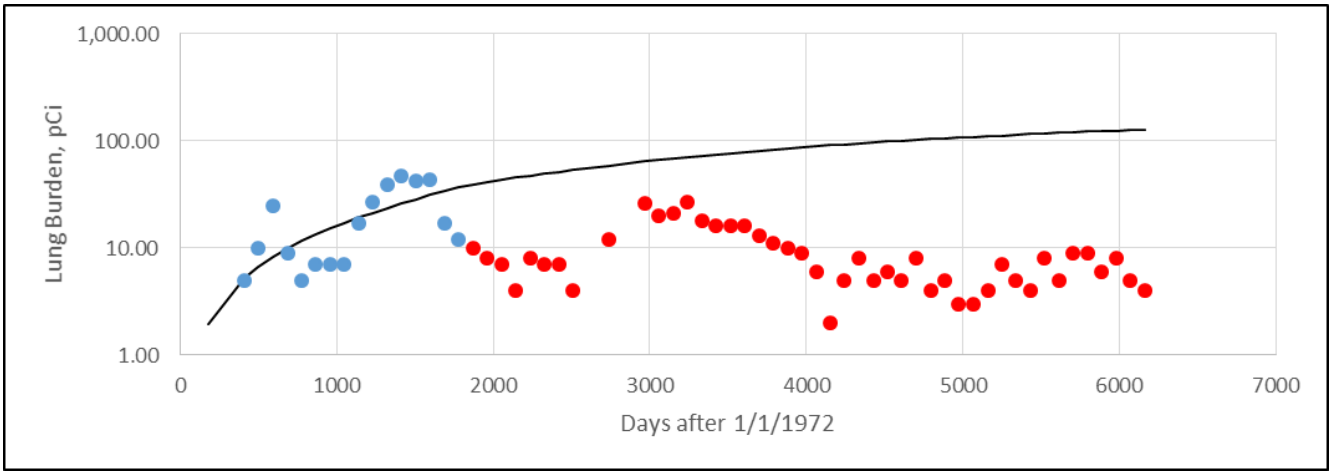

Figure D-82. Predicted americium bioassay results (line) calculated using IDOT-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1972 to 12/31/1976, 50th-percentile, type SS.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

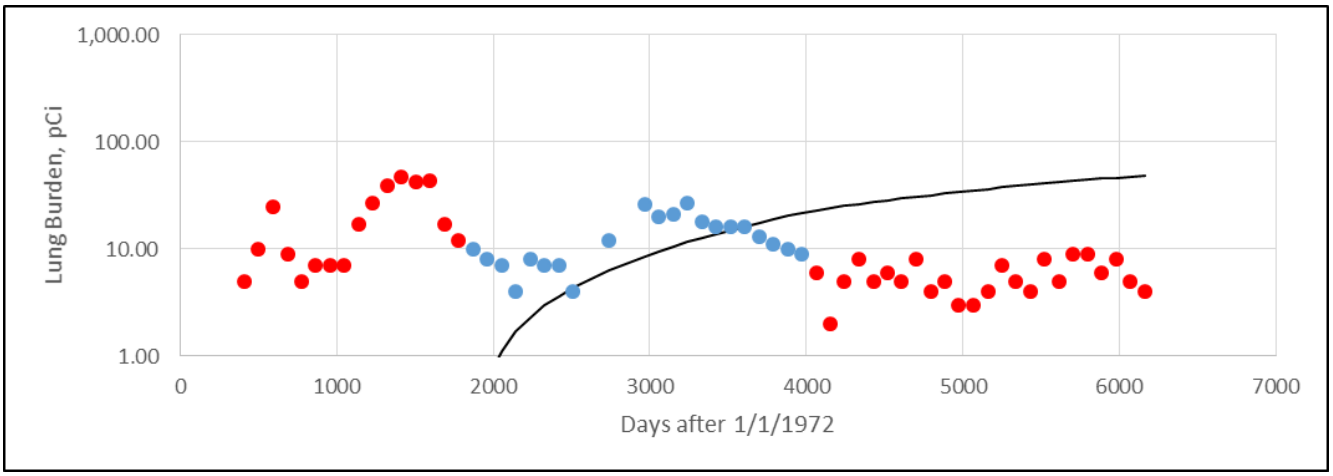

Figure D-83. Predicted americium bioassay results (line) calculated using IDOT-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1977 to 12/31/1982, 50th-percentile, type SS.

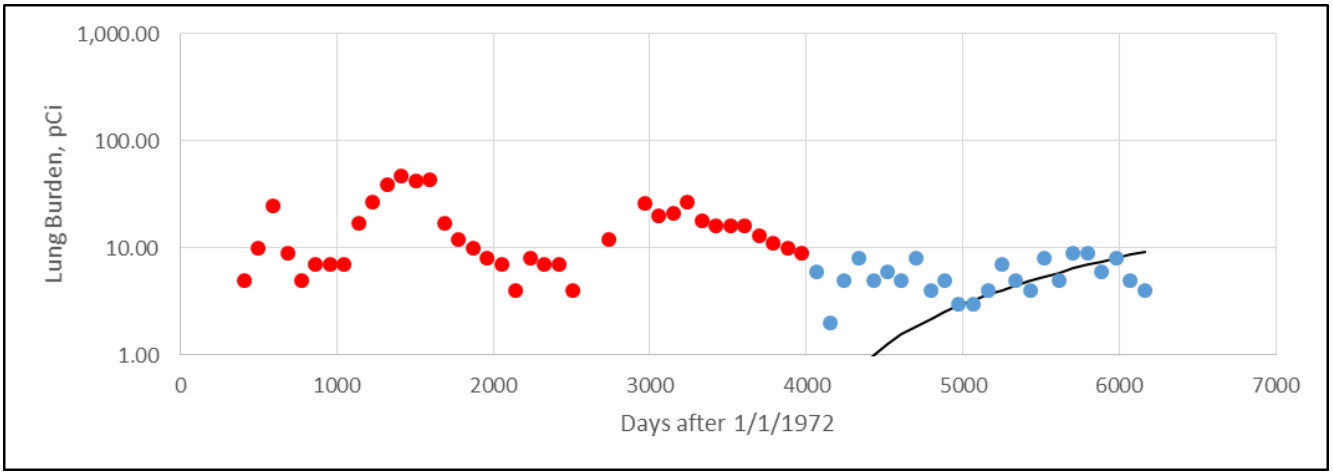

Figure D-84. Predicted americium bioassay results (line) calculated using IDOT-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1983 to 12/31/1988, 50th-percentile, type SS.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

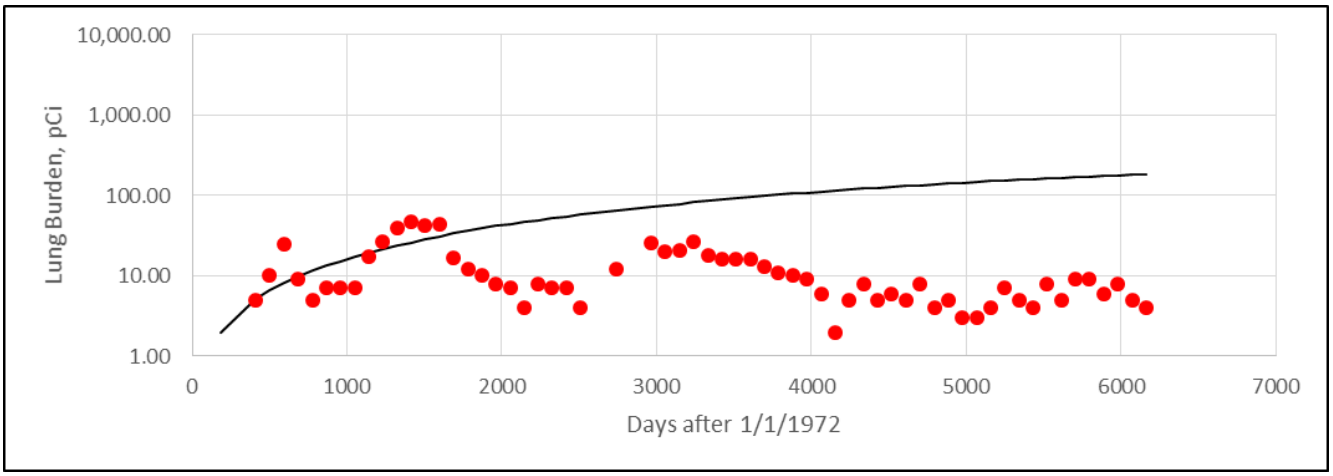

Figure D-85. Predicted americium bioassay results (line) calculated using IDOT-derived americium intake rates compared with measured americium lung burden results (dots) from all intakes 1/1/1972 to 12/31/1988, 50th-percentile, type SS.

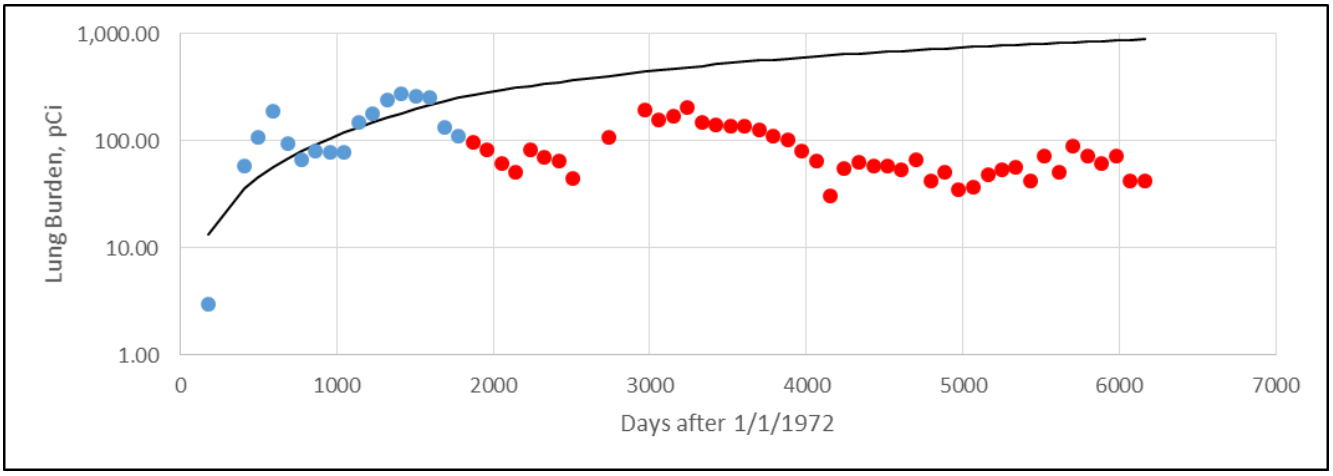

Figure D-86. Predicted americium bioassay results (line) calculated using IDOT-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1972 to 12/31/1976, 84th-percentile, type SS.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

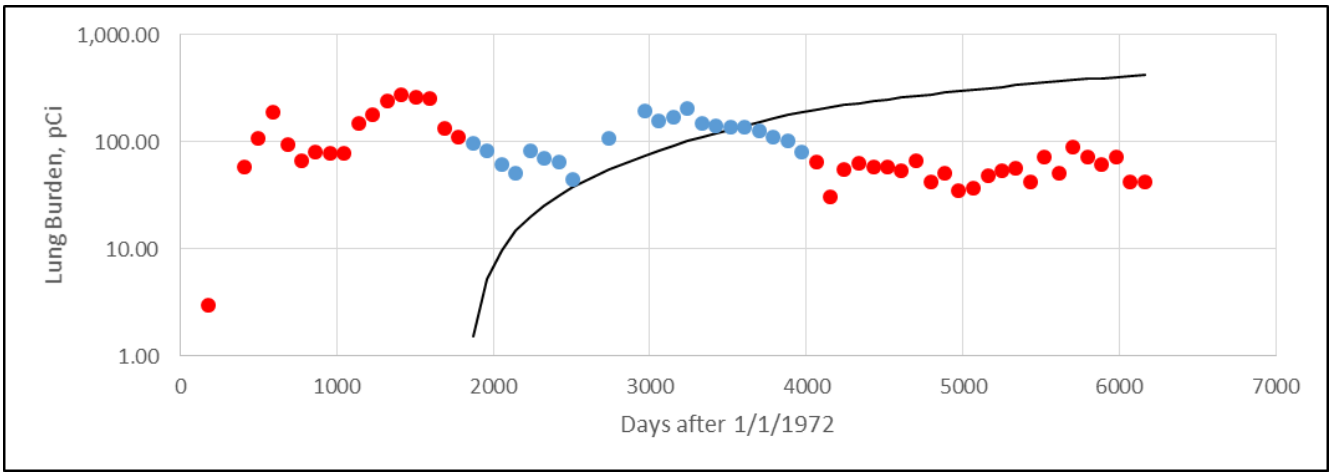

Figure D-87. Predicted americium bioassay results (line) calculated using IDOT-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1977 to 12/31/1982, 84th-percentile, type SS.

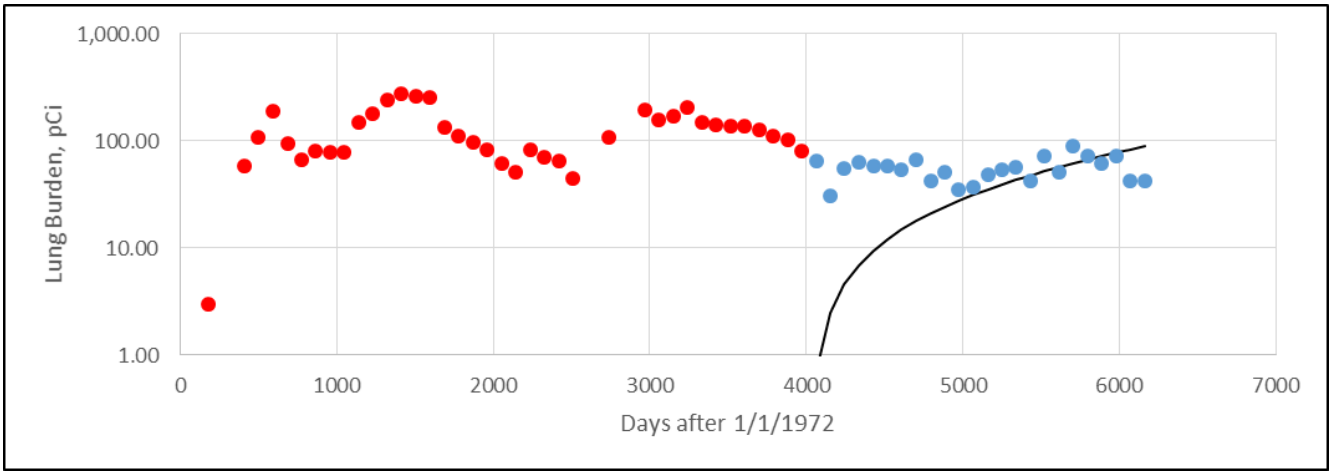

Figure D-88. Predicted americium bioassay results (line) calculated using IDOT-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1983 to 12/31/1988, 84th-percentile, type SS.

ATTACHMENT D
INTERNAL CO-EXPOSURE DOSIMETRY DATA (continued)

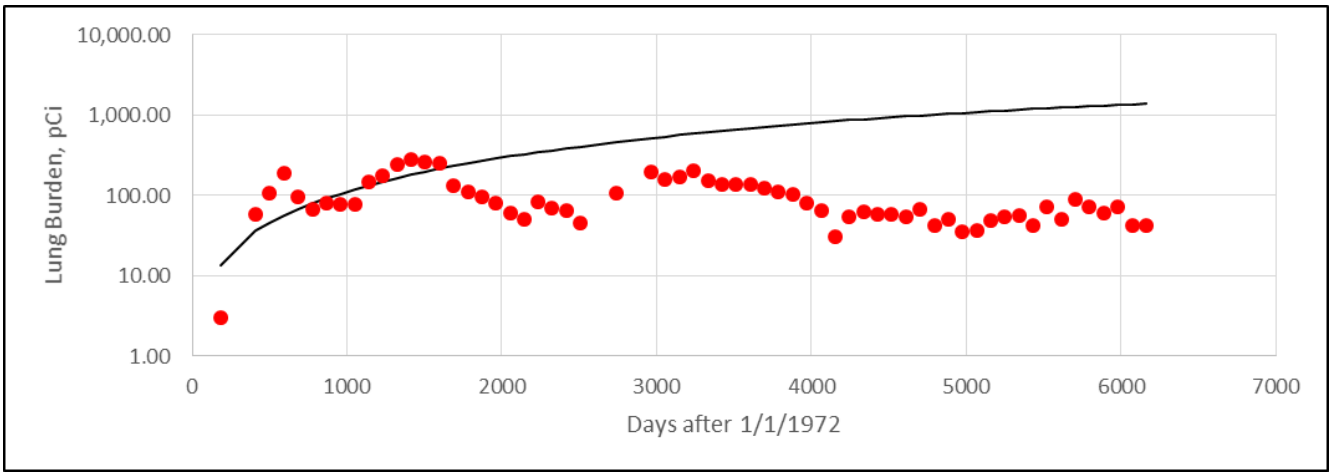

Figure D-89. Predicted americium bioassay results (line) calculated using IDOT-derived americium intake rates compared with measured americium lung burden results (dots) from all intakes 1/1/1972 to 12/31/1988, 84th-percentile, type SS.

ATTACHMENT E
EVALUATION OF THE POTENTIAL FOR INTERNAL DOSE FROM NEPTUNIUM-237

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ATTACHMENT E
EVALUATION OF THE POTENTIAL FOR INTERNAL DOSE FROM NEPTUNIUM-237 (continued)

E.1 INTRODUCTION

The following class of workers was added to the SEC (Sebelius 2013, p. 3):

> *All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Rocky Flats Plant in Golden, Colorado, from April 1, 1952, through December 31, 1983, 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 included in the Special Exposure Cohort.*

This recommendation derived, in part, from the conclusion that neptunium processing at RFP took place at least until 1983, and that available monitoring data are insufficient for estimating potential internal exposures to neptunium.

E.2 OPERATIONS

Neptunium Processing at the Rocky Flats Plant (Connor and Basso 1981, p. 11), states that processing included preparation of pure neptunium oxide, metal, and metal alloys as well as ^{237}Np recovery from a variety of residues. Processes included dissolution, anion exchange, precipitation, filtration, calcination, conversion to fluoride, and reduction to metal. Fabrication steps such as casting and rolling were also sometimes performed for the production of high-purity metal shapes and foils. Neptunium was recovered from residual materials including sand, slag, crucibles, casting skulls, and various alloys containing plutonium, tin, uranium, or zirconium. This description of RFP neptunium operations is echoed in *Actinide Processing at Rocky Flats* (Connor and Basso 1984, p. 327).

E.2.1 1962 to 1983

Neptunium was processed at RFP as early as 1962 (Connor and Basso 1981, p. 11). There is no evidence of continuous routine neptunium operations. Rather, it points to a series of discrete tasks performed from 1962 through 1983 involving a few to a few hundred grams, usually processed by request from other DOE laboratories.

The first special-order request for neptunium processing at RFP came from Lawrence Radiation Laboratory, now known as LLNL, which required high-purity neptunium without gamma-emitting impurities. Techniques for purifying neptunium (including the removal of other actinides) were developed as a result of this project. The project also called for preparation of NpPu alloys, which were prepared by coreducing neptunium and PuF_4 mixtures with calcium metal. This effort led to an ability to produce pure neptunium metal that could be cast with plutonium or uranium to form alloys. The first NpPu alloy was produced in 1964, and several kilograms of neptunium metal were produced in subsequent years for preparation of uranium and plutonium alloys.

Other specific projects involving ^{237}Np processing included the preparation of high-purity neptunium oxide for the Oak Ridge National Laboratory (ORNL) isotope pool, neptunium metal foils for the Savannah River Site (SRS), and neptunium metal disks for use in the neutron dosimeter program for the liquid-metal, fast-breeder reactor (Connor and Basso 1981, p. 12).

Based on reviews of numerous reports in the SRDB as well as reviews of classified documents, it is evident that only relatively small quantities of ^{237}Np were processed at RFP at any given time. The

largest single operation appears to have been a special order in 1966 that involved just over 300 g of neptunium (Byrne 1967).

E.2.2 After 1983

A single operation was described in *Production-Scale Plutonium-Neptunium Separation and Residue Recovery at Rocky Flats Plant* (Martella, Guyer and Leak 1987). This document does not make clear exactly when this operation took place, although it does indicate that the campaign lasted approximately 1 year. An interview of one of the authors, who was also the Principal Engineer who designed the process and directed the activities, estimated that the operation began around January 1985 (ORAUT 2014a).

This 1985 operation involved the processing of plutonium scrap containing as little as 0.5% (by weight) neptunium to separate and recover the two metals. The feed material for this process consisted of 63,728 g of plutonium containing 232 g of neptunium. The separation process involved oxidizing the plutonium residue, passing Pu(III) through an anion exchange resin bed, and leaving Np(IV) behind for subsequent elution, evaporation, denitrification, and calcination to oxide (Martella, Guyer, and Leak 1987, p. 10). The authors reported completion of 24 separations over the course of a year using this process, resulting in the purification of 58,282 g of plutonium by the removal of 222 g of neptunium (~96% recovery). NIOSH reinterviewed the coauthor (also Principal Engineer) who stated that project personnel consisted of five experimental operators who performed the glovebox work and the Principal Engineer (ORAUT 2014b).

The final purified plutonium that resulted from this operation contained only 0.0069% neptunium. The "neptunium product" consisted of 1,429 g of plutonium and 222 g of neptunium with a Pu:Np mass ratio of 6.4 (Martella, Guyer, and Leak 1987, p. 16). This operation involved no purified neptunium; the dose from any internal exposure would have been dominated by the overwhelming amount of plutonium in the mixture, making neptunium bioassay unnecessary. Given the much greater specific activity of ^{239}Pu, plutonium, bioassay would account for all organ doses, of which plutonium would be the dominant component. An evaluation of the relative dose contribution from plutonium and neptunium from an inhalation intake of the PuNp mixture is presented later in Section E.10.

According to the Principal Engineer, separation of plutonium and neptunium from alloys ended in 1987 (ORAUT 2014a, p. 10). This statement is consistent with another reference that states that the neptunium program at RFP began in 1964 and was terminated in 1988 (Vejvoda 2005).

E.3 INVENTORY TOTALS AND MEASURED QUANTITIES

Documents and interviews with former employees indicate that neptunium was present at RFP from 1962 to 2003. From 1963 through fiscal year (FY) 2003, neptunium quantities between 29 g and 1,318 g are recorded in Material Balance Area accounts (Vejvoda 2005; Meyers 2013; DOE 2014a). Neptunium-237 inventory amounts at RFP from 1963 to 2003 obtained from the National Nuclear Security Administration Nuclear Materials Management and Safeguards System (NMMSS) are shown in Table E-1. Although neptunium work began in 1962, no inventory information has been found for the end of 1962 (the first year of production). The neptunium program was reportedly terminated in 1988 (Vejvoda 2005), but neptunium remained in the RFP nuclear material inventory until 2003. Nuclear material holdup surveys were conducted sometime after 1990 in response to concerns by the Defense Nuclear Facilities Safety Board about unaccounted for nuclear material at Rocky Flats. These surveys identified gram quantities of residual neptunium in previously cleaned gloveboxes in Building 771, Rooms 153 and 182 (ORAUT 2014c, p. 11).

ATTACHMENT E
EVALUATION OF THE POTENTIAL FOR INTERNAL DOSE FROM NEPTUNIUM-237 (continued)

Table E-1. Neptunium-237 fiscal-year-end inventories.[a]

FY	Np-237 (g)	FY	Np-237 (g)	FY	Np-237 (g)
1963	29	1976A[b]	468	1990	937
1964	601	1977	458	1991	941
1965	1,292	1978	567	1992	941
1966	740	1979	492	1993	941
1967	1,215	1980	744	1994	941
1968	972	1981	486	1995	941
1969	1,190	1982	699	1996	935
1970	1,105	1983	869	1997	926
1971	1,318	1984	1,040	1998	930
1972	788	1985	931	1999	880
1973	768	1986	985	2000	900
1974	470	1987	995	2001	886
1975	485	1988	970	2002	636
1976[b]	485	1989	935	2003[c]	501

a. Sources: Vejvoda (2005), Meyers (2013), DOE (2014a).
b. Inventory values reported for FY 1976 and FY 1976A correspond with a change in the federal FY to begin October 1. The U.S. budget year began July 1 before this time (ORAUT 2013a, p. 2; OMB 2012, p. 8). The values here agree with those reported elsewhere (Meyers 2013) for the June 30 and September 30, 1976 quarterly inventory periods.
c. Last reported inventory was on June 30, 2003.

It is difficult to draw firm conclusions about neptunium processing from the inventory table above. The data are presented primarily to show that the total neptunium inventory at RFP was always relatively low, typically less than a kilogram (compared to much greater inventories of plutonium for the same period). For example, the RFP site description TBD states that Building 771 recovery operations were processing 240 kg of plutonium daily by 1965. This daily plutonium throughput exceeds any single neptunium processing run by 2 to 3 orders of magnitude (ChemRisk 1992). A look at the year-to-year values reveals that inventory variability was higher in the earlier years and more stable after the mid-1980s.

E.4 MEASUREMENTS OF NEPTUNIUM WASTE AT THE IDAHO NATIONAL ENGINEERING AND ENVIRONMENTAL LABORATORY

As shown later in Table E-5, RFP shipped neptunium to several DOE sites, including neptunium-containing waste to the Idaho National Engineering and Environmental Laboratory, now known as the Idaho National Laboratory (INL). A recent interviewee (ORAUT 2014d) identified two volumes of a 2004 document titled *Validation of the Rocky Flats Plant Radionuclide Inventory in the Historic Data Task Using SWEPP Assay Data* (Blackwood 2004; Blackwood and Hoffman 2004). Volume 1 of that document provides nondestructive assay (neutron and gamma) measurements of RFP waste stored at the INL Radioactive Waste Management Complex (RWMC). Table E-2 presents measurements showing that, in drums containing ^{237}Np, ^{239}Pu was also present at ^{239}Pu:^{237}Np mass ratios ranging from 105 to 6,450. Table E-3 presents measurements showing that the ^{239}Pu:^{237}Np mass concentration ratios ranged from 109 to 5,820. The predominance of ^{239}Pu in the waste reflects the prevailing ratios in the RFP operating environment where neptunium was processed. Potential organ doses to workers packaging and handling this waste would, as previously discussed, be dominated by the plutonium contribution.

ATTACHMENT E
EVALUATION OF THE POTENTIAL FOR INTERNAL DOSE FROM NEPTUNIUM-237 (continued)

Table E-2. Mean masses and ratios of ^{239}Pu and ^{237}Np in waste drums.[a,b]

Waste matrix	Pu-239 *n*	Pu-239 mean (g)	Pu-239 σ (g)	Np-237 *n*	Np-237 mean (g)	Np-237 σ (g)	Pu-239/Np-237 isotopic mass ratio
Graphite	1,307	2.60×10^{1}	2.75×10^{1}	4	4.03×10^{-3}	5.78×10^{-5}	6.45×10^{3}
Filters	68	2.76×10^{1}	3.16×10^{1}	54	5.07×10^{-3}	3.19×10^{-3}	5.44×10^{3}
Mixed metals	523	1.46×10^{1}	3.14×10^{1}	14	1.82×10^{-2}	3.64×10^{-2}	8.02×10^{2}
First- and second-stage sludge	3,095	6.17	8.74	281	5.86×10^{-2}	8.01×10^{-2}	1.05×10^{2}

a. Source: Graphite: Blackwood and Hoffman (2004, p. 75); filters: (p. 81); mixed metals: (p. 87); first- and second-stage sludge: (p. 93).
b. Data in this table are compiled from measurements of radioactive waste stored at the INL RWMC. The symbols *n* and σ are the number of measured drums and standard deviation, respectively, for the log-transformed data.

Table E-3. Mean mass concentrations and ratios of ^{239}Pu and ^{237}Np in RFP waste drums.[a,b]

Waste matrix	Pu-239 *n*	Pu-239 mean (g/kg waste)	Pu-239 σ (g/kg waste)	Np-237 *n*	Np-237 mean (g/kg waste)	Np-237 σ (g/kg waste)	Pu-239:Np-237 isotopic mass concentration ratio
Graphite	1,307	3.67×10^{-1}	4.16×10^{-1}	4	6.31×10^{-5}	5.78×10^{-5}	5.82×10^{3}
Filters	68	1.06	1.03	54	2.96×10^{-4}	3.80×10^{-4}	3.58×10^{3}
Mixed metals	523	2.43×10^{-1}	4.75×10^{-1}	14	6.92×10^{-4}	1.99×10^{-3}	3.51×10^{2}
First- and second-stage sludge	3,095	3.44×10^{-2}	5.10×10^{-2}	281	3.15×10^{-4}	4.05×10^{-4}	1.09×10^{2}

a. Source: Graphite: Blackwood and Hoffman (2004, p. 76); filters: (p. 82); mixed metals: (p. 88); first- and second-stage sludge: (p. 94).
b. Data in this table are compiled from measurements of radioactive waste stored at the INL RWMC. The symbols *n* and σ are the number of measured drums and standard deviation, respectively, for the log-transformed data.

E.5 INTERNAL DOSE MONITORING

Captured documents contain only two reported neptunium bioassays. A urinalysis result in June 1966 was "Below Significant Level" (Hammond 1966a, p. 79; Dow 1966–1969, p. 8). A urinalysis result in July 1966 reported 0.9 dpm/24 hr; a followup body count showed no detectable uptake (Hammond 1966b, p. 83; Dow 1966–1969, p. 10).

E.6 WORKPLACE AIR MONITORING

No neptunium-specific workplace air monitoring occurred.

E.7 CONTAINMENT MEASURES DURING OPERATIONS

The neptunium processing descriptions described in, *Neptunium Processing at the Rocky Flats Plant* (Connor and Basso 1981) support the understanding that, by 1983, all neptunium operations were being performed in glovebox facilities. For example, the document states (Connor and Basso 1981, p. 12):

> *A typical glove box for aqueous Np processing consisted of a "wet" section (for aqueous processes) and a "dry" section (for calcining precipitates and weighing powders) separated by an air lock. Each section had separate air inlet and exhaust filters. A door in the air lock was used to pass equipment and material between the wet and dry sections. A ⅛-inch-thick lead sheet was bonded to the stainless steel portion of the glove box and ¼-inch leaded glass was placed over the glove box windows as a shield against gamma radiation from the neptunium and its impurities.*

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EVALUATION OF THE POTENTIAL FOR INTERNAL DOSE FROM NEPTUNIUM-237 (continued)

Only one neptunium processing operation is known to have occurred after 1983. It involved the separation of neptunium from plutonium residues in 1985 (see Section E.3). According to the Principal Engineer who designed the process and directed activities, this operation was performed in gloveboxes and tanks. The tanks contained feed material (plutonium and neptunium nitrate solution) that was piped directly into the gloveboxes (ORAUT 2014b).

E.8 WORKPLACE INCIDENTS

No radiological incidents at RFP involving neptunium have been identified.

E.9 SHIPMENT AND RECEIPT DATA

NMMSS records of neptunium shipments and receipts by RFP from 1983 until site closure in 2003 (DOE 2014b to 2014e) were obtained and analyzed. Table E-4 shows the FYs during which gram amounts of neptunium were received at RFP from Los Alamos National Laboratory (LANL), SRS, ORNL, and LLNL. No material was received after FY 1986. The three-letter designations in parentheses after the site acronyms are the DOE Reporting Identification Symbol codes for the sites. Neptunium shipments from RFP to other DOE sites are shown in Table E-5.

Table E-4. Neptunium receipts from other sites (g).

FY	LANL (AUA)	SRS (DZA)	ORNL (FZC/FZG)	LLNL (LZB)
1983	0	556	199	105
1984	0	0	199	1
1985	99	0	0	228
1986	1	113	0	0

E.10 POTENTIAL FOR EXPOSURES AFTER 1983

From 1962 through 1983 there existed the potential for unmonitored worker exposures to ^{237}Np. There was insufficient data available to estimate intakes of ^{237}Np during that period.

After 1983, only one operation involving neptunium (Section E.3). Workplace monitoring for this 1985 project included continuous air monitors (CAMs) and contamination surveys typical of the plutonium processing areas. Radiation Monitors (technicians) were available to provide radiological safety support. Personnel involved in the project were on bioassay programs typical of the plutonium processing environment (i.e., routine urinalyses and body counts) (ORAUT 2014b).

The previous discussion noted that this operation did not involve purified neptunium, but rather plutonium with neptunium as a contaminant. Even the most highly concentrated neptunium product produced by this separation process was still mostly plutonium, with a Pu:Np mass ratio of 6.4. Because the specific activity of ^{239}Pu is about 90 times greater than that of ^{237}Np, the activity ratio of this PuNp mixture is greater than 500. As a result, all organ doses from an intake of the mixture would be dominated by the plutonium component. The neptunium dose component would contribute only about 0.1% for type M plutonium for most organs, and only about 1% for type S plutonium (ORAUT 2014e).

ATTACHMENT E
EVALUATION OF THE POTENTIAL FOR INTERNAL DOSE FROM NEPTUNIUM-237 (continued)

Table E-5. Neptunium shipments from RFP to receiving sites (g).[a]

Fiscal year	LANL (AUA)	SRS (DZA)	ORNL (FZC/FZG)	Hanford (HRA)	LLNL (LZB)	Nevada Test Site (NAB)	WIPP (VPA)	INL Waste (VSB)
1983	0	0	378	0	220	40	0	0
1984	0	0	0	0	159	0	0	0
1985	0	0	0	0	146	26	0	22
1986	73	0	0	2	0	0	0	0
1987	1	0	0	0	0	0	0	0
1988	2	8	0	0	0	0	0	0
1989	0	36	0	0	0	0	0	0
1990	0	0	0	0	0	0	0	0
1991	0	0	0	0	0	0	0	0
1992	0	0	0	0	0	0	0	0
1993	0	0	0	0	0	0	0	0
1994	0	0	0	0	0	0	0	0
1995	0	0	0	0	0	0	0	0
1996	0	0	0	0	0	0	0	0
1997	5	0	0	0	7	0	0	0
1998	0	0	0	0	0	0	0	0
1999	0	46	0	0	0	0	1	0
2000	0	18	0	0	0	0	0	0
2001	0	0	0	0	0	0	174	0
2002	0	297	0	0	0	0	324	0
2003	0	394	0	0	0	0	11	0

a. WIPP = Waste Isolation Pilot Plant.

As mentioned previously, interviews with the Principal Engineer and coauthor of the 1987 document *Production-Scale Plutonium-Neptunium Separation and Residue Recovery at Rocky Flats Plant* (Martella, Guyer, and Leak 1987) indicated that everyone who worked in a "hot" building had a urinalysis every few months as well as annual whole-body counts (ORAUT 2014a, 2014b). If this individual or any of his coworkers had received neptunium intakes from this operation, the resulting organ doses would be adequately accounted for by the plutonium bioassay program. Plutonium bioassay data are readily available for RFP claimants.

No other specific operations involving neptunium were identified, other than the 1985 operation. If other operations of a similar nature (i.e., neptunium associated with plutonium) did take place, plutonium bioassay would have similarly accounted for any doses due to neptunium.

E.11 CONCLUSION

In conclusion, there is no evidence that ^{237}Np intakes occurred at RFP after December 31, 1983. If intakes had occurred during this period from the one known neptunium operation, the resulting organ doses would be adequately accounted for by the available plutonium bioassay data.

ATTACHMENT F
REPORTED PRESENCE OF MAGNESIUM-THORIUM ALLOY

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ATTACHMENT F
REPORTED PRESENCE OF MAGNESIUM-THORIUM ALLOY (continued)

F.1 INTRODUCTION

Workers at RFP have provided statements that MgTh alloy plates were used, handled, or processed on site at RFP. One worker stated (Kusmierczak 2007):

> *Earlier this month a former Rocky Flats worker related to me thru a third party (who wants to remain anonymous) information concerning the use of magnesium thorium alloy plates at Rocky Flats. You may remember that Dow workers submitted affidavits that Dow shipped these plates to Rocky Flats.*
>
> *The information relayed to me was, "They were brought in on the 903 pad send* [sic] *to 881 to refine then sent to the MOD center for modification to fit the Semi Trucks as to make them bullet proof. A Semi bed was brought in stripped down and the sheets were refined to help armor plate the trucks for transport of materials. He also said they were used in the Train cars for support!"*

The affiant indicated that work was performed with this thorium material at Dow Chemical Company in Madison, Illinois, from 1962 to 1975 but did not provide any specific information on when the shipments might have been made to RFP. No corroborating evidence for the assertion that MgTh alloys were used or present at RFP was found during the research into MgTh alloy efforts in the DOE complex over the applicable period. RFP personnel who would have been aware of the receipt of these types of materials were interviewed and none were aware that MgTh alloy was ever present or used in any significant quantity at RFP (SC&A 2007a, pp. 66–67).

Two other Colorado facilities in the proximity of RFP handled nuclear materials and specifically handled MgTh alloys, nuclear weapons, or related equipment that would have used MgTh alloys in their designs (Rocky Mountain Arsenal and Lockheed Martin-PJKS Test Facility). The Dow Madison individual who had stated that MgTh alloys were sent from Madison to RFP was interviewed and asked about the possibility that the MgTh could have been shipped to one of the other locations rather than RFP. The individual indicated that this was a possibility and that he was not aware of the existence of the other two facilities.

F.2 MAGNESIUM-THORIUM SEARCH RESULTS

An initial search of documents was conducted to determine if a link could be established between MgTh alloy and RFP. This search included all listed sites and identified the following documents containing information on MgTh alloy:

- Owen 2007. A Freedom of Information Act (FOIA) request "for records that pertain to shipments of thorium-magnesium alloy metal between the Dow Chemical Company at its Madison, Illinois, facility or at the Walnut Creek-Pittsburgh, California, dual facility and the Rocky Flats nuclear weapons facility in Colorado between **1951 and 1973** [emphasis added]" resulted in a response from the DOE Office of Legacy Management identifying 10 shipping orders (Dow 1957a to 1957d, 1958a to 1958d, 1959a, 1959b). The identified documents, however, are all for shipments of plutonium samples from RFP (Dow) to SRS (DuPont) from **1957 to 1959**.

- DOE 2007. SRS responses to FOIA requests that contain request from an East Alton, Illinois attorney about MgTh alloy metal.

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REPORTED PRESENCE OF MAGNESIUM-THORIUM ALLOY (continued)

- SC&A 2007b. The SC&A review of the evaluation report for SEC-00030 (NIOSH 2006) for RFP has no estimate of dates when MgTh might be present at Rocky. Their finding (p. 18) states, "It is clear from NUREG-1717 and the other considerations presented above that knowledge of the approximate quantities, **periods**, and processing status of the MgTh alloy is needed before any reliable conclusions can be arrived at about doses to RFP workers from this material [emphasis added]."

- Author unknown undated a. *Production of Magnesium Thorium Alloy, 1954-73*, with production data from the Minerals Yearbook Metals and Materials, indicates that Dow Madison produced MgTh alloy from **1954 to 1965 and from 1967 to 1973** [emphasis added].

- SC&A 2007b. *A Focused Review of Operations and Thorium Exposures at the Dow Chemical Company Madison Plant*, in which four former RFP managers state that MgTh alloy was never used in any process for development or fabrication of weapons components.

- SRGA 2008. The minutes of the Advisory Board on Radiation and Worker Health (ABRWH) Meeting 56 in June 2008 on the Dow Madison SEC petition state that DOE concluded that Dow made MgTh alloys HK31A and HM21A for use in nuclear weapons in **1957 and 1958** (p. 32) and that MgTh alloys were used in atomic weapons from **1956 to 1969** (p. 38) [emphasis added].

Additional research and reviews were conducted on topics potentially related to the presence of MgTh alloys at RFP. Searches were performed with key words that included the terms "magnesium" and "thorium" as well as their acronyms. These searches identified further search terms, including HK-31 and HK-31A (codes used to identify MgTh). These additional searches identified the following documents about MgTh operations at Dow Madison in Illinois, Wyman-Gordon in Massachusetts, and the Kansas City Plant in Missouri, but none of these relates to RFP:

- Silverstein 1957 (Dow Madison). Dow memorandum about firefighting concerns associated with MgTh alloys.

- Peloubet 1959 (Dow Madison). Letter from Dow to the U.S. Atomic Energy Commission (AEC) about ventilation for MgTh welding.

- Silverstein 1958 (Dow Madison). Dow spectrographic analysis results for dust samples obtained from sanding HK-31.

- Mitchell 1958b (Dow Madison). Analytical results for air samples obtained during hand sanding HK-31.

- Levy 1957 (Dow Madison). Dow memorandum making a correction to a previous letter about exposure to HK-31.

- Dow 1957e (Dow Madison). Dow memorandum explaining the new federal law passed in January 1957 that placed a limit on the external radiation a worker could receive.

- Mitchell 1957 (Dow Madison). Dow memorandum documenting sample turnover of two dust samples collected during hand sanding of HK-31.

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REPORTED PRESENCE OF MAGNESIUM-THORIUM ALLOY (continued)

- Mitchell 1958 (Dow Madison). Dow memorandum documenting sample turnover of three dust samples collected during sanding of HK-31 with vertical metallurgical sanding discs.

- Author unknown undated b (Dow Madison). Undated description of the composition, hardness, and strength of HK-31A material.

- GZA 1994 (Wyman-Gordon). Report on a 1993 survey of MgTh residual contamination on the Wyman-Gordon site.

- Various 1961–1975 (Kansas City Plant). A variety of operational documents from the 1960s and 1970s about MgTh alloy and a Bendix association to Dow Madison.

- BAC 1970 (Kansas City Plant). Two memoranda discussing fabrication of MgTh units to be machined as part of the Radiation Detection Capability Program.

- BAC 1972 (Kansas City Plant). Two memoranda listing personnel who had received physicals, orientation, or clearances to work with MgTh waste.

- BAC 1957–1970 (Kansas City Plant). Document specifying the health and safety procedures to be followed when handling MgTh (2% Th).

An additional site visit and data capture was performed at the Environmental Management Consolidated Business Center (EMCBC) in Denver, Colorado. It included classified document reviews and interviews and generated the following MgTh results that do relate to RFP:

- ORAUT 2013b. EMCBC documents on the safe, secure transport (SST) and safe, secure railcar from the Atomic Materials Rail Transport Car system Modification Center, Books 1-4 (documents classified); classified document reviews for MgTh with the unclassified and released notes indicating no related information was identified.

- ORAUT 2013c. EMCBC interview; discussed SST and SSR work and provided photos of related operations; performed as a classified interview with the unclassified and released notes indicating that no MgTh-related information was identified.

- ORAUT 2013d. EMCBC railcar procedures; classified document reviews for MgTh with the unclassified and released notes indicating no related information was identified.

- ORAUT 2013e. EMCBC documents on the SSR; classified interviews for MgTh with the unclassified and released notes indicating no related information was identified.

- ORAUT 2013f, 2013g; NIOSH 2013b. Interviews about MgTh and any related operations. No related information was identified.

- NIOSH ca. 2013. Interview about MgTh operations. No related information was identified.

- Author unknown 1976b. RFP thorium shipment and receipt records for 1952 to 1976 with no indication or support of MgTh operations.

As indicated in the interviews, the armor or shielding plate installation operation associated with the potential use of MgTh alloys was performed at a specific location at RFP. Individuals who were

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REPORTED PRESENCE OF MAGNESIUM-THORIUM ALLOY (continued)

directly involved in the work were interviewed about the use of MgTh alloys. In addition, documentation and procedures associated with the operation were reviewed. While most of the information and documentation remains classified, the information that was released based on the reviews does not corroborate the use of MgTh alloys at RFP for this or any other operations.

Followup interviews and information indicates that RFP might have collaborated with Sandia National Laboratories (SNL) on the design of the armor plating installed on vehicles and equipment at RFP. Because SNL documentation about the armor plating could contain information on plating composition, a keyword search of classified and unclassified SNL documents for anything about this operation was performed. SNL-Albuquerque and SNL-Livermore indicated that no related documentation was discovered in their searches.

F.3 CONCLUSION

Based on the research for this assessment, the ORAU Team concludes there is no corroborating evidence for the use of MgTh at the RFP site. It is likely that there is confusion between RFP and other Denver-area sites, as well as confusion about MgTh alloy plates and other similar materials at RFP. All of the available information for MgTh alloys for other sites falls within the period of the SEC class for RFP (1952 to 1983). Therefore, if any undocumented MgTh operations did occur at RFP that related to these other sites' operations, the RFP operational dates would be covered under the existing RFP SEC class.

ATTACHMENT G
TRITIUM ISSUES

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TRITIUM ISSUES (continued)

G.1 INTRODUCTION

The potential for tritium exposure to RFP personnel was not considered significant until an unexpected release occurred in April 1973. Because tritium monitoring was not rigorous before this event, a tritium bounding method was developed for this analysis that uses information from the 1973 tritium incident as the maximum exposure scenario.

G.2 MONITORING DATA

Although tritium was used as a boost gas in weapons and as target material in neutron generators, it was not processed or handled in significant quantities at RFP. Tritium was monitored in the environment around the site for a time, but that monitoring ceased and was left to the State of Colorado for a brief period before an environmental release that occurred in April 1973. No analytical records have been found that might help establish the RFP workplace tritium environment before 1973.

The management of RFP responded quickly to the potential for tritium workplace and environmental contamination with the release in April 1973 of 500 Ci to 2,000 Ci of tritium, primarily from Building 779A, and its eventual detection in waters draining into a reservoir serving as a municipal drinking water supply (AEC 1973; Arkell 1976). The release did result in tritium exposure to a small number of RFP personnel. Subsequent workplace monitoring and personnel bioassay was implemented, in part to establish the baseline tritium environment against which future incidents could be evaluated. A smaller and less impactful tritium release occurred in September 1974 from Building 777. The subsequent investigation report (AEC ca. 1974) includes release details along with summaries of tritium workplace monitoring results before the incident for comparison. These data provide the basis for a model for bounding chronic tritium exposures to workers and of smaller, less-notable tritium releases that might have occurred before 1973.

Several factors single out the 1973 tritium release as bounding for the entire history of RFP operations. These factors include the large quantity of tritium involved, the chemical form of the released tritium, and the meteorological conditions at the time of the release. Other documented releases involved smaller quantities of elemental tritium having a much smaller DCF than the tritium oxide released in 1973. Bounds for personnel tritium exposures after the 1973 release can be developed based on measurement results from the personnel bioassay, air sampling, and workplace contamination monitoring data for tritium, which became more common after that release. There are only very limited tritium measurement results before 1973 because tritium was not perceived as a radionuclide of occupational or environmental interest at RFP. Methods for bounding tritium exposures before 1973 are more difficult to develop as a result of this lack of measurement data.

According to a ChemRisk report (1994), there was no environmental monitoring for tritium before 1970, and little in the way of workplace monitoring until after the 1973 tritium release; therefore, evidence of tritium releases before 1973 is primarily anecdotal. There was a 600-Ci release of elemental tritium from a different source in 1968. However, that release is not related to the tritiated plutonium shipments. The report said the following about possible releases from tritiated plutonium shipments (ChemRisk 1994, p. 285):

> *The 1973 findings associated with the tritiated plutonium initiated an investigation of other possible similar shipments and processing of tritiated plutonium. The investigation discovered three other shipments with maximum estimated tritium releases of 57 Ci (April 1969), 40 Ci (March 1971), and 29 Ci (November 1971).*

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TRITIUM ISSUES (continued)

The reported investigation and the documented 1968 release of elemental tritium are the only sources of information about other possible releases. The 1968 release was elemental tritium with no significant environmental or personnel exposure. None of the three identified potential releases from tritiated plutonium was near the magnitude of the 1973 release. There is no evidence of a tritium release comparable to the magnitude and effects of the 1973 release before that year.

Despite the lack of measurement data, it is possible to develop tritium exposure bounds for before 1973 based on measurement results in a Rocky Flats Area Office (RFAO) report after a tritium release in one of the RFP production buildings on August 30, 1974 (AEC ca. 1974). The information in this report includes measurement data (i.e., results from air samples, surface contamination surveys, and bioassay) from the production area where the release occurred as well as comparison data from other areas before, during, and after the release. Several factors support the use of these data as surrogates for bounding the tritium environment at RFP before 1973:

- Background tritium levels immediately before the incident, although undoubtedly elevated since the more significant 1973 release, were well below dosimetrically significant values and can be considered as fairly representative of typical background levels for this analysis. The background tritium levels monitored in the months before the 1974 incident are consistent with internal radiation doses from tritium of well under 1 mrem annually. They are dosimetrically insignificant in this sense.

- The quantity of released tritium (1.5 Ci) was significantly less than that in 1973 and was probably more typical of potential undocumented releases in work areas, particularly those from opening contaminated shipping containers.

- The 1974 1.5-Ci tritium release is the only documented release from a shipping container in the RFP workplace. It is taken to be typical because there are no other such documented releases to use in forming the model. There is documented concern about tritium releases, as noted by ChemRisk (1994, p.38):

  > *As early as 1962, Rocky Flats maintained instruments for detection of tritium gas in particular work areas of the plant because operations have sometimes resulted in the storage of tritium containers.*

The instruments available to RFP at that time were only semi-quantitative for indicating the presence of tritium; NIOSH has found no records of these results. Because NIOSH has only identified six documented releases from 1968 to1974 (an average of 1 per year), the application of a daily release would be a significant, and therefore bounding, overestimate of the number of RFP tritium releases.

- Tritium was released to the workplace environment rather than in a glovebox.

- The release involved elemental tritium (HT) but not tritium oxide [as tritiated water vapor (HTO)].[1]

[1] The impact of the 1973 tritium release was largely due both to the quantity (500 Ci to 2,000 Ci) and the chemical form (HTO) of the material. The presence of tritium oxide in the 1973 release resulted from peculiarities of the plutonium recovery operation from which it came. There is no indication that any other tritium release at RFP involved the oxide. Tritium in its elemental form (HT, T_2) is far more likely to have been a contaminant because of the nature of its possible RFP source terms: tritiated accelerator targets (neutron generators), plutonium hydride in recovery operations, and boost gas in returned reservoirs or pits.

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TRITIUM ISSUES (continued)

- The tritium was released from a contaminated shipping container that was procured by RFP in 1970 and can be taken as representative of shipping containers in use before 1973.

- The incident occurred close enough in time to the 1973 tritium release that work practices and controls were likely more similar to those before 1973 than to those even 1 or 2 years later, as procedures and controls evolved with greater sensitivity to the potential for tritium contamination.

The RFAO report provides the best source of monitoring data for use in bounding both chronic and accidental tritium exposures to RFP personnel before the unique circumstances of the 1973 release. The report states that elevated tritium concentrations were detected in air samples from Room 452 (Special Assembly Area) in Building 777 and from the Building 205 exhaust plenum servicing Building 776/777 from August 30 to September 4, 1974 (AEC ca. 1974, p.36). Subsequent sampling and investigation of the elevated sample results concluded that about 1.5 Ci of tritium was released from the exhaust system of Room 452, Building 777, when a shipping container (referred to as a "pressure cooker") received in July 1974 was opened on a downdraft table in Room 452 on August 30 (AEC ca. 1974, pp. 36–39). No elevated environmental tritium levels were detected as a result of the incident, but workplace tritium levels 7 times the applicable radioactivity concentration guide were detected in air samples collected on August 30 in Room 452 adjacent to the downdraft table, with average concentrations for the work week about 1.5 times the guide. Table G-1 below shows the reported values.

Table G-1. Reported tritium air concentrations from the August 30, 1974, release ($\mu Ci/m^3$).[a]

Sampling reference	Plenum 205, Building 776/777[b]	Room 452, Building 777
Normal concentrations	$<1 \times 10^{-2}$	$<1 \times 10^{-2}$
August 29–30, 1974	0.148	37.7
September 3–4, 1974	2.51	1.1

a. Source: AEC (ca. 1974, pp. 9, 93–96).
b. The Special Assembly Glovebox Line in Room 452, Building 777, was normally served by Plenum 206, but exhaust air from this area was vented through Plenum 205 from February 11 to August 7, 1974, while a new Plenum 206 was constructed. A tritium air sampler for Plenum 206 was installed on August 30, 1974, but showed no elevated results. However, both plenums showed elevated removable tritium contamination (AEC ca. 1974, pp.74-82).

Air Sample Results

Results from daily air samples in Room 452, Building 777, are available from June 3 to September 11, 1974. The air sampler was near the downdraft table entry to the Special Assembly Line where the tritium-contaminated "pressure cooker" was opened, and it was the only tritium air sampler in Building 777 at the time. Room air samples were collected in a water bubbler during the day shift (approximately 6 to 8 hours sampling time) at an air flow rate of 2 L/min.

The average and standard deviation of daily air sample results before August 30, the day of the tritium release from the contaminated shipping container, are 5,343 ±4,518 pCi/m^3 (AEC ca. 1974, pp. 87–89). The result on August 30 is 37,676,609 pCi/m^3, and the sample taken on September 3 indicated a tritium concentration in the room air of 1,098,901 pCi/m^3. However, the September 3 result is suspect because the sample was collected in the same vessel that was used on August 30, which had not been cleaned. Smear surveys of Room 452 on September 3 failed to show significant tritium contamination (AEC ca. 1974, pp.37–38). Tritium levels in Building 777 were known to be somewhat

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TRITIUM ISSUES (continued)

elevated over normal background because of residual contamination present since the 1973 tritium release.

Bioassay Results
The practice of pulling a sample of air from within shipping containers through a tritium air monitor to check for contamination was implemented after the 1973 tritium release. This practice was discontinued after urinary tritium results in the range of 0.75 to 1.3 µCi/L were detected in May 1974 for the [redacted] who performed the monitoring. The [redacted] urinary tritium dropped to less than 0.1 µCi/L beginning in early July 1974 (AEC ca. 1974, pp. 18–19).

Everyone who worked in Room 452, Building 777, submitted urine samples after the August 30 tritium release, with a high result of 32,320 pCi/L (AEC ca. 1974, p. 90).

The report indicates that both a Denver resident and a RFP worker who did not work in radioactive material-handling areas were sampled with results <0.01 µCi/L (<10,000 pCi/L) (AEC ca. 1974, p.30). The Denver resident is identified in Table A1-3 of that report, and Worker ID [redacted] is believed, by implication, to be the RFP nonradiological worker.

Work Area Smear Surveys
Over 200 smear results for tritium are tabulated in the RFAO report (AEC ca. 1974, pp. 74–82). Most appear to be surveys inside gloveboxes, but there are also workplace area results that can be used as indicators of likely sources of internal contamination of workers after an event such as that in August 1974. There were 42 workplace smear results. They range from <100 pCi/smear to 1×10^7 pCi/smear, with an average of about 8.2×10^4 pCi/smear.

The exhaust plenums and the Kathabar air driers (which use a lithium chloride solution called Kathene) appear to have collected the greatest amount of tritium after the release. Workers responsible for changing filters in the plenums or recharging the Kathabar systems would appear to be at greatest risk for tritium uptake after the initial release.

Assessment of the 1974 Incident
The 1.5-Ci tritium release from a contaminated shipping container occurred on August 30, 1974. The RFAO report provides air survey, bioassay, and smear survey results (AEC ca. 1974). Specific urine sample collection dates were not included in the report, but data were matched to two NOCTS claims that reported a collection date of September 5, 1974. A dose assessment was performed assuming an intake date of August 30, 1974, and the largest reported bioassay result collected after the incident, 36,320 pCi/L. There was a slight discrepancy (one digit) between the result in the RFAO report and that in the NOCTS case file. The NOCTS value is assumed to be correct because it is the handwritten urinalysis record and is also the larger of the two values. The resulting dose is <1 mrem (0.15 mrem).

Conclusion
The RFAO report of a 1.5-Ci tritium release on August 30, 1974, from a contaminated shipping container (“pressure cooker”) provides air survey, bioassay, and smear survey results that can be used to model similar releases (AEC ca. 1974). Assuming an event like this one occurred every workday of every year (i.e. 250 times a year), the resulting annual dose would be 37.5 mrem/yr. This should be assigned as a bounding estimate for all workers’ potential tritium exposure before 1973.

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TRITIUM ISSUES (continued)

G.3 DOSE ASSIGNMENT FOR 1973

The *Investigation of the Tritium Release Occurrence at the Rocky Flats Plant* (AEC 1973) describes the 1973 incident, which prompted the site to sample a number of workers for tritium exposure. A shipment of scrap plutonium from LLNL was discovered to have been contaminated with tritium. This material was processed at RFP from April 9 to 25, 1973, in Building 779A. Because it was not immediately identified as being contaminated, monitoring of potentially exposed individuals did not begin until late September 1973.

Two-hundred-fifty people were sampled after the discovery; this included all personnel who worked in areas in which the contaminated scrap was processed or who were involved in the processing of wastes from this scrap. Due to the large sample load, raw urine samples were first analyzed in many of the cases. It was noted that the counting efficiency was only about 3% for these analyses, and that the corrections made for spectral shift could lead to abnormally high readings. Nineteen workers were initially identified as having elevated tritium levels in their urine. These samples were distilled and reanalyzed, which found that fourteen of these workers had levels below the 10,000 pCi/L action level established by the site. The [redacted] most exposed individuals were identified and details of their potential exposures, including bioassay results, are included in the above-cited investigation report. [redacted] of these [redacted] individuals can be found in NOCTS.

Tritium contamination was associated with plutonium scrap material; therefore, ^{3}H doses should be assigned to all individuals who were monitored for plutonium in 1973. Doses should be assessed on an individual basis using reported ^{3}H bioassay results and any additional information in the NOCTS file. For those who were not monitored for ^{3}H, dose should be assigned based on assumptions that are favorable to the individual claimant.

The tritium release incident report (AEC 1973) discusses the tritium sampling of 250 people as of October 15, 1973. There are no results or specific sample dates in the report for those individuals who did not exceed the ^{3}H bioassay action level of 10,000 pCi/L. Only [redacted] individuals were found to have results exceeding the action level, so this current assessment assumes that they are the maximally exposed workers in the incident and that those who were not monitored would not have been exposed at those same levels. After discussions about the tritium model at the RFP ABRWH Working Group meeting, NIOSH decided to adopt the use of the tritium model described in Richardson and Dunford (2001) for this assessment.

Based on this information, the following assumptions were applied:

- ^{3}H was in the form of HTO;
- Model for inorganic ^{3}H is described by Richardson and Dunford (2001);
- Mode of intake was injection (for modeling with IMBA);
- Intake date was April 9, 1973, the first day the material was processed;
- Sample was collected on October 15, 1973; and
- Result was 10,000 pCi/L (14,000 pCi/d).

Using this information, the total dose is 103 mrem.

G.4 DOSE ASSIGNMENT AFTER 1973

The ORAU Team performed a co-exposure study using data from NOCTS for 1974 and 1975 in accordance with ORAUT-OTIB-0075, *Use of Claimant Datasets for Coworker Modeling* (ORAUT

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TRITIUM ISSUES (continued)

2016). There are 38 individuals with tritium data in 1974 and 37 in 1975. When assessing tritium intakes for most sites, it is assumed that intake potential existed only for these individuals with tritium data in 1974 and 1975.

When assessing tritium intakes for most sites, it is assumed that intake potential existed only while tritium bioassay monitoring was being performed because monitoring is cheap, easy, and requires only spot samples, which is less of a burden than other forms of bioassay on both the employer and the worker. Because tritium was not of primary concern at RFP and was present only as a potential contaminant on equipment, a given individual was not placed on a routine sampling program. Instead, a program was established whereby one-tenth of the urine samples collected for plutonium analysis were also analyzed for tritium content (Bowman 1974) as well as the collection of samples when a particular concern was identified. Samples available in NOCTS for these 2 years indicate that analyses were performed throughout the year, with most individuals sampled only once.

For the purpose of the co-exposure study, it was assumed that each worker had the potential to be exposed at a constant level throughout the year in which the urine sample was collected. The 95th percentile was used because one-tenth of the population was sampled. The co-exposure study for 1974 to 1975 yielded doses of much less than 1 mrem for everyone.

For the years after 1975, there are fewer than 11 claims in NOCTS with tritium data, which is insufficient for performing a co-exposure study. Results for these years are consistent with those from the previous years and show a generally decreasing personnel exposure trend.

Therefore, after 1973 no additional unmonitored dose due to tritium should be assigned to RFP workers.

ATTACHMENT H
CRITICAL MASS LABORATORY

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CRITICAL MASS LABORATORY (continued)

H.1 INTRODUCTION

This attachment assesses the potential internal exposure at the RFP CML. The CML operated at a maximum reactor power of 10 mW. A typical experiment duration of 1 hour was used to estimate the MFAP inventory built up over time in highly enriched uranium (HEU) solution fuel.

H.2 PERSONNEL EXPOSURE CONCERNS

Fission products in irradiated fuels and activation products in both the fuel and containment materials are sources of external radiation dose to personnel using or working around the fuels, and they present an internal dose potential for personnel who might ingest or inhale them. Personnel dosimeters assigned to RFP radiation workers document the external exposures. Internal exposures might result (1) during operations from resuspension of contamination on surfaces, or (2) during facility demolition from airborne dust. CML staff was provided routine bioassay (urinalysis and whole-body counts) to detect intakes of plutonium, uranium, or americium, but MFAPs were not routinely monitored.

Surface contamination in CML experimental areas was extensive and predominantly due to spills of HEU in the form of UNH solution[2] over the course of the facility's history (Rothe 2005, pp. 447, 449, 452, 454–458, 464, 467–471, 479, 486, 498, 500–502). Therefore, the MFAPs important from either an acute or chronic internal exposure perspective are those from CML criticality experiments involving HEU solution. MFAPs atom ratios on contaminated surfaces in the CML and resuspended in air from these surfaces are presumed to be the same as those in solution.

H.3 HISTORY OF URANYL NITRATE CRITICALITY EXPERIMENTS

Criticality experiments involving UNH solution were conducted from the beginning of CML operations until the last experiment in 1987. These experiments are documented in some detail in a published history of the facility (Rothe 2005, pp. 376–394). The 778 experiments performed at the CML with HEU solution or systems of HEU solution plus HEU metal (Oy) spheres took place in 10 campaigns between May 1967 and October 1987.

H.4 REEVALUATED FISSION AND ACTIVATION PRODUCT LEVEL ESTIMATES

MFAP buildup in uranium solution experiments at the CML was calculated using ORIGEN-S, a program for calculating time-dependent concentrations of radionuclides that are simultaneously generated or depleted by processes such as fission, neutron absorption or transmutation, and radioactive decay. Initial assumptions about power levels and experiment durations were taken from a DOE document (DOE undated, p. 3) that stated that the experiments conducted in Building 886 (housing the CML) generally involved power levels of no more than 10 mW for no more than 1 hour. It also stated that approximately half of the experiments in Building 886 actually achieved criticality, and only rarely were the radiation levels such that it was not possible to directly touch the fissile material and testing apparatus immediately after the experiments.

RFP documents contain neutron flux and thermal power estimates by CML staff using analytical measurements made during or after criticality experiments. These calculations were performed for at

[2] Eleven spills involved solutions or dried salts from solution experiments, compared with two contamination events involving other solid fuels (one spill each of low-enriched uranium and plutonium powders).

ATTACHMENT H
CRITICAL MASS LABORATORY (continued)

least three different EU fuel configurations: (1) HEU Oy spheres, (2) Oy spheres immersed in UNH solution; and (3) UNH only.

H.4.1 Thermal Power and Fission Rate Estimates in Bare Metal Experiments

Measurements of surface gamma fluence rates from a 7-cm-radius Oy sphere used in a 1967 1-hour criticality experiment led to an estimate of 1.9508×10^{-9} MW (1.9508 mW)[3] thermal power, corresponding with 2.18×10^{11} total fissions[4] (Rockwell 1967, p. 6).

H.4.2 Thermal Power and Fission Rate Estimates in Metal-Plus-Solution Experiments

CML staff made two estimates of total fissions in experiments involving Oy immersed in UNH. One 1976 estimate used the residual gamma photon flux from the Oy + UNH to estimate an upper bound of 1.8×10^{18} total fissions over the 12 years of CML operation from 1965 to 1976[5] and an average power of 25 W (Rockwell 1976, pp. 2–4). However, the senior scientist who performed the measurements and derived the estimate noted that the gamma background from a new unirradiated Oy part was the same as the gamma flux from irradiated parts used for the analysis. This means that measurements made 6 years after the last criticality experiment were unable to detect additional gammas from previous experiments because of radioactive decay of the MFAPs. The original estimate did not take into account the intrinsic gamma emissions from unirradiated EU and was, therefore, invalid[6] (Rockwell 1976, p. 2).

A second estimate of 3.44×10^{10} fissions in an average Oy + UNH experiment with a 20-minute run time was derived by estimating the number of neutrons produced (8.4×10^{10}) using the current generated by a neutron detector associated with an April 1977 experiment (Rockwell 1977a, p. 3). This fission rate corresponds with a thermal power level of 0.92 mW.

H.4.3 Thermal Power and Fission Rate Estimates in Solution-Only Experiments

The CML staff's most rigorous estimates of fission rates were derived for UNH (solution-only) fuel by measuring the rate of gamma decay in the fuel of Experiment 2-8-170. This experiment was configured as a 2 by 2 array of 8-in.-diameter tanks and was performed on May 4, 1977 (Rockwell 1977b). The gamma count rate of a 2-mL aliquot withdrawn from the fuel after a 70.5-minute run was determined at 2- to 3-minute intervals using a "well crystal" calibrated to ^{137}Cs photons. The total number of fissions in the reactor was calculated from the measured gamma emission rate (Rockwell 1977c, p. 7):

$$\varphi_\gamma \cong 1.9 \times 10^{-6}\, \tau^{-1.2} \text{ photons/s/fission} \qquad \text{(H-1)}$$

---

3 The conversion from MW to mW in the reference document incorporated a factor of 10 error, incorrectly concluding that the power level was 19.508 mW; the error is corrected here.

4 Using 1 W = 3.1×10^{10} fissions per second.

5 The estimate of total fissions is based on 27 days of continuous run time occurring in the middle of the 12-year period and corresponds with 7.7×10^{11} fissions per second.

6 The laboratory notebook entry reads: "All [gamma radiation is] due to fission 6 years ago. (No believable background available.) Note: Part no. 80 never used and yet had same background/kg as others. [Therefore] Feel fission too low to detect by this method."

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CRITICAL MASS LABORATORY (continued)

where:

φ_γ = reactor gamma photon emission rate at time τ
τ = days after shutdown

An initial evaluation[7] of the data (Rockwell 1977b) arrived at a power estimate of 38 mW during the last 10 minutes of the experiment (when most of the corresponding 7.1×10^{11} fissions occurred), giving an average power of 5.4 mW for the 70.5-minute duration of the experiment.

A subsequent June 7, 1977, estimate from Experiment 2-8-170 employed more precise values for the aliquot and reactor volumes (Rockwell 1977c, p. 4). The estimate gave 25 mW at the end of the experiment, corresponding with 4.73×10^{11} fissions (or 3.6 mW average power). These results were within 2% of those obtained by the method of source multiplication (LANL 1977; Rockwell undated), using the equation below:

$$f = s\mu\overline{t} \qquad \text{(H-2)}$$

where:

f = total number of fissions over time t
t = time
μ = average multiplication from the neutron detector response during the experiment
s = beginning neutron flux from the ^{252}Cf seed source

On June 3, 1977, before the more precise June 7 calculation was completed, CML staff communicated officially to the U.S. Energy Research and Development Administration (ERDA). They provided an estimate of 8.8×10^{11} fissions and an average power of 6.7 mW for a typical (70.5-minute) experiment (Schuske 1977). There is no indication why the values reported to ERDA were almost 25% higher than results of the May 4, 1977 calculation.

H.4.4 Conclusions about Reactor Power and Fission Rates in Experiments

Table H-1 provides a summary of the different CML criticality experiments for which thermal power and fission rate were estimated from measurements.

CML staff described Experiment 2-8-170 to ERDA as having a higher-than-normal power level (Schuske 1977, p. 2). A thermal power estimate of 3.6 mW averaged over 70.5 minutes (corresponding with 4.73×10^{11} total fissions) was based on careful evaluation of measurements made on Experiment 2-8-170. This is considered the most precise estimate available for this relatively high-power experiment. CML staff, however, reported to ERDA an average thermal power of 6.7 mW over 70.5 minutes (8.8×10^{11} total fissions) based on the same experiment. The value reported to ERDA is considered the most appropriate value to represent the typical CML criticality experiments for the purpose of estimating doses from MFAPs. This value is less than the earlier assumption that power averaged 10 mW for experiments lasting an hour (1.1×10^{12} total fissions). However, the value is more favorable to claimants because it exceeds the most careful estimate of

[7] Reactor power was evaluated using the integral of $\dot{\Phi}$ over the operating time of the reactor, giving the gamma energy emission rate as a function of reactor power and time after shutdown.

ATTACHMENT H
CRITICAL MASS LABORATORY (continued)

power for the experiment and because not all CML experiments achieved criticality (DOE undated, p. 3).

Table H-1. Average reactor power and fission rates derived from measurements at CML.[a]

Configuration	Duration	Average power (mW)	Total fissions	Fission rate (s^{-1})	Comment
Oy	60 min	1.9508	2.18E+11	6.06E+07	Based on exposure rate at the surface of an Oy sphere converted to γ photon flux.
Oy + UNH (a)	27 d	25,000	7.7E+11	3.30E+05	Estimate is invalid; measurement results were indistinguishable from background.
Oy + UNH (b)	20 min	0.92	3.44E+10	2.83E+07	From neutron detector current.
UNH (a)	70.5 min	5.4	7.1E+11	1.68E+08[b]	Initial calculation on May 4, 1977, for Experiment 2-8-170
UNH (b)	70.5 min	6.7	8.8E+11	2.08E+08[b]	Values reported to ERDA on June 3, 1977, based on results of Experiment 2-8-170.
UNH (c)	70.5 min	3.6	4.73E+11	1.12E+08	Recalculated values from Experiment 2-8-170 on June 7, 1977. Result is within 2% of that obtained from evaluation of source multiplication.

a. Sources: Oy from Rockwell (1967, p. 6); Oy + UNH (a) from Rockwell (1976); Oy + UNH (b) from Rockwell (1977a); UNH (a) from Rockwell (1977c, p. 7); UNH (b) from Schuske (1977); UNH (c) from Rockwell (1977c, p. 4).
b. These values are not reported in the original reference, but are calculated from total fissions and duration.

H.5 SURFACE CONTAMINATION

H.5.1 Radiological Survey Results

Section H.9 lists SRDB Ref IDs that contain radiological survey results for Building 886 and 875 for January 1981 to December 1990. These files include results of daily removable contamination surveys in the office and experimental areas, and on equipment, as well as weekly penetrating dose measurements in the nuclear material storage and experimental areas of Building 886.

H.5.2 Maximum Reported Values

Based on the data, Table H-2 shows the maximum reported values of removable alpha surface contamination for each year.

Based on the data, Table H-3 shows maximum gamma and neutron penetrating dose rates.

H.5.3 Evaluation of Removable Contamination Survey Results

Contamination surveys were conducted daily at control points (hallways, doorways to offices), and weekly in offices within the unrestricted-access portion of Building 886. Surveys were conducted weekly in the experimental areas. Results of periodic contamination surveys in Building 875 and its air filtration plenums are also documented. Survey results for the office area have been found through December 20, 1990 (Rockwell 1990a, p. 778), and for the experimental and material storage

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Table H-2. Maximum removable alpha contamination (dpm/100 cm^2).[a,b]

Year	Bldg. 886 offices[c]	Bldg. 886 experimental area[c]	Bldg. 886 equipment	Bldg. 875 outside plenum	Bldg. 875 inside plenum
1981	30	366	156	12	11,196
1982	18	174 (3,930)[e]	NS	12	14,346
1983[f]	<20	576	NS	NS	NS
1984	<20	54,000	NS	<20	15,660
1985	<20	174	NS	<20	4,788
1986	18	186 (2,784)	3,954	12	5,394
1987	18 (24)	564	NS	NS	NS
1988	12	1,683	NS	12	15,240
1989[g]	18	18,707	NS	NS	NS
1990[g]	18	16,212	NS	6	NS

a. Source: See Section H.9.
b. NS = not surveyed.
c. "Offices" includes control points (corridors and common areas) typically surveyed daily and individual offices surveyed weekly.
d. The experimental area (Rooms 101, 102, and 103) were surveyed weekly in 1981, but the frequency gradually diminished, apparently in response to decreased activities in these areas. Surveys were made approximately monthly by 1990.
e. Values in parentheses are singular outliers and not representative.
f. Data in 1983 are available only for November and December.
g. Elevated values of this order were observed in weekly surveys of the Room 101 walk-in hood from December 1989 through April 1990, as discussed in the text.

Table H-3. Maximum dose rates in CML experimental and material storage areas.[a,b]

Year	Gamma[c] (mR/hr) UNH Tank Room	Gamma[c] (mR/hr) Solid Fuel Storage	Gamma[c] (mR/hr) Experiment Room	Neutron[d] (mrem/hr) UNH Tank Room	Neutron[d] (mrem/hr) Solid Fuel Storage	Neutron[d] (mrem/hr) Experiment Room
1981	2.0	2.9	NS	0.3	12.9	NS
1982	2.0	2.5	3.7	0.2	14.7	5.4
1983	NS	NS	NS	NS	NS	NS
1984	5.0	3.2	NS	0.2	0.6	NS
1985	2.0	2.8	NS	0.1	0.5	NS
1986	2.0	6.0	0.4	1.0	2.6	0.2
1987	3.0	2.0	3.0	1.7	2.0	3.4
1988	2.4	1.5	1.8	2.0	0.8	0.6
1989[e]	2.8	4.2 (250)[f]	0.0	1.0	1.8 (6.0)	0.6
1990	2.53	3.0	NS	0.3	0.5	NS

a. Source: See Section H.9.
b. NS = not surveyed.
c. Gamma dose rate surveys were made at least weekly during the entire 10-year period.
d. Neutron dose rate surveys were made weekly, along with gamma surveys, until 1990 when the frequency was reduced to monthly.
e. Atypically elevated dose rate readings for July 6, 1989, are associated with a survey location within an area marked off inside the solid storage area.
f. Values in parentheses are singular outliers and not representative.

areas through April 1990 (Rockwell 1990a, p. 330). No average values were found, except in 1989 documents, which contained a few graphs showing average contamination values in different areas for limited periods (Rockwell 1989a, pp. 3–6). These graphs are shown in Figures H-1 through H-4.

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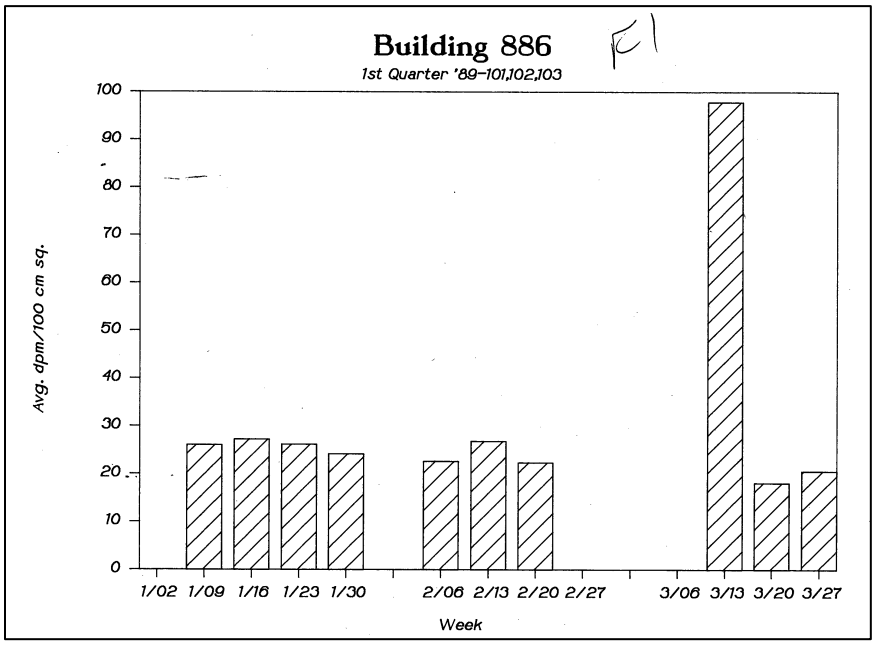

Figure H-1. Weekly averages of removable alpha contamination in the Building 886 experimental and material storage areas, first quarter 1989 (Rockwell 1989a, p. 3).

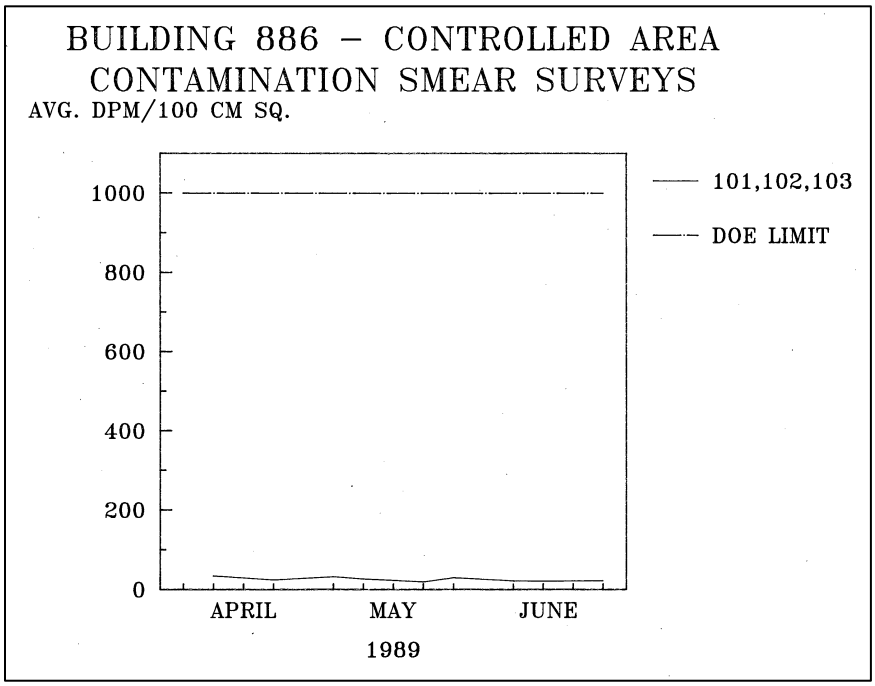

Figure H-2: Average removable alpha contamination in the Building 886 experimental and material storage areas, second quarter 1989 (Rockwell 1989a, p. 5).

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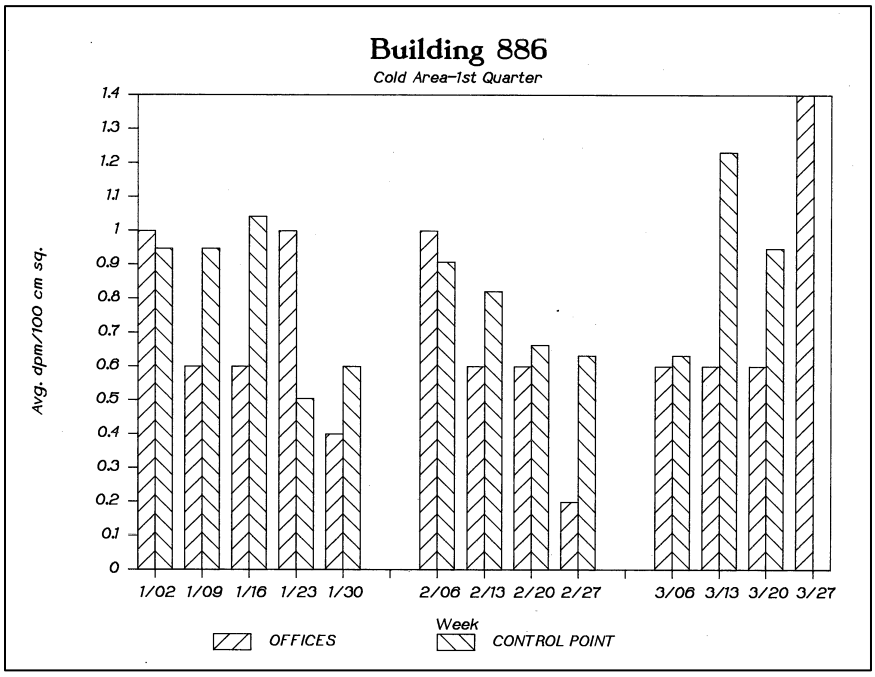

Figure H-3. Weekly averages of removable alpha contamination in the Building 886 offices and control points (hallways and conference rooms), first quarter 1989 (Rockwell 1989a, p. 6).

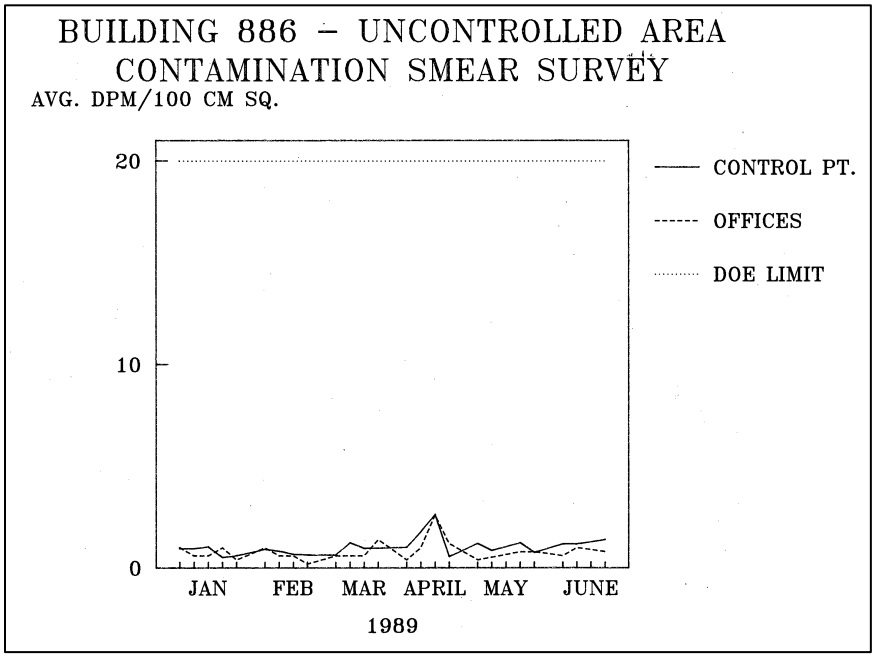

Figure H-4. Average removable alpha contamination in the Building 886 office area, January to June 1989 (Rockwell 1989a, p. 4).

Table H-2 above, showing maximum values for the removable alpha survey results in both the office and experimental areas, demonstrates that radiological containment of the contaminated areas was quite effective. Excursions of removable contamination above the 20 dpm/100 cm^2 DOE limit in “cold”

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(office) areas is seen to occur only a few times (and only at low levels) in 1981 and once in 1984. Survey results in offices and hallways were typically well below 20 dpm/100 cm^2, particularly toward the end of the period from 1981 to 1990 (e.g., see Figures H-3 and H-4). The solid line in Figure H-4 represents removable alpha contamination; the heavier dashed line represents the DOE limit for uranium.

Review of the survey documents indicates that excursions of removable alpha contamination above the Contamination Area limit in the experimental area (Room 101) and the material storage area (Rooms 102 and 103) were usually reduced below 1,000 dpm/100 cm^2 within a few days, as noted below for entries in Table H-2.

- The survey sheet showing 3,930 dpm/100 cm^2 for removable contamination in Room 103 (the Mixing Room where uranium solution was housed) on September 10, 1982, for instance, has notation showing that the area was decontaminated to 12 dpm/100 cm^2 on the same day (Rockwell 1982a, p. 23).

- A survey result of 54,000 dpm/100 cm^2 on February 27, 1984 (the highest found in any of the documents), was in the walk-in hood of Room 101 (Rockwell 1984a, p. 13). A second survey sheet with the same date shows <20 dpm/100 cm^2 for all survey locations in Rooms 101, 102 and 103, and bears the handwritten note, "Retake of survey after [Redacted] and [Redacted] cleaned all of Rm 101."

- The high value of 2,784 dpm/100 cm^2 in 1986 is reported on a survey sheet dated June 17 along with the handwritten notation, "Table Top 3954 d/m/100 cm^2; Rest of Table 25.8 d/m/100 cm^2; deconned 6/18/86" (Rockwell 1986a, p. 8). The survey sheet for June 18, 1986 indicates that all results in the controlled area were ≤24 dpm/100 cm^2 (Rockwell 1986a, p. 9).

- The 1988 high result of 1,683 dpm/100 cm^2 was found just outside the walk-in hood of Room 101 on October 7, 1988 (Rockwell 1988a, p. 37). A similar elevated result (1,422 dpm/100 cm^2) was found inside the hood. Contamination at the location outside the hood had reduced to 60 dpm/100 cm^2 a week later on October 14 (Rockwell 1988a, p. 38), but the elevated value inside the walk-in hood persisted. It had reduced to 126 dpm/100 cm^2 in the October 21 survey report (Rockwell 1988a, p. 39).

Prompt decontamination practices were evident during most of the period from 1981 to 1990; however, elevated contamination levels in the Room 101 walk-in hood were recorded in December 1989 and persisted through April 1990. The persistent contamination is reflected in the maximum recorded values of 1989 and 1990 in Table H-2 and is detailed in the weekly contamination survey results in Table H-4. The FBI raid of RFP in June 1989 resulted in the curtailment of plutonium operations. Afterward, the plant's resources were redirected toward remediation of the issues resulting from the raid. According to *A Technically Useful History of the Critical Mass Laboratory at Rocky Flats*, "Manpower was not available to decontaminate Room 101 in the late 1980s because Maintenance personnel had all been dedicated to solving the plant's larger problems" (Rothe 2005, p. 395). Handwritten notes on the survey sheets sometime indicate that full-face respirator protection was required in Room 101 when contamination levels exceeded the DOE limit (e.g., Rockwell 1989a, p. 109). Figure H-5 shows the floor plan for Room 101 where criticality experiments were performed; the walk-in hood is the enclosure left of center.

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Table H-4. Room 101 walk-in hood contamination results (dpm/100 cm^2), November 1989 to April 1990.[a]

Date	Location 20	Location 21	Location 22	Source	Page
11/21/1989	27	642	NS	Rockwell 1989a	92
11/30/1989	239	NS	NS	Rockwell 1989a	116
12/07/1989	897	12,579	18,707	Rockwell 1989a	160
12/14/1989	9	100	90	Rockwell 1989a	183
12/21/1989	72	51	5,235	Rockwell 1989a	196
01/18/1990	168	693	6,990	Rockwell 1990a	60
01/25/1990	84	672	6,963	Rockwell 1990a	81
02/01/1990	93	1,230	12,219	Rockwell 1990a	99
02/08/1990	105	1,290	13,452	Rockwell 1990a	117
02/15/1990	255	15,099	10,839	Rockwell 1990a	135
02/22/1990	717	12,369	9,978	Rockwell 1990a	156
03/01/1990	639	14,163	6,933	Rockwell 1990a	171
03/08/1990	616	14,193	6,987	Rockwell 1990a	183
03/15/1990	468	12,423	7,923	Rockwell 1990a	207
03/22/1990	1,035	16,212	5,403	Rockwell 1990a	234
03/29/1990	1,893	15,708	7,842	Rockwell 1990a	252
04/05/1990	2,748	12,951	8,967	Rockwell 1990a	273
04/19/1990	2,556	14,967	13,683	Rockwell 1990a	291
04/12/1990	1,572	13,593	10,353	Rockwell 1990a	300
04/26/1990	2,037	14,463	11,973	Rockwell 1990a	330

a. Locations 20, 21, and 22 correspond with the entry portal, Solution Base (S), and Vertical Split Table (V), respectively, as shown on the floor plan in Figure H-5.

Air flow through Room 101 came from outside through the walk-in hood to the exhaust plenum in Building 875, then through high-efficiency particulate air filters before being exhausted to the outside air. The walk-in hood was, therefore, at a lower pressure than Room 101, drawing resuspended contamination to the exhaust plenum rather than the rest of Room 101. This engineered feature reduced exposure to Room 101 personnel and, along with workforce priorities redirected as a result of the FBI raid, probably explains why the hood was not decontaminated in late 1989 and 1990. The last CML criticality experiment concluded in October 1987 (Rothe 2005, p. 393); no routine work was performed in Room 101 after this date, further reducing the potential for personnel exposure as the result of contamination in the walk-in hood in 1989 and 1990.

H.5.4 Discovery of Sealed Strontium-90 Sources

The file of survey results for June 1982 contains a copy of a note detailing radiological characteristics of three sealed ^{90}Sr sources (Rockwell 1982b, p. 3). The note is reproduced as Figure H-6 below. It is the only documentation in hand indicating the presence of ^{90}Sr at CML. There is no information about why the sources were at CML, how long they might have been there, or their disposition. Integrity of the source encapsulations was confirmed by removable contamination surveys of the sources and the cabinet in which they were found.

H.5.5 Conclusions About Surface Contamination

Surveys for removable alpha contamination at CML were conducted regularly. Excursions above the applicable DOE limits (20 dpm/100 cm^2 in uncontrolled office areas and 1,000 dpm/100 cm^2 in controlled-access experimental areas) were confined to discrete areas and were quickly decontaminated below the limits. Values above the limit in uncontrolled areas were only rarely

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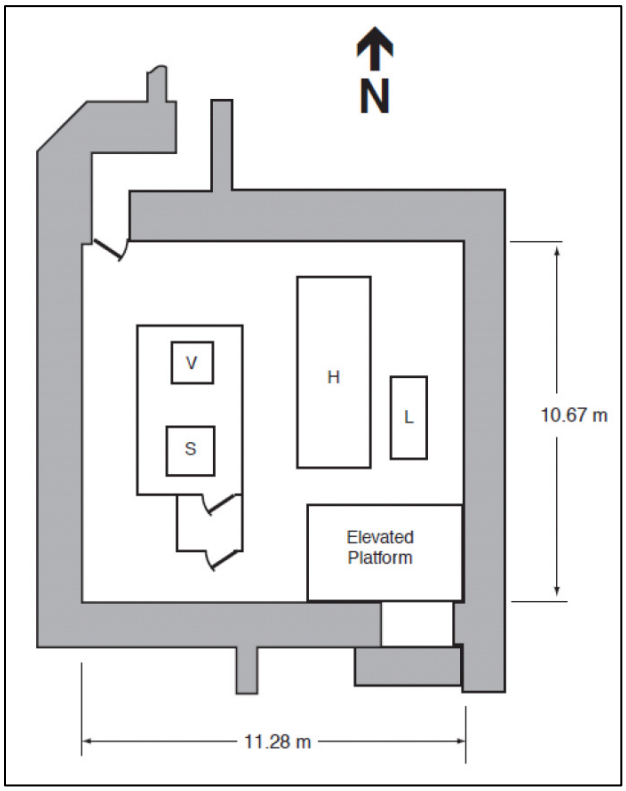

Figure H-5. Floor plan of the assembly room (Room 101) showing the walk-in hood containing the vertical split table (V) and solution base (S) experimental locations (Rothe 2005, p. 120).

observed. The largest measured value for removable contamination was 54,000 dpm/100 cm^2, found at sampling location 21 in the walk-in hood (Rockwell 1984a, pp. 13–14). This amount of contamination, if spread uniformly over the entire surface area[8] of the controlled area, would be 1,800 dpm/100 cm^2; it was decontaminated on the same day it was found. All other values for removable contamination correspond to <1,000 dpm/100 cm^2, if distributed uniformly. All were quickly decontaminated, except for that in the walk-in hood from November 1989 to April 1990, which was contained and ventilated during a time of little or no personnel activity in Room 101.

In light of the above information, the assumption that average removable contamination available for resuspension in the experimental and material storage areas (Rooms 101, 102, and 103) was equal to or less than the Contamination Area limit of 1,000 dpm/100 cm^2 is favorable to claimants.

[8] Dimensions of the walk-in hood (Rothe 2005, p. 129) in Room 101 were 3.0 m by 4.9 m. Sampling location 21 was in a part of the hood making up about half this surface area, 7.35 m^2. Estimated floor areas for the individual rooms are 120 m^2 for Room 101 (Rothe 2005, p. 120); 40 m^2 for Room 103 (Rothe 2005, p. 167); and 60 m^2 for Rooms 102 and 108 (connecting hallway) combined, assuming this area to be about half that of Room 101 (Rothe 2005, p. 110). The total estimated floor area is the sum of these values, 220 m^2. Distributing the contamination at sampling location 21 over the entire surface area would result in an average contamination level reduced by the ratio 7.35/220.

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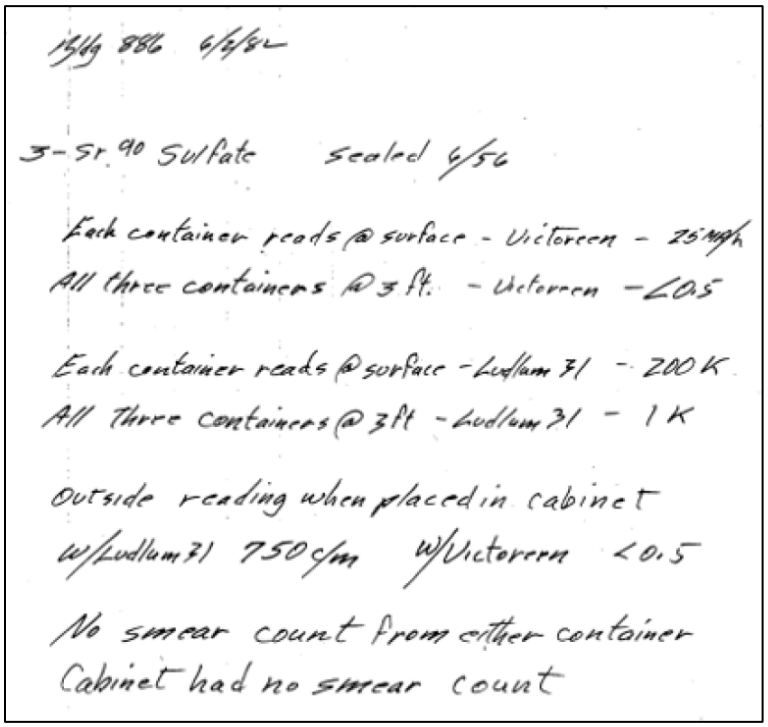

Figure H-6. Memorandum on beta/gamma penetrating and removable surface contamination measurements made upon discovering three sealed ^{90}Sr sources in the Building 886 material storage area (Rockwell 1982b, p. 3).

H.6 WORKPLACE AIR MONITORING

Bounds on internal dose from MFAPs were previously based on airborne concentrations calculated by applying a resuspension factor to surface contamination limits for the facility. This approach was based on interview comments from a former [redacted], who said that no routine air monitoring was performed at the CML before 1990, when this individual was assigned responsibility for Building 886 (ORAUT 2015a, pp. 4, 7). Formal plantwide procedures describe a particulate air monitoring program during the period 1980 to 1989 for alpha emissions from uranium, plutonium, and americium at sampling locations selected by process knowledge or professional judgment. Additional documents indicate that these procedures appear to have been followed and that routine alpha air monitoring was performed at the CML during the period 1980 to 1989.

H.6.1 Workplace Air Monitoring Requirements

Formal procedures describe monitoring requirements and practices for routinely evaluating concentrations of alpha-emitting particulates (uranium, plutonium, and americium) in the RFP workplace air over the period 1979 through 1990 (RFP 1976–1996, pp. 19–136). The earliest known procedure (issued November 1979) and its subsequent revisions through December 1990 are listed below (RFP 1976–1996):

- *Routine Air Sampling*, HS-RM-4.1, November 13, 1979 (pp. 19–29);
- *Routine Air Sampling*, RMPM 4.1, August 2, 1982 (pp. 30–43);

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- *Routine Air Sampling*, RMPM 4.1, January 13, 1989, replaced August 2, 1982 (pp. 65–81);
- *Routine Air Sampling*, RMPM 4.1, June 1989, replaced January 13, 1982 (pp. 44–64);
- *Routine Air Sampling*, ROI 4.1, December 18, 1989, replaced June 15, 1989 (pp. 102–118); and
- *Routine Air Sampling*, ROI 4.1, Rev. 3, December 20, 1990, replaces December 19, 1989 (pp. 119–136).

Review of the procedures shows consistent requirements for monthly calibration of gross alpha CAMs and selective alpha air monitors (SAAMs). CAM filters were collected and sent for analysis each weekday, except holidays, until at least January 1989, when RMPM 4.1 specified that the exchange frequency was to be determined by Operational Health Physics personnel (RFP 1976–1996, p. 67). Subsequent revisions assigned the responsibility for establishing sampling frequencies to Building 123 Count Room personnel (p. 49), Operational Health Physics (p. 104), or to Radiological Engineering personnel (p. 121). When collected, filters were monitored with a handheld alpha survey instrument. Filters with excessive alpha activity were held for specified periods before analysis to allow decay of the short-lived radon and thoron progeny and a time-dependent Koval factor was applied to the analytical results to account for undecayed progeny (pp. 22–23, 36–37, 52, 72). Final results were reported as percentages of the RCG airborne limit for the material in question, or 70 dpm/m^3 for uranium[9] (pp. 25, 39, 73). In June 1989, the airborne limit for uranium (in any form) was changed to 44 dpm/m^3, corresponding to the DOE Order 5480.11 (DOE 1988) derived air concentration (RFP 1976–1996, pp. 54, 104, 121).

H.6.2 Air Sampling Locations

Air particulate samplers were in Buildings 886 and 875, as marked on monthly air head calibration sheet maps (Rockwell 1987a to 1989c); Figure H-7 shows an example. The locations appear to be unchanged during the period 1980 to 1989. Building 875 housed the air exhaust plenums from the Building 886 experimental (Room 101) and material storage (Rooms 102 and 103) areas. The air head samplers were numbered RR-1 through RR-11 and fixed in the following locations:

- Building 886:
  - RR-1 through RR-4, Room 103 (material storage);
  - RR-5, Room 102 (material storage); and
  - RR-6 and RR-7, Room 101 (experimental area).
- Building 875:
  - RR-8 and RR-9, outside the plenum;
  - RR-10, inside the first plenum; and
  - RR-11, inside the tunnel from Building 886.

[9] The RCG used by RFP in its air-sampling program corresponds with the maximum allowable (air) concentration used by AEC contractors in referring to the AEC's "preferred level" of 50 µg/m^3 for all uranium compounds on the basis of chemical toxicity. The level was also stated as 70 dpm/m^3 for natural uranium (ORAUT 2006, p. 8).

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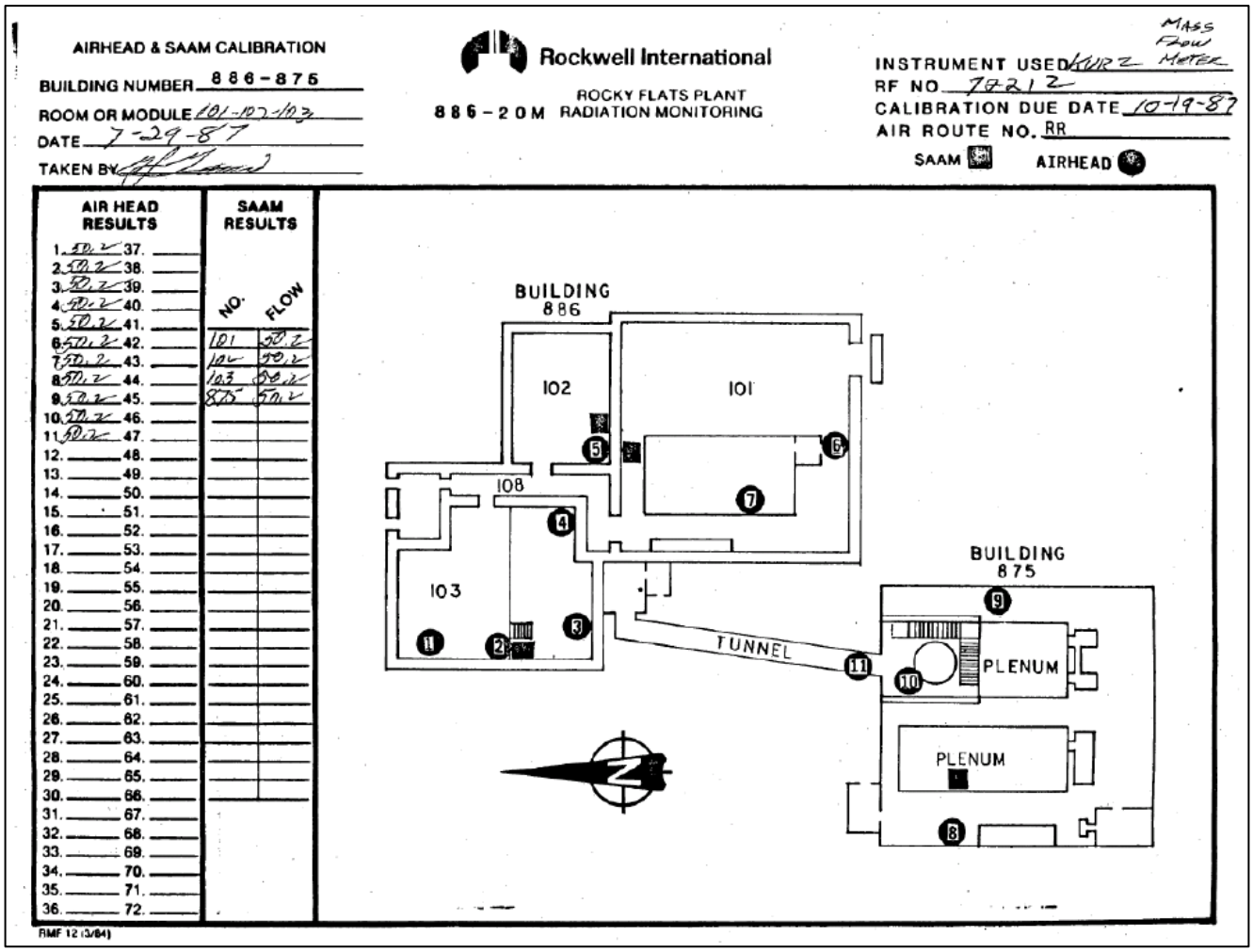

Figure H-7. Air sampler locations in the Building 886 cluster (Rockwell 1987–1988, p. 6).

SAAM samplers were also in Rooms 101 to 103 of Building 886 and in the second plenum of Building 775. The detectors on these samplers were equipped with high-voltage discriminators that registered only alpha energies above a certain threshold, which provided a means to discriminate against lower-energy alpha emitters that would otherwise give false alarms. The filter material from the SAAMs was normally discarded without analysis unless the instrument had alarmed or the filter failed a handheld radiation monitor check (RFP 1976–1996, pp. 26, 35, 51, 70, 88, 105–106, 123).

Section H.9 includes files that contain air sampling results from 1981 to 1989 for samples labeled "875-A", "875-B" and "875-C," but their locations in Building 875 have not been identified.

CML personnel did not routinely access Building 875, and consideration of air monitoring results for this building is limited to those in the "RR" series, with known locations and direct correlation with daily samples from Building 886.

H.6.3 Air Sampling Results

Air sample results in 41 of 120 months have been found for the period of 1980 to 1989. Results from samplers RR-1 through RR-7 (Building 886) and RR-8 through RR-11 (Building 875) are available for

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the period December 1980 to June 1989, although all but two of the results fall in the period from May 1983 to November 1988. The data, summarized in Table H-5, indicate that procedural requirements for daily air monitoring appear to have been met at the CML. The number of analytical results per month for a particular sampler location is expected to be between 19 and 22, depending on the number of workdays. In the data, daily sample collection is demonstrated with certainty (results available for 19 or more days) in 15 of the 41 months (37% of the time for which records are available, but only 1% of the period from 1980 to 1989). Sample collection on at least an alternate-day average (results available for 10 to 18 days) is demonstrated with certainty during an additional 10 months. The availability of air sampling data does not seem to correlate with known spills or criticality experiments. Only one sampling day's results were found for operations before 1983, but 207 sampling days correspond with three operational periods after 1983. Four contamination incidents occurred from 1980 to 1989, but air monitoring results were only found for the period in which two incidents occurred closely together in 1987. Experiment periods, contamination incident dates, and associated records of airborne contamination results are summarized in Table H-6.

Table H-5. Air sample results summary for the CML Building 886 nuclear materials storage and experimental areas.[a,b]

Year	Month	Source	Sample days[c]	No. of days 10%-99% RCG[d]	No. of days ≥100% RCG[e]
1980	Dec	Rockwell 1980	1	0	0
1983	May	Rockwell 1983a	2	0	0
1983	Sep	Rockwell 1983b	1	0	0
1983	Oct	Rockwell 1983c	17	2	0
1984	Jul	Rockwell 1984b	2	0	0
1984	Aug	Rockwell 1984c	22	1	1
1984	Sep	Rockwell 1984d	19	2	0
1984	Oct	Rockwell 1984e	20	6	0
1984	Nov	Rockwell 1984f	22	8	3[d]
1984	Dec	Rockwell 1984g	15	5	0
1985	Jan	Rockwell 1985a	19	4	0
1985	Feb	Rockwell 1985b	19	8	2
1985	Mar	Rockwell 1985c	22	1	0
1985	Apr	Rockwell 1985d	21	0	0
1985	May	Rockwell 1985e	22	3	0
1985	Jun	Rockwell 1985f	22	0	0
1985	Jul	Rockwell 1985g	23	2	0
1985	Aug	Rockwell 1985h	20	2	0
1985	Sep	Rockwell 1985i	15	0	0
1985	Oct	Rockwell 1985j	9	0	0
1985	Nov	Rockwell 1985k	19	1	0
1985	Dec	Rockwell 1985l	15	0	0
1986	Jan	Rockwell 1986b	19	0	0
1986	Feb	Rockwell 1986c	1	0	0
1986	Jun	Rockwell 1986d	4	0	0
1986	Jul	Rockwell 1986e	14	0	0
1986	Nov	Rockwell 1986f	3	0	0
1986	Dec	Rockwell 1986g	9	0	0
1987	Jan	Rockwell 1987a	15	1	0
1987	Feb	Rockwell 1987b	22	0	0
1987	Mar	Rockwell 1987c	8	0	0
1987	May	Rockwell 1987d	6	0	0
1987	Jul	Rockwell 1987e	3	0	0

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Year	Month	Source	Sample days[c]	No. of days 10%-99% RCG[d]	No. of days ≥100% RCG[e]
1987	Aug	Rockwell 1987f	13	0	0
1987	Sep	Rockwell 1987g	7	0	0
1987	Dec	Rockwell 1987h	11	0	0
1988	Jan	Rockwell 1988c	18	0	0
1988	Feb	Rockwell 1988d	15	0	0
1988	Apr	Rockwell 1988e	5	0	0
1988	Nov	Rockwell 1988f, 1988g	5	0	0
1989	Jun	Rockwell 1989c	1	0	0
Totals	N/A	N/A	**526**	**46**	**6**

a. N/A = not applicable.
b. Documents include results from only portions of years 1980 and 1983 to 1989, as shown in the table.
c. The number of days in the specified month for which air monitoring data indicate that samples were collected. Procedures required that sample filters be changed and analyzed daily, except over weekends and holidays.
d. The number of days for which the specified fraction of the uranium RCG of 70 dpm/m^3 alpha was met or exceeded.
e. Notations on the air results indicate that respirators were worn.

Table H-6. Experiment periods, contamination incident dates, and associated records of airborne contamination results.

Experimental campaigns[a]	Contamination incident dates[b]	Airborne monitoring days captured
None	11/25/1980	0
02/1978–09/1981	None	1
Summer 1982–12/20/1982	None	0
05/1983–09/1984	None	64
None	07/07/1984–07/20/1984	0
07/1985–08/1986	None	139
None	02/14/1987 and 03/13/1987	30
04/1986–10/1987	None	104

a. Source: Rothe (2005, pp. 389–393).
b. Source: Rothe (2005, pp. 486–487, 498, 500–501).

Building 886/875 results appear in the same reports as those for Building 865, which housed the Metal Research and Development Laboratory[10] and which processed nonplutonium metals including DU (DOE 2011b, p. 2). Reports were reviewed and initialed, and instructions on the report sheet specify that a copy was to be sent "TO RADIATION MONT 881" (to Radiation Monitoring, Building 881).

Individual sample results are occasionally lined out in the reports, with or without explanation. There are multiple instances when results are lined out with the notation "W/C" or "Wrong Color." The meaning of this notation is not clear, and these samples are not included in evaluating the results.

A total of 526 days of results were evaluated against the RCG of 70 dpm/m^3. One or more results from the seven samplers in Building 886 exceeded the RCG on 6 days; results between 10% and 100% of the RCG were recorded on 46 additional days. All results were below 10% of the RCG for the remaining 473 days. Respirators were worn on 3 of the 6 days in which the RCG was exceeded (Rockwell 1984b, pp. 33, 36, 52). Results on the other 3 days in which the RCG was exceeded were: 522.02% RCG (Rockwell 1984c, p. 5), 111.72% RCG (Rockwell 1985c, p. 37), and 117.76% RCG (Rockwell 1985c, p. 41).

[10] Each day's results for Building 886 are labeled RR-1 through RR-7; those for Building 875 are labeled RR-8 through RR-11; Building 865 results begin with a UU- designator.

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Results from Building 875, which housed the effluent air exhaust plenums from the Building 886 experiment room, exceeded the RCG on only one occasion[11] during the same 526 days; results between 10% and 100% of the RCG were recorded on 2 days.

H.6.4 Conclusions About Workplace Air Monitoring

A robust and well-defined workplace air monitoring program for alpha-emitting radioisotopes was required by RFP procedures on a continuing basis during the period from 1980 to 1989. In all air monitoring records, air particulate samples from the CML, Building 886, and its air exhaust filtration plenums in Building 875 were routinely analyzed and reported along with those from the Metal Research and Development Laboratory, Building 865. Sampling results were evaluated for uranium alpha emissions and reported as a percentage of the RCG airborne limit for uranium, 70 dpm/m^3. The analytical reports were reviewed and initialed, and were to be sent to Radiation Monitoring in Building 881. Excursions in excess of the RCG were uncommon, occurring in Building 886 on 6 of 526 days for which monitoring results were found and in Building 875 only once.

A bounding value for activity concentrations in breathing air can be calculated as the weighted average $\overline{C}$ of air results using recorded values for three results in excess of the RCG with no indication that respirators were worn, and by making the assumptions that results were 70 dpm/m^3 for 46 results between 10% and 100% of RCG and 7 dpm/m^3 for the remaining 477 recorded samples (which are favorable to the claimant), as shown in the equation below:

$$\overline{C} = \frac{\left[5.22 + 1.12 + 1.18 + (46)(1.0) + (477)(0.1)\right]}{526\,\text{d}}\text{RCG-d} \qquad \text{(H-3)}$$

$$= 13.5\ \text{dpm/m}^3$$

Results for the 526 days of monitoring data found for the period from 1980 to 1989 are assumed to be representative of data that have not been found for that period for the following reasons:

- The same plant health physics procedures requiring air monitoring were in effect continuously (with revisions) for the entire period,
- The assumption is made that daily samples were collected and analyzed in accordance with plant procedures over the entire period, and
- The results are probably similar because CML operations were similar over the period.

These data were available for routine review by health physics personnel, who also had access to information about operations for making personnel monitoring decisions. It is therefore unlikely that an unrecorded intake of alpha-contaminated airborne particulates occurred during this period because of a lack of relevant air monitoring data.

H.7 ASSESSMENT OF UNMONITORED RADIATION DOSE

Radiation dose from intake of MFAPs at CML could have occurred during cleanup of numerous fuel spills, predominantly from enriched UNH solution, or from inhalation of dried, resuspended

[11] The sample designation was RR-12 for the one result exceeding the RCG, without explanation of the location or purpose for the sample. A location in Building 875 is assumed.

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contamination deposited on surfaces as the result of these spills (Rothe 2005, pp. 447–449, 452–462, 464–473, 486–487, 498, 500–501). RFP workers (including those assigned to the CML) with the potential for receiving intakes of plutonium, americium, or uranium were monitored by periodic urinalysis and body counts (NIOSH 2006, p. 30). However, there is no indication that confirmatory bioassays were performed for workers involved in cleanup of any of the accidental UNH spills. MFAPs, which decay primarily by beta/gamma emission, are not likely in any case to have been detected by bioassay intended to detect alpha particles from uranium or transuranic radionuclides.

Maximum MFAPs internal doses to CML workers were estimated by modeling a representative UNH experiment and calculating the MFAPs inventory based on the historical record of CML experiments with UNH, and on the average thermal power and duration of CML UNH criticality experiments. Intakes of resuspended UNH contamination with the same MFAPs-to-uranium atom ratio as the fuel were estimated from the weighted average of air monitoring results in the experimental and materials storage areas of the CML. Doses were calculated by applying ICRP Publication 68 (ICRP 1995) DCFs for three solubility categories of dosimetrically significant radionuclides, using the method described in ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2015b).

H.7.1 Calculation of Fission and Activation Product Content of Uranyl Nitrate Solution

A series of experiments was performed at the CML during the mid-1970s to determine the critical height of UNH in suspended cylindrical tanks. Experiments were performed using different uranium concentrations, different tanks, and with and without neutron reflectors. The same uranium enrichment was used in all experiments; only the concentration was varied.

One of the unreflected suspended tank experiments from the mid-1970s was chosen to represent UNH experiments performed over the CML's history. There were 10 unreflected suspended tank experiments in all. Two of them used a stainless-steel tank and the rest used an aluminum tank.

One of the experiments that used the stainless-steel tank was selected so that the calculated MFAP content would include iron activation products. The tank had an inside diameter of 27.92 cm and an inside height of 41.6 cm. For the selected experiment, the tank was filled with UNH with a uranium enrichment of 93.172 weight percent ^{235}U at a concentration of 145.68 g/L. The other unreflected stainless-steel tank experiment used UNH with the same enrichment, but at a concentration of 346.73 g/L. The lower concentration was selected because it represents the middle of the range of concentrations used across the 10 unreflected tank experiments. The critical solution height for the selected experiment was found to be 31.20 cm. The selected experiment is documented as Case Number 1 in "Minimally Reflected Cylinders of Highly Enriched Solutions of Uranyl Nitrate," HEU-SOL-THERM-001, from the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (Palmer 2004).

The MFAP composition for the selected experiment was calculated using SCALE. SCALE is a modular system of computer codes for nuclear- and radiological engineering-related analyses from ORNL. SCALE's TRITON module was used to develop a case-specific cross-section library for the selected suspended tank experiment. The library was then used by the ORIGEN-S code to determine the time-dependent MFAP content of the UNH and the stainless-steel tank over the CML's history.

The TRITON module performs neutron transport and depletion calculations for irradiated nuclear fuels. Neutron transport is performed using either discrete ordinates methods (via the NEWT code) or via Monte Carlo calculations (using the KENO-V or KENO-VI codes). TRITON's capabilities include

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creating case-specific cross-section library files that can be used by other elements of the SCALE code system, notably the ORIGEN-S code. Among the numerous capabilities of ORIGEN-S is calculation of the inventory of activation products, actinides, and fission products in a composition as a function of time and burnup (as applicable).

TRITON was used to model the stainless-steel tank and UNH solution geometry of the selected suspended tank experiment as a system of cylinders representing the tank bottom, tank walls, and the UNH volume. The tank and solution heights were truncated to the critical solution height of 31.20 cm. Neutron transport was performed via the KENO-VI Monte Carlo code.

The case-specific cross-section library created by TRITON was subsequently used in an ORIGEN-S calculation to determine the MFAP content of the solution and the tank volume at the end of the tenth campaign and thereafter. This reflects a modeling assumption that the same tank and solution were used for all 778 experiments involving HEU solutions at the CML over its operating history. The composition used for the ORIGEN-S calculations was a homogenized mixture of the HEU solution, the 304 stainless-steel tank walls and bottom, and the associated impurities. The composition was irradiated and decayed in the ORIGEN-S case using the history in Table H-6 above. The first campaign was represented by a 38-hour irradiation followed by 32 days of decay, and so on, through the final 19-hour irradiation representing the tenth campaign.

Each experiment was assumed to have lasted for 70.5 minutes and to have produced an average thermal power[12] of 6.7 mW (the average power and duration reported to ERDA in 1977) (Schuske 1977, p. 6).[13]

The ORIGEN-S calculation produced time-dependent MFAP inventories, with radioactive decay between each of the 10 campaigns, for the HEU solution defined in the selected benchmark case, and using the timeline for all uranium solution experiments conducted over the CML's history. Radioactive decay corrections were also applied at the end of the final experiment, using a decay period of 180 days from those specified in ORAUT-OTIB-0054 (ORAUT 2015b).

H.7.2 Calculation of Inhalation Intakes and Committed Organ Doses

Solution spills resulting in surface (floor) contamination occurred throughout CML operations, as shown in Table H-7. Most spills occurred in the late 1960s, but there were several spills during the 1980s. Organ dose calculations were performed after applying a decay interval of only 180 days to the ending MFAP inventory for the HEU solution. This is favorable to the claimant given that the calculation of the MFAP inventory represents the entire operating history of the facility (i.e., includes the period after the last major spill).

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[12] The power in this context is an average value representing a given number of fissions over a given interval of time because nuclear criticality experiments are not steady state and are not typically considered in terms of a power level. An average power was used for modeling purposes to account for change in the composition of the fissile solution (i.e., for depletion effects) and to compute the ingrowth of fission products as the solution was used. In reality, most of the fissions would have occurred near the end of a given experiment when the system was alternately placed in slightly subcritical and slightly supercritical states.

[13] The 6.7-mW value is used, rather than the more precise 3.6 mW for 70.5 minutes estimated by CML staff after the communication with ERDA. There is no indication why 6.7 mW was reported instead of a lower value of 5.4 mW initially calculated by CML staff before refining their estimate. The higher value was adopted as more favorable for claimants.

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Table H-7. HEU solution spills at CML.[a]

Date	Volume (L)	Uranium mass (kg)	Contaminated area	Page(s)
07/02/1965	0.5	0.225	22 m^2 (floor)	447
07/14/1965	Unknown	Unknown	10 m^2 (floor)	448–449
07/22/1965	Unknown	Unknown	Small amount (floor)	449
11/30/1967	Unknown	9	(Inside large duct, filter housing, vent line)[b]	452–462
02/16/1968	Unknown	1.14[c]	(Floor and cable trenches)	464-467
05/11/1968	0.06	Unknown	(Worker's knee)	467
05/09/1969	150.1	16.1	20 m^2(mixing room floor)[d]	467–473
11/25/1980	7	2.66	(Assembly room hood)	486–487
07/07–20/1984	Unknown	Unknown	(Walk-in hood)[e]	498
02/14/1987	Unknown	Unknown	(Personnel, facility and fixtures)[f]	500–501
03/13/1987	Unknown	Unknown	(Personnel, facility and fixtures)[f]	501

a. Source: Rothe (2005).
b. Although contamination was confined to ducts and a filter housing, cleanup of this incident resulted in a blowback of dried salts, resulting in facial contamination of a staff member.
c. Two conflicting accounts refer to this value as either the solution mass or uranium mass.
d. Standing HEU solution covering the floor was cleaned up by a staff member using a critically safe vacuum and wearing plastic booties and a half-face respirator.
e. Potentially contaminated workers repairing a [redacted] were required to evacuate when a criticality alarm was triggered by an electrician.
f. Two essentially identical events resulted in personnel, fixture, and facility contamination by resuspended HEU UNH salts accidentally knocked from the surface of a large reactivity shim.

ORAUT-OTIB-0054 (ORAUT 2015b) describes a method to reduce the large number of MFAP isotopes in an ORIGEN-S result to a set of 36 dosimetrically significant nuclides. The same 36 nuclides were considered in the inhalation intake and committed organ dose calculations for the CML. Intakes were computed using the 180-day activity values for the HEU solution shown in Table H-8, corrected for the average airborne concentration level, and assuming an intake period of 4,000 hours (2 working years) at a breathing rate of 1.2 m^3/hr. The 2-year intake period was selected to be consistent with ORAUT-OTIB-0054, which provides a basis for assigning internal dose from unknown inhalation of MFAP mixtures. Radioactive decay of the isotopic mixture over the 2-year intake period was not considered, which is favorable to the claimant.

Note that the inventory in the UNH fuel was accumulated without radioactive decay until the final experiment, after which the activities of individual isotopes were decay-corrected for 180 days to yield these values.

The evaluation of air monitoring data (Section H.6), determined that a weighted average concentration of 13.5 dpm/m^3 for airborne alpha activity was favorable to the claimant. Inhalation intakes were computed by assuming that the airborne alpha activity consisted entirely of HEU having a specific activity of 70 µCi/g. Therefore, the airborne mass concentration was 8.7×10^{-8} g/m^3 and the total intake over the 4,000-hour period was 4.2×10^{-4} g.

The UNH fuel, as modeled, contained 2,782.8 g HEU. The ratio of total inhalation intake over the 2-year intake period and HEU solution mass gives an intake fraction $f = 1.5 \times 10^{-7}$. The inhalation intake I for the 36 dosimetrically significant nuclides for the 2-year period was then determined as the product of the ORIGEN-S result for each nuclide after 180 days of decay (activity A) and the intake fraction (i.e., $I = fA$).

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Table H-8. Accumulated activities of dosimetrically significant MFAPs in UNH fuel at the CML.

Isotope	Activity (Bq)
Mn-54	1.800E+02
Fe-55	1.664E+03
Co-58	1.359E+02
Co-60	3.077E+00
Sr-89	1.287E+04
Sr-90	2.506E+04
Y-90	2.507E+04
Y-91	1.984E+04
Zr-95	2.734E+04
Nb-95	5.025E+04
Mo-99	5.206E-14
Ru-103	4.732E+03
Ru-106	3.889E+03
Cd-113m	1.658E+00
Cd-115m	9.955E-01
Sb-125	3.236E+02
Te-129m	1.051E+02
Te-132	3.937E-11
I-131	8.936E-02
I-132	4.055E-11
Cs-134	7.004E-02
Cs-136	4.318E-02
Cs-137	2.567E+04
Ba-140	3.719E+01
La-140	4.281E+01
Ce-141	5.509E+03
Ce-144	5.417E+04
Pr-143	6.690E+01
Pr-144	5.417E+04
Nd-147	3.204E+00
Pm-147	2.171E+04
Pm-148m	0.000E+00
Sm-151	6.867E+02
Eu-154	1.691E-03
Eu-155	3.225E+02
Ta-182	0.000E+00

Committed organ doses were computed as the sum of products of the 180-day intakes, computed for each of the 36 dosimetrically significant nuclides and their corresponding inhalation DCFs from ICRP Publication 68 (ICRP 1995):

$$H(50) = \sum_{i=1}^{36} I_i DCF_i = f \sum_{i=1}^{36} A_i DCF_i \qquad \text{(H-4)}$$

The DCFs were partitioned into three solubility categories to account for different absorption types, using the method described in ORAUT-OTIB-0054 (ORAUT 2015b). Three committed dose values were therefore computed for each of the 25 individual organs defined in ICRP Publication 68 (ICRP

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1995), corresponding to soluble, moderately soluble, and insoluble materials. Table H-9 shows the maximum committed organ dose values for the three solubility categories.

Table H-9. Maximum committed organ doses from inhalation of airborne MFAPs at CML.

Absorption type	*H(50)* (Sv)	Tissue
Type F	2.5×10^{-9}	Bone surface
Type M	2.3×10^{-9}	Bone surface
Type S	2.4×10^{-9}	Lung

These values are much less than previously calculated committed doses of 3.7×10^{-7} Sv (soluble, bone surface), 4.0×10^{-7} Sv (moderately soluble, lung), and 6.1×10^{-7} Sv (insoluble, lung). Earlier committed doses were calculated using airborne concentrations derived by applying a resuspension factor to the DOE limit on removable surface contamination (which is favorable to claimants). The orders-of-magnitude difference in the two sets of dose values is due to a previous miscalculation in converting from 100 cm^2 to m^2. The corrected calculation uses a resuspension factor of 1.5×10^{-4} m^{-1} and 2,000 dpm/100 cm^2 removable alpha. This calculation is applied over the entire 220 m^2 footprint of the CML experimental and material storage areas, giving an estimated air concentration of 30 dpm/m^3. The 13.5-dpm/m^3 weighted average alpha air concentration from routine monitoring results used in calculations for this analysis further reduces the dose estimate from that obtained previously.

Previous estimates of MFAP inventories assumed a thermal power of 10 mW for 60 minutes, compared with the lesser (but still favorable to claimants) value of 6.7 mW for 70.5 minutes used in this analysis to describe conditions for the typical CML criticality experiment.

H.8 SUMMARY AND CONCLUSIONS

An evaluation of the unmonitored personnel dose, air monitoring data. and reactor performance estimates for the CML has resulted in estimates on the order of a few nanosieverts for maximum organ doses due to inhalation of resuspended contamination containing MFAPs. The greatest contributor to the large reduction in estimated doses is a correction in the calculation of a conversion factor. Lesser contributors to the reduction are lower estimates of reactor power in a typical criticality experiment at CML, and a lower value for respirable alpha air concentrations based on routine air monitoring results.

Based on this modeling, no significant personnel dose to RFP workers or contractors resulted from the generation of MFAPs in the UNH fuel or resuspended contamination from fuel spills as a result of criticality experiments at CML over its lifetime.

H.9 DATA SOURCES

The following are the complete lists of survey and sampling results that were reviewed as part of this evaluation:

- Radiological survey results. 156807, 156808, 156809, 156810, 156811, 156812, 156813, 156814, 156815, 156816, 156817, 156818, 152233, 152245, 152247, 152248, 152250, 152251, 152253, 152255, 152257, 152259, 152261, 152262, 152263, 152264, 152266, 152267, 152268, 152269, 152270, 152276, 152281, 152288, 152292, 152295, 156820, 152271, 152272, 156822, 156824, 156825, 156826, 159203, 159204, 159205, 159207,

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159210, 152229, 152273, 152277, 152280, 152284, 152286, 152289, 152293, 152299, 152301, 152302, 152304, 152318, 152323, 152325, 152331, 152335, 152340, 152342, 152343, 152344, 152346, 157168, 157170, 157172, 157173, 152298, 152306, 152308, 152310, 152312, 152314, 152317, 152320, 152326, 152351, 152353, 152355, 152327, 152328, 152329, 152330, 152332, 152341, 152348, 152357, 152359, 152360, 152362, 152500, 152502, 152507, 152508, 152509, 152511, 152515, 152516.

- Air sampling results. 159122, 159124, 159125, 159126, 159127, 159128, 159129, 159130, 159131, 159132, 159133, 159135, 159137, 159138, 159136, 159154, 156834, 159155, 159158, 159160, 159162, 156857, 156858, 156859, 156860, 156861, 156862, 156864, 156875, 156877, 156879, 159139, 156863, 156867, 156876, 156878, 160323, 159142, 159143, 159144. 159145. 159146, 159147, 159148, 156881, 156883, 156884, 156885, 156886, 156888, 156890, 156891, 156892, 156893, 156894, 156895, 156897, 156880, 156882, 156887, 156889, 160325, 156896, 159140, 159149, 159123, 159150, 159151, 159152, 159153, 159156, 159157, 156898, 156899, 156901, 156903, 156905, 156906, 159141, 156900, 156902, 156907, 159159, 159161, 159163.

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I.1 INTRODUCTION

This evaluation originated from the followup review of the SEC-00192 RFP evaluation report (NIOSH 2013a) by the RFP ABRWH Working Group. The SEC-00192 petitioner initially identified issues relating to allegations made by a former RFP worker in interviews on October 24 and 25, 1989, by the EPA National Enforcement Investigation Center's Office of Criminal Investigations and the FBI. These interviews resulted from a telephone call by the interviewee to the FBI Rocky Flats Hotline on June 16, 1989, alleging safety violations and manipulation of laboratory samples at RFP. A redacted transcript of the interviews was provided by the EPA Office of Criminal Investigations (EPA 1989a) and reviewed for this evaluation. Specifically, the EPA and FBI interviews were technically reviewed to assess the allegations and their relevance to potential data falsification and data invalidation in Building 123 in the context of the technical basis for dose reconstructions under EEOICPA.

I.2 INTERVIEWS

Most of the information and incidents described by the FBI interviewee do not provide sufficient detail to support a followup investigation of the claims. Four individuals with potential related knowledge or information that might have provided insight into the statements offered by the FBI interviewee were identified and interviewed. A summary of the interviews is presented below.

- A former RFP [redacted] who worked from [redacted] to [redacted] managing both [redacted] (ORAUT 2013h) was interviewed. He returned as a worker from [redacted] to [redacted] working in [redacted]. In [redacted] to [redacted], he also worked with the [redacted]as manager of their [redacted]. As such, his RFP experience was related to the period directly after the raid (he had no information or experience relating to the period before the raid). These observations have relevance if one assumes that the 1989 protocols in place in the FBI investigation timeframe were essentially the same as those in place when he began work in 1990, which is supported by Building 771 Laboratory sample-handling procedure reviews before and after the period around the 1989 raid (Fraser 1996). During the interview, the following observations were made (summarized):

  – Environmental monitoring and personnel dosimetry were separate programs, although their respective samples were analyzed in the same low-level onsite laboratory. Around 1997 to 1998, the onsite laboratory was shut down and everything was contracted out. Turnaround times on samples was sometimes a problem, especially for plutonium, but not as much for tritium. While the interviewee was there, there was no routine tritium monitoring program, just some pre- and post-job tritium analyses. There were no significant tritium intakes during his employment.

  – Based on his RFP experience, there is no specific link between environmental and bioassay results; the same numbers might have a different significance in environmental versus bioassay samples.

  – RFP had a state-of-the-art program for dealing with compromised personal protective equipment. They had CAM alarms, nasal swipes, and bioassay. Bioassay would be done immediately if there was a suspected exposure, not necessarily at the end of the work shift. RFP also had a wound counting program.

  – In bioassay analysis, the RFP laboratory staff used hoods. There was a complete industrial hygiene staff. The interviewee is sure the airflow was tested and is not aware

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of any injuries or complaints about hoods. Strong acids were used in bioassay for fecal samples, but the interviewee recalled no incidents.

- An individual serving as a RFP [redacted] from [redacted] to [redacted] and an RFP [redacted] from [redacted] to [redacted] was interviewed (ORAUT 2013i). Based on the employment period, this individual also only had RFP experience outside of the period of the FBI raid. The interviewee had no information or experience about the time before the raid and had no specific information or feedback pertinent to this review of the raid or data falsification at RFP.

- A former RFP worker indicated that the raid involved specific people; therefore, information was provided only on a need-to-know basis (i.e., those not specifically involved in the investigation received no information about what was going on). Because the raid was related to environmental issues (as opposed to occupational radiological issues), there was no involvement from the bioassay program perspective and no radiological program changes were made as a result of the raid. This interviewee was not informed of any aspects of that raid, but indicated that the department did not know the raid had happened until it was in the news. The interviewee also indicated that, at the time of the interview, no information had been presented about any aspect of the raid (ORAUT 2013j).

- A second former RFP worker confirmed that the raid involved specific people and only those who were involved received information about what was going on. The interview confirmed that the raid was related to environmental issues and not occupational radiological issues. Because there was no involvement from the bioassay program perspective, there were no radiological program changes made as a result of the raid (ORAUT 2013k).

Based on the information from the interviewees, the focus of the assessment and the raid was very specific to environmental impacts and monitoring. Because the personnel radiological monitoring program was not involved in the raid or associated assessment, there is no direct translation of the identified environmental findings and deficiencies into findings or deficiencies in the worker monitoring program. Further, no such formal allegations were made. Personal monitoring is the primary focus of individual dose reconstruction under EEOICPA.

I.3 *AN INSIDER'S VIEW OF ROCKY FLATS: URBAN MYTHS DEBUNKED*

An Insider's View of Rocky Flats: Urban Myths Debunked by a former RFP worker was reviewed for this attachment (Hobbs and Warden 2010). The author implies (and most articles accessed via the Internet seem to agree) that the FBI raid on RFP found no issues with worker protection or the worker monitoring program. The only violation cited for RFP was an environmental release.

An audit was performed by a DOE Special Assessment Environmental Team that focused on environmental issues, finding 95 deficiencies of varying types (DOE 1989a). The following two deficiencies mentioned the Building 123 laboratory:

- Under Radiation. Quality assurance (QA) and quality control (QC) practices for radiochemistry analyses in the Building 123 Health, Safety, and Environment Laboratory do not conform to generally accepted practices. The description of the finding further defines that the analyses were specific to environmental monitoring at RFP.

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- Under Quality Assurance. QA and QC practices at Building 123 Health, Safety, and Environment Laboratory for environmental analyses are not adequate to document validity of data.

Based on the information in the DOE Special Assessment report, the focus on the assessment was very specific to environmental impacts and monitoring. Although the report discusses data QA and validation issues with the analytical laboratories in Building 881 and Building 123, there were no situations identified in which falsification or invalidation of data would affect the ability to perform dose reconstruction under EEOICPA (DOE 1989a, Sections 7 and 8).

I.4 FOLLOWUP RESEARCH: AVAILABILITY AND ACCESSIBILITY OF RELEVANT DOCUMENTS

The initial documentation that related to the raid and subsequent litigation includes:

- A report from a DOE Environmental Special Assessment Team (one of four assessment teams that also included Management and Operations, Safety, and Legal Matters) (DOE 1989a). These teams were mobilized by the Secretary of Energy to perform a separate evaluation in parallel with the FBI investigation in order to provide the department with an independent assessment of RFP at the time of the raid.

- A 1995 symposium presentation titled *Are You Prepared To Survive an FBI Raid At Your Facility?* This specifically discusses aspects of the 1989 raid as well as the legal charges resulting from the raid (Swenson 1995).

- A detailed published response from a manager in the Environmental Department who was apparently an individual of investigative interest during the raid (*An Insider's View of Rocky Flats: Urban Myths Debunked*) (Hobbs and Warden 2010).

- A petitioner representative provided the DOE's Initial Agency Decision for Case No. VWA-0031 dated August 6, 1999 (DOE 1999a). This case involves a complaint from a former RFP worker alleging management reprisals after disclosures of possible health and safety violations and site mismanagement. The scope of the allegations is outside the period and location under evaluation in this attachment and does not affect NIOSH dose reconstructions for RFP.

Although it initially appeared that many of the site documents relevant to the period of the FBI's raid were sealed in files associated with the litigation, coordination with individuals who manage RFP site records resulted in access to additional pertinent documents that support this assessment. This attempt included efforts to locate the three remaining special assessment Tiger Team reports (for Management and Operations, Safety, and Legal Matters) that relate to the available DOE Environmental Special Assessment Team report (DOE 1989a). On August 15, 2013, the RFP records managers reported via e-mail that they performed an extensive search and were unable to locate the other three Tiger Team reports. The documents the RFP records managers did provide include:

- The complete Grand Jury Report, dated January 24, 1994, on the allegations and evidence from the FBI's RFP raid (DOJ 1992). This report provides specific explanations of the RCRA and environmental violations associated with the raid. No personnel monitoring violations or other occupational radiological monitoring program deficiencies are identified in the report.

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- Copies of RFP Occupational Radiological Control Program procedures, including pre- and post-raid versions for determining if the site revised its procedures as a result of the raid. In addition, a Building 771 Laboratory Sample Handling procedure includes a series of procedure revisions spanning the pre- and post-raid period. The revisions support the notion that no procedural changes resulted from the raid (Fraser 1996).

- Section I.7 lists SRDB Ref IDs that contain examples of post-raid air sampling, bioassay monitoring, contamination monitoring, nasal swab, and instrument operations procedures and manuals. According to one interviewee, the site did not commence archiving previous procedures until the late 1980s; before that, obsolete versions were destroyed when new versions were put into place. This previous policy might explain why the above archived procedures only go back to the early 1990s (ORAUT 2013j).

A collection of Denver EMCBC classified and unclassified documents was obtained that would support the assessment of impacts during the time of the FBI raid. These documents are stored at a classified document storage location. The information from the captured documents supports the notion that any radiological program document and/or procedure revisions that occurred around the time of the FBI raid were made as a result of a review and assessment of the Radiological Control Program directed by the site managers before the FBI raid. This site's review and assessment included responses to a GAO audit and Technical Safety Appraisal that appears to have included input from the Institute of Nuclear Power Operations (RFP 1984–1990; DOJ 1989a; FBI 1989a; DOJ 1989b; Sanchini 1988a; 1988b; Rockwell 1989d; FBI 1989b; Sanchini 1987; 1988c, 1988d; 1988e; FBI 2005; Sanchini 1988f; DOE ca. 1988; DOE 1989b; Norton 1989; DOJ 1989c; FBI 1989c; FBI 2001; Rigsby 1997; Simonson 1989; DOJ 1990; Middleton 2005; FBI 1989d).

Based on the review of the documents collected in total as part of the followup document collection, a conclusion was made that the reviewed information resulted in no impact that affects the ability to adequately reconstruct individual doses under EEOICPA. The specific information from the collected documents that support this conclusion includes:

- As stated by the U.S. Attorney in the RFP sentencing memorandum, there were no identified situations that posed an imminent threat to RFP workers, the public, or the surrounding environment (Swenson 1995, p. 13).

- Although the initial FBI investigation identified potential issues at RFP, the resulting FBI raid did not result in the same findings that initially seemed apparent to the EPA and FBI (based on the previous allegations and investigative characterizations that led to the raid) (Hobbs and Warden 2010).

- The end result was a settlement that included an agreement between parties eliminating further pursuit of individual indictments (Swenson 1995, pp. 12–14).

- The charges against Rockwell at RFP were specific to environmental RCRA and Clean Water Act Laws and the impact to the environment; the charges did not specifically call out a data falsification, data validity issues, or a data quality violation (Swenson 1995, p. 13–14).

I.4.1 Additional Colorado Visit for Data Capture and Interviews

Additional contacts that might have pertinent or contain information relevant to the issues being investigated as part of this followup were identified from ABRWH Working Group meetings. One

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individual (ORAUT 2013l) was scheduled for an unclassified telephone interview (followed by a classified telephone interview and followup call). A second individual (ORAUT 2015a) specifically requested a classified interview and was interviewed in a secure setting at a location in Idaho Falls. The data and information from these interviews are discussed here.

- The individual involved in the unclassified interview provided information about involvement in shredding documents (as pertinent to this petition and the EEOICPA dose reconstruction process). The individual indicated that the direction for document destruction came from RFP management. The individual discussed feeling uncomfortable about destroying the documents because they were originals containing information on RFP operations, including monitoring data and incident reports. Several potential interviewee names were provided during the interview and the individual provided examples of the documents that were destroyed as part of the process (examples were received after the interview). The individual also stated a desire to participate in a secure interview to discuss other information relating to the document destruction process that might be classified.

  A followup interview was performed in February 2014 (ORAUT 2014f) to collect additional information and documentation that was discussed during the initial interview with the first interviewee. A secure telephone interview was scheduled and completed in May 2014 at the request of the interviewee (ORAUT 2014g) and all information discussed during that interview was cleared for uncontrolled use and dissemination. This secure telephone interview resulted in information from the initial interview being restated with additional information about what was believed to be the projects related to the documents that were destroyed. A final followup telephone discussion occurred in June 2014 (ORAUT 2014h) to discuss the results of the document and/or information reviews and information provided during the preceding interviews.

  The findings of the discussions and information reviews provided by this interviewee are as follows:

  – While the documents could have been field surveys, it does not appear that those surveys have an impact on the ability to bound or reconstruct dose for the class as long as the personnel monitoring data exist. Based on a review of some of the files that were provided as examples of documents that the interviewee believed were destroyed, the records were found to exist in the associated personnel files in NOCTS. Therefore, it appears unlikely any dose records were destroyed.

  – A potential additional interviewee was mentioned during this interview who was already on the list of desired interviewees; however, there was no success coordinating with this individual to schedule an interview.

- The individual interviewed during the classified interview at Idaho Falls (as part of the December 2013 Denver data capture trip) provided information about the assessment of data falsification at RFP. This individual did confirm the separation of the Environmental and Occupational Radiological Analysis programs within Building 123. The individual relayed information pertinent to the ability to reconstruct radiation dose about bioassay and personnel monitoring and specifically involved information about penciling-in and changing dosimeter readings and misplaced or lost bioassay samples, as well as contamination incidents and safety issues at RFP.

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Some specific information from this interview is:

– Penciling-in dosimeter information. The interviewee discussed a situation where dosimetry technicians wrote dose rate information in reports in pencil, which would allow RFP management to later direct changes to keep the production going at RFP. It was relayed that this issue was addressed through a grievance at RFP.

  Based on NIOSH professional judgment and field experience, the issue of penciling-in information appears to refer to radiological field survey records that directly relate to ongoing production operations. The only dosimetry information that might be included in such field surveys would be from direct-reading dosimeters or personal ion chambers (PICs, a.k.a. pencil dosimeters). While field survey information is used for comparison purposes in the performance of EEOICPA dose reconstruction, the primary and most important source of radiological information for the purpose of individual dose reconstruction is the individual TLD dosimetry and bioassay information. TLD and bioassay analyses are performed in a laboratory and not documented in the field, in contrast to the surveys and reports discussed in the claim raised by this individual. Therefore, it is not expected that the original handwritten documents that the interviewee referred to in the destruction process are about an individual's TLD or bioassay results (with electronic readouts; Inkret undated; RMRS 1999; RFP 1994).

– RFP bioassay issues. The interviewee relayed personal concerns with the bioassay program and relayed a specific concern about bioassay sample analysis results (false positives and [statistical] variations in bioassay results (see the similar second interviewee claim and response in Section I.4.2 below. While no other specific concerns about falsification of records were brought up, the interviewee did raise other general concerns about bioassay sample handling and processing.

  This concern does not raise issues that invalidate the use of personnel bioassay data in the performance of dose reconstructions under EEOICPA. The dose reconstruction process accounts for the potential for missed doses and incorporates methods that are favorable to the claimant. Therefore, the conclusion was made that this issue does not affect the dose reconstruction process.

– Personnel contamination problems and other contamination incidents. The interviewee brought up a significant number of radiological and some nonradiological contamination incidents and exposure-control issues in reference to the overall safety program at RFP.

  While contamination incident and survey data are used to supplement the personnel monitoring data in the performance of dose reconstructions under EEOICPA, personnel monitoring data are considered the primary data sources for the process. Therefore, the conclusion was made that this issue does not affect the dose reconstruction process.

– Tritium bubblers, neptunium, MgTh alloy, and CML. The interview provided some information on these other issues at RFP.

All of these issues are addressed in various sections of this TBD.

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I.4.2 Site Visit and Data Capture Interviews after Denver Visits

Five additional potential interviewees with information on the subject were identified from ABRWH Working Group meetings, petitioner information, and individual interviews. Attempts were made to schedule interviews with each of them. Two of the five were contacted and interviewed, one about data falsification and one about the FBI raid; the remaining three either did not agree to be interviewed or did not return messages requesting an interview. Only the data and information from interviews as part of the effort are discussed in this Attachment.

I.4.2.1 Data Falsification Interviewee

The individual interviewed about data falsification (ORAUT 2014i) had served as a [redacted] and [redacted] during employment at RFP. The interviewee confirmed that Environmental Radiological Program changes did occur as a result of the FBI raid. The interviewee also brought up concerns associated with personal radiological monitoring records and the radiological monitoring program at RFP, which documented varying positive and negative bioassay results in an individual's dose records.

No information from this interview supported the allegation of document destruction activities at RFP. While the individual discussed concerns with the implementation of radiological limits and controls as well as dose reporting during employment, up to the point of the implementation of the DOE Radiological Control Manual at the site (late 1980s to early 1990s) and the FBI raid, there were no identified effects on the ability to bound dose for the portion of the class of RFP workers being assessed for this attachment. It was confirmed that the focus of the FBI raid was environmental radiological issues and that program changes did occur as a result of the raid. The other concerns that were relayed were associated with personal radiological monitoring records and the documentation of the statistical or sample-counting variations that can produce positive and negative bioassay results associated with the analysis of a potential exposure situation in an individual's dose records. This interviewee specifically mentioned situations in which the site would increase limits to preclude exceeding them. In other cases, the interviewee believed that while reported results would indicate "no data available" but that that there were results that were available (i.e., based on their knowledge of the area survey results, personnel should have a recorded dose value). The interviewee indicated that when there were problems with the dosimetry readings, the individuals would receive an average of his coworkers' doses. The interviewee believed that there are cases where these averages are incorrect. These issues were assessed by the ORAU Team, which considered the interviewee information, reviewed worker files for comparable or corroborating situations, and assessed the effects of the verified issues on the completion of the EEOICPA dose reconstruction process. The assessment specifically focused on potential negative effects on individual dose reconstructions based on the interviewee's concerns. The ORAU Team did not identify situations or issues that would affect the ability to reconstruct dose for the RFP worker class being assessed as part of this attachment.

I.4.2.2 FBI Raid Interview and Followup Information Reviews

The SEC-00192 RFP petitioner identified the lead FBI agent involved in the 1989 FBI raid and provided the applicable contact information for the former agent (ORAUT 2014j). The agent was interviewed and provided a significant number of papers and documents about the FBI's raid of RFP in 1989. Many of the documents were the property of other government agencies or entities and required release by the applicable agency general counsel before they could be used or referenced in

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a EEOICPA project document. The following subjects and issues were identified during the interview and subsequent followup with the agent:

- The contention of the agent was that there was an additional August 1989 aerial MSS at RFP in addition to the one in June and July of 1989. The agent submitted a FOIA to DOE requesting that additional MSS information be provided. DOE completed searches and discovered no additional survey information. The agent submitted an appeal of that response, which was subsequently denied by the DOE Office of Hearings and Appeals citing that the search was adequate and no additional information is available from Legacy Management (DOE 2012).

- Part of the response to the FBI raid at RFP was for DOE to initiate its own special assessment of the site using Tiger Teams, which paralleled the operations of the FBI raid. The report that is currently available is titled *Assessment of Environmental Conditions at the RFP*, written by the U.S. DOE Special Assignment Environmental Team in August 1989 (DOE 1989a). In addition, three other reports for management and operations, safety, and legal matters are identified; however, those reports are not available and do not appear to exist based on extensive coordination and searching with the FBI agent as well as DOE and Legacy Management in Denver.

- There is a contention that the flyover data indicate the presence of the isotopes ^{137}Cs and ^{90}Sr, which is used to imply that an unreported criticality occurred at RFP. This relates to the contention from the agent about the existence of an August 1989 flyover MSS.

  No August 1989 flyover survey has been found. In addition, no specific information was found that supported a criticality event. Individuals were interviewed to assess the criticality claim with the following results:

  – Subsequent interview discussions and a report obtained from an interviewee do not corroborate the occurrence of a criticality at RFP. This includes the "1989 Criticality Safety Assessment at Rocky Flats" of the *Assessment of Environmental Conditions at the RFP* (DOE 1989a, p. 277). The Assessment Team found no indication that a criticality accident ever occurred at RFP.

  – An interviewee provided a report after his interview that concludes there is no evidence to support that a criticality occurred at RFP (Scientech 1989). The report does go on to identify issues with the Criticality Safety System and Program at the site and includes fixes for the issues, but no conclusive evidence, monitoring or other, was identified to support the occurrence of a criticality incident.

- The agent discussed the Waste Stream Identification and Characterization (WISC) Reports and identified them as a major source of information to support the raid at RFP. The entire set of WISC reports was obtained and reviewed for applicability to the SEC evaluation and dose reconstruction processes for RFP. While the information might have been useful to characterize conditions and issues at the time of the raid, no information to support evaluation of or reconstruction of dose at RFP were identified (Rockwell 1987i to 1987ah).

- As part of the response to the interview, the agent provided over a thousand pages of documents. Other than the documents specifically called out in this section, the remainder are considered the property of another government entity or legal interest, and coordination

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between NIOSH, Centers for Disease Control and Prevention, Office of General Counsel, and the legal representatives of the other organizations is required for approval to obtain and reference the applicable documents. The documents applicable to the assessment presented here include: (1) RFP personnel interviews (by both the FBI and EPA) at the time of the raid; (2) FBI research leading up to the raid; (3) Building 123 information associated with the FBI investigation and warrants; and (4) 1988 aerial survey documentation of the RFP site. The documents that are the property of the FBI were released by the FBI General Counsel on January 26, 2015; the general counsel release approval for the DOE and EPA documents was obtained on April 21, 2015:

- FBI Interviews. FBI (1991a to 1991c, 1989e to 1989i).
- EPA Interviews. RFP (1975–2004, pp. 1214, 1215–1222, 1223–1224, 1225–1226, and 1227).
- FBI research documents leading up to the raid. RFP (1975–2004).
- Building 123 notes about FBI investigation and warrants. DOJ (1989a to 1989c).
- 1988 aerial survey of the RFP site. Author unknown (1988); note that no 1989 survey identifying radiological levels to support a criticality has been found.

- The agent discussed and provided some notes from a personal notebooks of an RFP manager about the RFP radiological program (Sanchini 1988b). Additional portions of the notebooks were obtained in subsequent data capture efforts (Sanchini 1987, 1988a, 1988c to 1988e). While some notes discussed needed improvements in the radiological program and identified program confidence issues, they do not indicate program deficiencies that represent an inability to bound dose.

- The agent's response to the documented interview included the identification of a significant number of names and contacts associated with the raid. Of these names, five individuals' contact information was identified and three of the five were successfully contacted and agreed to an interview. The three interviews are summarized below:
  - The first interview (ORAUT 2014k) was with the DOE Senior Site Representative at the time of the FBI raid. This individual indicated awareness that the issues were associated with environmental violations and not with occupational radiological monitoring violations. The individual also discussed issues that involved personnel complaints about availability of monitoring records, which the interviewee deemed hearsay.
  - The second interview (ORAUT 2014l) was with a Rockwell and DOE Office spokesperson. During the raid, the individual mainly dealt with the news media and provided responses about the raid. Other than that, the individual had no direct involvement in dealing with the raid. Part of the issue that resulted in the raid involved late night burning and operations of a waste incinerator. All the issues of which the interviewee was aware were environmentally related and had no occupational radiological components. This individual did have some personal records at home and coordination is occurring to obtain copies of any pertinent information.

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– The third interview (ORAUT 2014m) was with an individual who had never been an employee or contractor at RFP. The only two associations this person had with RFP was as part of a Governor's panel to investigate unreported criticalities and serving as part of a NIOSH Health Surveillance Program. As part of the response, the individual reported that no indications were identified that supported the occurrence of a criticality at RFP. The individual provided a copy of the report that followed. As part of the investigation, the individual looked for anomalies in personnel doses as well as flyover data. The investigation revealed no suspicious information and indicated that there was good continuity in the data. The Health Surveillance Program also included the assessment and revision of site personnel neutron doses.

The information from the FBI agent, including interview information (from the agent and other knowledgeable individuals identified by the agent) and associated documentation, support the idea that the basis of the FBI raid was RFP environmental issues. While some information collected and assessed at the time of the raid does cross over into occupational radiological issues (including the RFP manager personal notebooks), nothing was discovered that supports a data falsification or destruction issue that would affect the ability to reconstruct dose for the RFP worker class being assessed as part of this attachment. The review of the flyover and MSS information indicates that there might have been support for the FBI raid from an environmental perspective. However, these documents and files provided no evidence or information that disputes the ability to bound RFP worker dose under the EEOICPA program. The claim of an unreported criticality incident at RFP was investigated from several points of view in the documents that were reviewed, as well as during the interviews of knowledgeable individuals; no supportive evidence of a criticality incident was found. The conclusion was made that no information about this issue affects the ability to reconstruct individual dose under EEOICPA.

I.4.3 Review of Petitioner-Identified Technical Safety Appraisal Issues

As part of the followup issues raised by the petitioner during the Idaho Falls ABRWH meeting, a GAO report was referenced (GAO 1988) that, in turn, referenced an RFP Technical Safety Appraisal (TSA) (DOE 1988b). The petitioner specifically cited the following statements from the TSA:

- DOE 1988b, pp. 10–11. *For example, radiation monitoring is adversely affected by poor quality instrumentation, inadequate calibration techniques, and improper use of equipment. The Radiological Health Quality Assurance Program is ineffective as evidenced by some of the preceding concerns, failure to comply with DOE-prescribed standards, and deficiencies in maintaining exposure records and tracking bioassay samples.*

- DOE 1988b, p. 174. *During the past few weeks, several SAAM's were turned off without notifying either radiation monitoring or the instrument technicians. These instruments were operational when turned back on. There is no electronic method to automatically display their operational status in the monitoring office.*

- DOE 1988b, p. 180. *The health physics instruments used for personnel protection do not all conform to appropriate performance requirements of applicable standards.*

The RFP TSA was reviewed for information applicable to the assessment of personnel exposure information, records, and exposure tracking. The appraisal contains information about the radiological interface between the RFP radiological protection personnel and other plant organizations as

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described in the report; Section L focuses specifically on performance objectives, findings, and concerns about radiological protection (DOE 1988b, pp. 161–191). The concerns identified in Section L are later listed in Appendix C, "List of Concerns," under "L. Radiological Protection" (DOE 1988b, pp. 243–245).

The reviewed information was assessed to determine if the TSA findings have any substantial effects on the dose reconstruction process. In the "Major Problem Areas" summary of the appraisal (see the first bullet above; DOE 1988b, pp. 10–11), there are two noted deficiencies that directly relate to the primary dose reconstruction data sources: (1) proper use of radiation monitoring equipment [dosimetry], and (2) maintaining exposure records and tracking bioassay samples.

- The first issue arises from the observation of inadequate external dosimeter placement on a person's body. The provided example was truck drivers wearing dosimeters on their chest when radioactive materials were behind them (Concern RP.5-2; DOE 1988b, pp. 171, 244).

  Although any determinations on the reconstruction of dose for offsite shipments and/or couriers is outside the scope of this review, the assessment of external exposure geometry issues related to this topic was performed as part of previous evaluations, specifically in the SEC-00030 RFP evaluation report (NIOSH 2006). As such, the external dose adjustment to account for exposure geometry is considered an individual dose reconstruction issue to be handled on a case-by-case basis.

- The second issue involves internal dosimetry and bioassay and is associated with tracking routine and special bioassays, the need for an independent audit program, and the need for a program to compare in vivo and in vitro results for particular employees. These issues do not speak to a lack of internal dosimetry data, but rather methods to better track and compare them.

The remaining issues in the statements cited by the petitioner in the bulleted paragraphs above are about field monitoring surveys and instrumentation associated with RFP operations; they are not specific to the personnel internal or external radiological monitoring data used on the NIOSH dose reconstruction project.

Based on a review of the RFP TSA in its entirety, there was no identified effect on radiological personnel monitoring data that are used to support bounding doses or dose reconstruction for the RFP worker class assessed.

I.4.4 Review of Petitioner-Identified Data Falsification Issues Report

The petitioner-provided document (DOE 1999b) in support of data falsification issues is associated with a complaint filed by an RFP worker under the DOE Contractor Employee Protection Program, 10 CFR Part 708. In the complaint, the worker contended that reprisals by the employer were taken after certain disclosures were made concerning possible health and safety violations and mismanagement at the RFP site. It was found that the employer did take acts of reprisal against the employee prohibited under 10 CFR § 708.5, and that the employee was entitled to remedial action from the employer.

The petitioner-provided report centers on an allegation of records falsification involving mislabeling waste for shipment off site. The findings discuss allegations of potential retaliation for protected activities; however, it only contains potential effects in terms of environmental dose. No information

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affects the ability to reconstruct radiation dose with sufficient accuracy under the EEOICPA Program for the RFP worker class assessed.

I.4.5 Additional Information from RFP Manager Personal Notebooks

On January 28, 2015, the petitioner provided excerpts from the notebook of the previously discussed RFP manager who made notes on the radiological program and who was present at the time of the FBI raid. The petitioner made reference to the notebook statement that the program data was "bad," but the statement in the notebook is in reference to collecting samples but not counting them (Sanchini 1998e). On March 30, 2015, the petitioner provided additional excerpts from the same notebooks denoting that the notebooks indicate "questionable" operations in the Building 123 bioassay laboratory (Sanchini 1986–1988, 1988–1989).

While some information collected and assessed at the time of the raid does cross over into occupational radiological issues (including the RFP manager personal notebooks), nothing was discovered that supports a data falsification issue that would affect the ability to reconstruct dose for the RFP worker class assessed.

I.5 ASSESSMENT OF AVAILABLE PERSONNEL RADIOLOGICAL MONITORING DATA

As part of the followup to the assessment for this attachment, a review was performed of the quantity of available personnel radiological monitoring data available. A conclusion was made that there exists a sufficient quantity of individual external monitoring data to support the assessment of RFP personnel external doses.

Specifically, there are sufficient individual external dosimetry TLD data to support reconstruction of RFP personnel external dose for after 1983 using the methods defined in the main text. Figure I-1 shows the number of workers who were badged over various years. The solid line derives from EEOICPA claim data; the broken line shows the results of a similar analysis on the RFP electronic files for the entire plant population. The difference is due to discrepancies in termination dates in the electronic files. These dates are verified as worker files are reviewed, providing a better understanding of the badging process as well as a more accurate individual worker record.

For RFP personnel internal doses, an assessment of the quantity of data available from after 1983 was performed. The assessment involved a review of the internal RFP co-exposure study of data from 1983 to 1988 (which was later extended beyond 1988; see Attachment D). The review identified the results shown in Table I-1.

Information on the variances in numbers of bioassay samples notes that there was a significant increase in employment in 1984 (to a peak of 5,990 in 1984), and that plutonium processing ceased in 1990. While no clear correlation can be made for 1985 to 1986 and 1988 to 1989, the 1985 date does correspond to the incorporation of new air sample filter media in the air monitoring program, and the 1989 date corresponds to EG&G taking over the site and the occurrence of the FBI/EPA raid at RFP.

Based on this information and the information in the preceding sections of this assessment, there is no issue associated with the FBI raid or issues about data falsification or invalidation that affect the data from after 1983 in a way that would preclude individual dose reconstructions with sufficient accuracy under EEOICPA.

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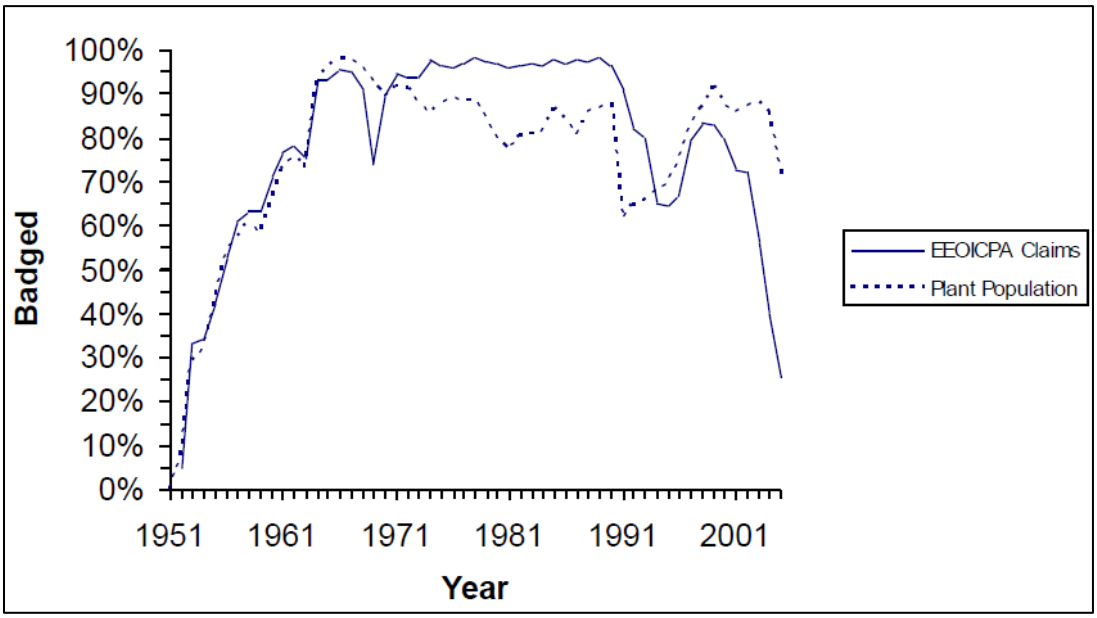

Figure I-1. Percent of workers badged for dosimetry (NIOSH 2006, Figure 6-1).

Table I-1. Available internal sample data, 1980 to 1995.

Year	Total urine samples	Pu urine	U urine	Pu fecal	Total in vitro samples	Total lung counts
1980	5,906	5,628	278	0	5,906	4,050
1981	6,349	5,991	358	0	6,349	4,384
1982	8,527	7,109	1,418	0	8,527	4,876
1983	9,181	7,506	1,675	0	9,181	6,052
1984	10,468	8,830	1,638	0	10,468	6,777
1985	10,381	8,767	1,614	0	10,381	5,922
1986	7,075	5,893	1,182	0	7,075	5,460
1987	3,979	3,429	550	0	3,979	4,966
1988	4,808	4,415	393	0	4,808	5,735
1989	6,977	4,777	2,200	78	7,055	3,618
1990	5,172	4,168	1,004	69	5,241	1,799
1991	495	483	12	225	720	638
1992	960	762	198	758	1,718	553
1993	1,550	1,197	353	368	1,918	663
1994	3,325	2,722	603	770	4,095	892
1995	2,106	1,688	418	69	2,175	239

I.6 GENERAL SUMMARY AND CONCLUSIONS

This section provides a general summary of the sequence of topics addressed above:

- A review of the interviewee allegation relevant to data falsification and data invalidation in Building 123 was performed. The interviewee made statements about the inadequacy of fume hoods and the improper handling and preparation of environmental, bioassay, fecal coliform, and stack samples. From a radiological perspective, no scientific basis was found for concluding that the issues raised about environmental samples would compromise radiological count results, and the reviewed information does not corroborate a link between the environmental and occupational radiological programs. Nevertheless, four individuals were

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interviewed with potential related knowledge or information who confirmed that the focus of the assessment and the FBI raid was very specific to environmental impacts and monitoring. There is no relationship between the identified environmental findings or deficiencies to the worker monitoring program, and no such formal allegations were made.

- A review of *An Insider's View of Rocky Flats: Urban Myths Debunked* was performed (Hobbs and Warden 2010). The authors implied that the FBI raid found no issues with worker protection or the worker monitoring program; the only violation cited for RFP was an environmental release. Based on the DOE Special Assessment report, the focus of the assessment was very specific to environmental impacts and monitoring (DOE 1989a). Although the report discusses data QA and validation issues with the analytical laboratories in Building 881 and Building 123, the conclusion was made that no situations were identified in which falsification or invalidation of data would affect the ability to perform dose reconstruction under EEOICPA.

- RFP site records managers provided access to additional pertinent documents that support this assessment, including the complete Grand Jury Report, RFP Occupational Radiological Control Program procedures, and examples of post-raid air sampling, bioassay monitoring, contamination monitoring, nasal swab, and instrument operations procedures and manuals. Based on its review, the ORAU Team concluded that the information does not affect the ability to adequately reconstruct individual doses under EEOICPA. There were no identified situations that posed an imminent threat to RFP workers, the public, or the surrounding environment. The FBI raid did not result in the same findings that initially led to the raid. Although Rockwell pled guilty and paid a fine, it appears the settlement was based on the company's desire to close the prolonged litigation. Furthermore, the charges against Rockwell were specific to environmental RCRA and Clean Water Act Laws and environmental impacts; the charges did not specifically call out a data falsification, data validity issues, or a data quality violation.

- One individual provided information about involvement in shredding documents. The ORAU Team concluded that while the documents being destroyed could have been some kind of field surveys, it does not appear that those surveys have an effect on the ability to bound or reconstruct dose, as long as the personnel monitoring data exist. Based on a review of some of the files that were provided as examples of documents that the interviewee believed were destroyed, it was determined that the records did exist in the associated personnel files in NOCTS; therefore, those files were not destroyed.

- An interviewee raised the issue of dosimetry technicians writing down dose rate information in pencil, which would allow RFP management to later direct changes to keep the production going. Based on professional judgment and field experience, the issue of penciling-in information appears to refer to radiological field survey records that directly relate to ongoing production operations. The only dosimetry information that might be included in such field surveys would be from direct-reading dosimeters or personal ion chambers, which are used for comparison purposes in EEOICPA dose reconstruction. The primary and most important source of radiological information for individual dose reconstruction is the individual TLD dosimetry and bioassay information. TLD and bioassay analyses are performed in a laboratory and not documented in the field.

  In addition, this interviewee relayed concerns about bioassay sample analysis results (false positives and [statistical] variations); bioassay sample handling and processing; personnel

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contamination and contamination incidents; and issues about tritium bubblers, neptunium, MgTh alloy, and the CML. The ORAU Team concluded that the concerns about bioassay do not raise any issues that invalidate the use of personnel bioassay data in the performance of dose reconstructions under EEOICPA. The dose reconstruction process accounts for the potential for missed doses and incorporates methods that are favorable to the claimant. Contamination incident and survey data are used to supplement the personnel monitoring data in the performance of dose reconstructions under EEOICPA, but personnel monitoring data are considered the primary data sources for the process. All of the issues concerning tritium bubblers, neptunium, MgTh alloy, and the CML are outside the scope of this attachment and are addressed in other topic-specific documents.

- Two of five potential interviewees were interviewed based on information identified from ABRWH Working Group meetings, petitioner information, and individual interviews. The remaining three either did not agree to be interviewed or did not return messages requesting an interview.

  – The first interviewee confirmed that Environmental Radiological Program changes did occur as a result of the FBI raid and brought up concerns associated with personal radiological monitoring records and the RFP radiological monitoring program, which documented varying positive and negative bioassay results in an individual's dose records. The ORAU Team concluded that no information provided during this interview supported the allegation of document destruction activities at RFP or affected the ability to bound dose for the portion of the class of RFP workers being assessed.

  – The second interviewee was the lead FBI agent in the 1989 FBI raid, who provided a significant number of papers and documents about the raid. Many of these documents are awaiting release by the applicable agencies. The agent contended that there was an additional August 1989 aerial MSS at RFP in addition to the one in June and July of 1989, and that the flyover data indicate the presence of the isotopes ^{137}Cs and ^{90}Sr, which was used to imply that an unreported criticality occurred. NIOSH could locate neither the August 1989 flyover survey nor evidence supporting a criticality event. Other individuals were interviewed to assess the criticality claim and could produce no corroborating evidence. The information from the FBI agent supports the idea that the basis of the FBI raid was environmental issues. The ORAU Team concluded that the interview, documents, and files provide no evidence or information that disputes the ability to bound RFP worker dose under the EEOICPA program.

- A review of the entire RFP Technical Safety Appraisal (TSA; DOE 1988b) was made after the petitioner cited three statements concerning poor-quality instrumentation, standards, and record tracking. In the "Major Problem Areas" summary of the appraisal, there are two noted deficiencies that directly relate to the primary dose reconstruction data sources for NIOSH under EEOICPA: (1) proper use of radiation monitoring equipment [dosimetry]; and (2) maintaining exposure records and tracking bioassay samples.

  – The ORAU Team concluded that any determinations on the reconstruction of dose for offsite shipments or couriers is outside the scope of this review. However, the assessment of external exposure geometry issues related to this topic was performed as part of previous evaluations and is incorporated into other sections of this TBD. As such, the external dose adjustment to account for exposure geometry is considered an individual dose reconstruction issue to be handled on a case-by-case basis.

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  - The second issue involves internal dosimetry and is associated with tracking routine and special bioassays, the need for an independent audit program, and the need for a program to compare in vivo and in vitro results for particular employees. The ORAU Team concluded that these issues do not speak to a lack of internal dosimetry data but rather to methods to better track and compare these personnel monitoring data. Based on its review of the RFP TSA in its entirety, there is no identified effect on radiological personnel monitoring data that are used to support bounding and/or reconstructed dose for the RFP worker class assessed.

- A petitioner-provided report was reviewed centering on an allegation of record falsification involving mislabeling waste for shipment off site. The findings discuss allegations of potential retaliation for protected activities; however, it only contains potential impacts in terms of environmental dose. The ORAU Team concluded that none of this information affects the ability to reconstruct radiation dose with sufficient accuracy under the EEOICPA Program for the RFP worker class assessed.

- The petitioner provided excerpts from the notebooks of an RFP manager who made notes on the radiological program and who was present at the time of the FBI raid. The ORAU Team concluded that while some information collected and assessed at the time of the raid does cross over into occupational radiological issues, nothing was discovered that supports a data falsification issue that would affect the ability to reconstruct dose for the RFP worker class assessed.

- A review was performed of the quantity of personnel radiological monitoring data available at the time of this assessment. The ORAU Team concluded that there exists a sufficient quantity of individual external monitoring data to support the assessment of RFP personnel external doses.

Based on this information and the information in the preceding sections of this attachment, issues associated with the FBI raid or data falsification or invalidation do not affect the post-1983 data and do not preclude individual dose reconstructions under EEOICPA.

I.7 EXAMPLES OF POST-RAID PROCEDURES AND MANUALS

The following SRDB Ref IDs contain examples of postraid air sampling, bioassay monitoring, contamination monitoring, nasal swab, and instrument operations procedures and manuals that were reviewed for this analysis: 126927, 126926, 126925, 126924, 126923, 126922, 126920, 126919, 126917, 126916, 126913, 126906, 126902, 126901, 126899, 126896, 126895, 126892, 126890, 126888, 126887, 126886, 126885, 126884, 126883, 126882, 126881, 126880, 126878, 126877, 126876, 126874, 126873, 126872, 126871, 126869, 126868, 126867, 126866, 126864, 126863, 126836, 126833, and 126831.