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☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

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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.

TRADEMARK INFORMATION

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ACRONYMS AND ABBREVIATIONS

AMAD	activity median aerodynamic diameter
CEDR	Comprehensive Epidemiology Data Resource
Ci	curie
cm	centimeter
cpm	counts per minute
CWT	chest wall thickness
d	day
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
DTPA	diethylenetriaminepentaacetate
DU	depleted uranium
EDTA	ethylenediaminetetraacetate
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EU	enriched uranium
g	gram
GSD	geometric standard deviation
hr	hour
HSDS	Health Sciences Data System
HTO	tritiated water vapor
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
IREP	Interactive RadioEpidemiological Program
keV	kilovolts-electron, 1,000 electron volts
kg	kilogram
L	liter
L X-ray	low-energy X-ray
LLNL	Lawrence Livermore National Laboratory
MDA	minimum detectable activity
min	minute
mL	milliliter
MLT	minutes live time
mm	millimeter
MPLB	maximum permissible lung burden
mrem	millirem
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
NOCTS	NIOSH-Office of Compensation Analysis and Support Claims Tracking System
ORAU	Oak Ridge Associated Universities

pCi	picocurie
PER	Program Evaluation Report
PGT	Princeton Gamma Tech
PHA	pulse height analysis
POC	probability of causation
ppm	parts per million
RFP	Rocky Flats Plant
ROI	region of interest
s	second
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
TBD	technical basis document
TBP	tributyl phosphate
TOPO	trioxyl phosphene oxide
TTA	thenoyltrifluroacetone
U.S.C.	United States Code
WG	weapons-grade
wk	week
yr	year
ZPPR	Zero Power 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 sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer facility" or a "Department of Energy [DOE] facility" as defined in the Energy Employees Occupational Illness Compensation Program Act [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA defines a DOE facility as "any building, structure, or premise, including the grounds upon which such building, structure, or premise is located … in which operations are, or have been, conducted by, or on behalf of, the Department of Energy (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program)" [42 U.S.C. § 7384l(12)]. Accordingly, except for the exclusion for the Naval Nuclear Propulsion Program noted above, any facility that performs or performed DOE operations of any nature whatsoever is a DOE facility encompassed by EEOICPA.

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

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

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

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

5.1.1 Purpose

This technical basis document (TBD) 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.

The primary modes of intake would have been chronic or acute inhalation or through breaks in the skin (wounds). The primary bioassay data are the urine data (the activity of the radionuclide of interest that is excreted in the urine after an inhalation or wound intake) and the lung count data (the activity of the radionuclide present in the lungs after an inhalation intake) [1]. Sections 5.3 and 5.4 discusses these two data sets 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.

5.1.3 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. For this reason, the following class of RFP employees has been added to the Special Exposure Cohort (SEC) (NIOSH 2013):

> *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.*

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.

5.2 SOURCE TERM

In 1993, the Secretary of Energy formally announced the end of nuclear production at Rocky Flats. Remediation was completed at the RFP in late 2005. Coworker 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:

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

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

3. 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.

For weapons-grade (WG) plutonium, which was present at RFP throughout most of its 1952-to-1989 production history, the ratio of the activity of ^{241}Pu to the alpha activity of the other plutonium isotopes is 5.1 and the ^{240}Pu content is about 6% by weight. Table 5-1 lists the weight percent and fraction of alpha activity for each isotope.

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

Isotope	Weight percent	Fraction of alpha activity[b,c]
Pu-238	0.01	0.023
Pu-239	93.79	0.8
Pu-240	5.8	0.18
Pu-241	0.36[b]	–
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 (RFETS 2002, p. 120).
c. – = not applicable.

The Zero Power 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 [2].

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

Isotope	Weight percent	Fraction of alpha activity[b]
Pu-239	87.6	0.7
Pu-240	10.0	0.3
Pu-241	2.4	–

a. These ZPPR values are based on extracted data in a working file from an undocumented source.
b. – = not applicable.

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 multiplying the ^{239,240}Pu activity by [48.2 × ppm ^{241}Am ÷ (1×10^6 – ppm ^{241}Am)]. For ZPPR plutonium, the ^{239,240}Pu activity is multiplied by [44.6 × ppm ^{241}Am ÷ (1×10^6 – ppm ^{241}Am)] to obtain the activity of ^{241}Am in the intake mixture [3].

5.2.1.2 Plutonium Solubility and Particle Size

Most plutonium in metalworking operations and fire incidents was insoluble (i.e., type S). Exceptions such as plutonium metal in solvents such as carbon tetrachloride can be assumed to be more soluble (type M) if this is what the data show or if it is more favorable to claimants to do so [4].

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 Super S) characteristics [5].

Plutonium in chemical processing operations can be either soluble (type M), insoluble (type S), or a mixture of solubilities. Dose reconstructors should select the material type that is most favorable to the claimant [6]. 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 1965a,b), 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. An approach that is favorable to claimants is to assume 1-µm AMAD for all plutonium fires unless the qualifying cancer involves the tissues of the extrathoracic regions [7].

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 clean up 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 [8].

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 1994a, p. 83). 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 1994b, p. 79).

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) 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, 1994b,c). All of these compounds were involved in the recovery and recycling processes for EU in Building 881 (RFETS 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 (see 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. 2-38).

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 (RFETS 2000a). The solubility classification is ambiguous, falling somewhere between type S and type M (RFETS 1998a, Section 6.1, p. 165; 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 et al., 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 coworker, 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; Ulsh 2008; Author unknown 1976). 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 existence of thorium on site in the early 1950s through the development of internal and external thorium-monitoring processes (Dow 1953, 1956, 1958).

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 (RFETS 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, NIOSH has determined that unmonitored neptunium doses at the RFP cannot be reconstructed.

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 follow-up DTPA chelation. Assume a 24-hour excretion period unless the record indicates that the actual excretion period was different [9].

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 [10].

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 [11].

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 [12].

Another source of interference was contamination of the tracer (^{236}Pu or ^{242}Pu) by the analyte isotopes ^{239}Pu and ^{240}Pu, which was an infrequent occurrence [13].

EDTA or DTPA 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 [14].

5.3.1.2 Plutonium Reporting Levels, Minimum Detectable Activities, 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. Starting in approximately 1990, urine results were not normalized to a 24-hr sample. Instead, the results are dpm/sample regardless of the sample volume [15].

The minimum detectable activity (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 (RFETS 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 (RFETS 1998b) for any laboratory that processed the sample, although the required MDA was not consistently achieved by the onsite laboratory [16]. 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 [17].

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. The reported value was the 2-sigma standard error and included only uncertainties of counting statistics that were adjusted by the sample-specific recovery. Starting in approximately 1986, contributions from other sources of uncertainty were included, and the reported value was the 1-sigma standard error [18]. 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 [19].

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 [20].

The plutonium-to-americium alpha activity ratio (^{239,240}Pu dpm/24-hr sample divided by ^{241}Am dpm/24-hr sample) for paired plutonium and americium urine results provides a credibility check. An alpha activity ratio less than 2 (corresponding to a parts-per-million value for ^{241}Am of 10,000 or greater) is not credible unless the worker was involved with (1) separated ^{241}Am (Line 1 in Building 771), (2) the molten salt process in Building 776, (3) research and development projects involving pure americium, (4) material from the ZPPR project, or (5) waste identified for those operations [21].

Dose reconstructors should use the plutonium urine data instead of the ^{241}Am urine data to assess intakes of WG plutonium [22]. 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 Activities, 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 [23]. 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 [24].

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. It is favorable to claimants to assume that the result is all EU [25].

5.3.3.1.2 Reporting Levels, Minimum Detectable Activities, 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). It is undetermined when urinalysis for EU was stopped at RFP, although the stoppage probably occurred in the early 1970s [26].

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 trioxyl phosphene 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 (RFETS 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 [27].

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. Rather, it is favorable to claimants to assume that the reported urine result pertains only to ^{238}U and to determine additional intakes for the other uranium isotopes [28]. 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 Activities, 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 [29]. 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	dpm/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.

Table 5-10. Median MDAs for DU from May 1964 to the present.[a,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
1993–present	0.1

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.

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

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 [30]. 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 Activities, 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 probably coded as G in the urine data reports [31].)

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 should be considered to be upper bounds because of the nonspecificity of the analysis [32].

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

Uncertainties for the gross alpha 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). This uncertainty does not include the effect of interferences, which is a significant issue for a nonspecific analysis like gross alpha measurement [33].

5.3.5 Tritium

5.3.5.1 Pre-1973 Unmonitored Tritium Exposure

A pre-1973 tritium exposure method was developed based on measurement results provided in a Rocky Flats Area Office (RFAO) report issued subsequent to a tritium release in one of the Rocky Flats production buildings on August 30, 1974 (AEC ca. 1974). The information contained 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 prior to, during, and after the release. Several factors support the use of these data as surrogates for bounding the tritium environment at Rocky Flats prior to 1973:

1. Background tritium levels immediately prior to the incident described in the RFAO report, 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 prior to the 1974 incident are consistent with internal radiation doses from tritium of well under 1 mrem annually. They are dosimetrically insignificant in this sense.

2. The quantity of tritium released (1.5 Ci) was significantly less than that released in 1973, and is probably more typical of potential undocumented releases in work areas – particularly those resulting from opening contaminated shipping containers.

   The 1974 1.5-Ci tritium release is the only documented release from a shipping container in the Rocky Flats workplace. It is taken to be typical since there are no other such documented releases to use in forming the model. There is documented concern about tritium releases, as shown in the following quote from the ChemRisk report (ChemRisk 1994, pdf 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 Rocky Flats at that time were only semi-quantitative for indicating the presence of tritium; NIOSH has captured no records of these results.

   Because NIOSH has only identified six documented releases from 1968-1974 (an average of 1 per year), the application of a daily release would be a significant/bounding overestimate of the number of RFP tritium releases.

3. Tritium was released to the workplace environment, and not in a glovebox.

4. The release involved elemental tritium (HT, T2), and not tritium oxide (HTO).

5. The tritium was released from a contaminated shipping container which was procured by Rocky Flats in 1970 and can be taken as representative of shipping containers in use prior to 1973.

As stated in Item 2, the 1974 1.5-Ci tritium release is the only documented release from a shipping container in the Rocky Flats workplace. It is taken to be typical since there are no other such documented releases to use in forming the model. There is documented concern about such releases, as shown in the following quote from the ChemRisk report (ChemRisk 1994, pdf 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 Rocky Flats at that time were only semi-quantitative for indicating the presence of tritium; NIOSH has captured no records of these results.

Because NIOSH has only identified six documented releases from 1968-1974 (an average of 1 per year), the application of a daily release would be a significant/bounding overestimate of the number of RFP tritium releases.

6. The incident occurred close enough in time to the 1973 tritium release that work practices and controls were likely more similar to those prior to 1973 than to those even a year or two later, as procedures and controls evolved with greater sensitivity to the potential for tritium contamination.

An assessment of the 1974 1.5-Ci tritium release from a contaminated shipping container on August 30, 1974 was made based on the data in an RFP report (AEC ca. 1974). Specific urine sample collection dates were not included in the report, but data were matched to two claims in the NIOSH-Office of Compensation Analysis and Support Claims Tracking System (NOCTS), which reported a collection date of September 5, 1974. An assessment of this data was performed using an intake date of August 30, 1974, and the largest reported result collected after the incident (36,320 pCi/L). This resulted in a dose of about 0.15 mrem. Assuming one incident per workday at 0.15 mrem for 250 workdays per year results in an annual dose of 37.5 mrem/yr for the pre-1973 period. This should be assigned to all unmonitored RFP radiation workers.

5.3.5.2 1973 Tritium Release Exposure Method

The report, *Investigation of the Tritium Release Occurrence at the Rocky Flats Plant* (AEC 1973), describes a 1973 incident that prompted the site to sample a number of workers for tritium exposure (examples include ORAUT 2012a, 2012b, 2012c). A shipment of scrap plutonium from the Lawrence Livermore National Laboratory (LLNL) was discovered to have been contaminated with tritium. This material was processed at the RFP from April 9 to 25, 1973, in Building 779A, Room 154. 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 employees who worked in areas in which the contaminated scrap was processed. The waste stream from the processing of this material was also contaminated, which resulted in the potential for intakes of tritium at later dates. Therefore, all employees who were involved in the processing of wastes from this scrap were also included in the urinalysis program. The collection of samples from a tritium-contaminated water bubbler on September 19 and September 25, 1973, were also identified as possible sources of intakes.

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 for spectral shift could lead to abnormally high readings. Nineteen employees were initially identified as

having elevated tritium levels in their urine. These samples were distilled and reanalyzed. This recheck found fourteen of these employees were below the 10,000-pCi/L action level at the site. The five most-exposed individuals were identified, and details of their potential exposures, including bioassay results, are included in the investigation report. One of these five individuals is in NOCTS. The results of the five workers who exceeded the 10,000 pCi/L action level were reviewed by NIOSH.

Exhibit 14 of the report contains a section on Personnel Exposure Data. The following is an excerpt (AEC 1973):

> *SAMPLING PROTOCOL*
>
> *Dow began by sampling urine from all employees who were thought to have had the best chance of being exposed to tritium. As of October 15, 1973, about 250 employees have been tested. Dow is continuing to trace leads to other possible exposure and will sample them as they are found. Dow intends to sample many employees who have had only a remote chance of coming in contact with tritium. Dow also tests the urine of any employee who requests this whether or not they are candidates for exposure.*
>
> *ACTION LEVELS*
>
> *An "action level" of 10,000 pCi/l was tentatively chosen for resampling. This level was chosen for several reasons such as:*
>
> *1. An article by Fitzsimmons indicated that people wearing tritiated watches could excrete levels of 10,000 pCi/l.*
>
> *2. A calculation of worst possible circumstances indicate that an employee would have to exceed levels of 23,000 pCi/l before any permissible yearly levels of whole body radiation would be exceeded.*
>
> *3. The sample load was such that Dow could handle resampling only a limited number of employees on a frequent basis. It turned out that a relatively small number were over 10,000 pCi/l, but a large fraction were in the 5,000 and 10,000 pCi/l range.*
>
> *4. Without predistilling the urine samples the counting efficiency drops to about 3% and the corrections made for spectral shift can lead to abnormally high reading.*
>
> *5. With a large sample load, counting time devoted to each sample must be restricted so that 10,000 pCi/l might be considered lowest detection limit available under the present circumstances. All samples above 10,000 pCi/l are redone by counting the distillate of the original sample.*

RFP identified five workers with tritium urinalysis results above the action level of 10,000 pCi/L. Results from these five workers are reviewed here. Fourteen other workers had results initially above 10,000 pCi/L, but these fell below this level on recount (as noted above, the distillates of the original samples were counted, which offered better counting statistics during recount).

The document contains information, including tritium bioassay results and brief work histories, about the five workers with the largest tritium sample results. This information was used to assess the doses to the affected workers and is displayed in italics in the sections below. All five cases had initial samples that were not distilled, with one to five later samples that were distilled. In general, the undistilled and distilled sample results tended not to agree with the distilled samples, which yielded

lower values. This is to be expected in light of item 4 in the action levels discussion above. The predistilled results were used in the development of this analysis because there were more results available and they yielded doses that are favorable to claimants. The following assumptions were used in this assessment:

- Equal weight of all samples (measurement error the same for all samples),
- Only predistilled samples for fits,
- Tritium in the form of tritiated water vapor (HTO),
- The Integrated Modules for Bioassay Analysis (IMBA) model for inorganic tritium, as described in *Guidance on Use of IMBA Software for DOE Safety Applications* (DOE 2006),
- Injection intake (for modeling with IMBA), and
- Intake dates based on worker information and examination of fit to urine sample results.

The five workers who had the largest tritium urinalysis results are assessed below. Text in italics indicates an excerpt from the incident report.

Case A

> *Case A worked in Room* [location redacted] *from* [date range redacted]*.*
>
> *He was involved in the hydrating* [sic: likely hydriding] *and processing of the parts in question from* [date range redacted]*, along with Cases* [case identifiers redacted]*. He was not involved in any of the following special projects:*
>
> *a.* [date, special project name redacted]
>
> *b.* [date, special project name redacted]
>
> *c.* [date, special project name redacted]
>
> *He was involved in taking samples from a tritium-contaminated* [device redacted] *on* [dates redacted]*. On* [date redacted]*, this was done without a* [item redacted]*.*
>
> *From this history, it would appear the most likely exposure occurred on* [dates redacted]*. If an exposure had occurred between* [date range redacted]*, it is likely that both Cases* [case identifiers redacted] *would have been exposed to the same source, and subsequently, excreted the same quantities of tritium.*

The RFP document also states:

> *In Case A, a history of his work assignment and his urine results for the first two weeks indicate that he sustained a recent exposure. At the present time he is excreting tritium with an elimination half life of less than 10 days. According to Sanders and Snyder, this is the pattern of elimination from an exposure up to 90-days post exposure.*

The statement that Case A's intake appears to be recent agrees with current models for HTO intakes. If an intake on [date redacted] is assumed, a very poor fit to the data is achieved.

Based on the worker's history and the bioassay result pattern, an acute intake was assumed on September 19, 1973. Using the results of samples from September 25 to October 4, and applying a uniform error to each of the samples, the intake is 38.7 µCi. The corresponding dose is 2.6 mrem. These samples are presumed to be predistilled because later samples from October 5 to 12 are labeled as "distilled." This yields a very good fit to the predistilled results.

Case B

> *He has worked in* [location redacted] *since* [date redacted]. *He was in the room when* [action redacted].

Assuming a chronic intake from July 1 through September 25 (date of first urine sample) yields an intake rate of 0.33 µCi/d (for a total intake of 28.1 µCi) and provides a reasonable fit to the results. The dose is 1.90 mrem.

Assumption of an acute intake on September 19 (date of the first bubbler sample) yields an intake of 7.28 µCi. This fit is almost identical to the first scenario.

A single acute intake on the first day in the area (July 1) yields an intake of 720 µCi and a dose of 49 mrem.

The single acute intake on July 1 does not provide a good fit to the later predistilled results. The first two scenarios (chronic intake from July 1 through September 25, and acute intake on September 19) provide similar fits that reasonably follow the pattern of the predistilled samples. The chronic intake yields a larger intake so it is used for the best estimate.

Case C

> *He worked in* [location redacted] *since* [date redacted]. *He was not in the room when* [action redacted].

Because the worker did not start in the area until August 27, an acute intake was assumed on this date. Using only the predistilled sample results, the intake is 21.3 µCi with a dose of 1.4 mrem.

If a chronic intake is assumed to have started on his first day of potential exposure (August 27) and continued until the date of his first sample (September 25), the resulting intake is 0.24 µCi/d for a total intake of 7.08 µCi.

The two fits are very similar, so the acute intake is selected as the best fit because it results in a dose that is more favorable to the claimant.

Case D

> *He worked in* [location redacted], *between* [date range redacted]. *He has not been exposed to tritium since* [date redacted].

Case D submitted samples on only three days, although there are two results on two of those days. In one instance, one of the samples was distilled; on the other day, there is a note stating "repeated with sample channel ratio." On the latter day, the results differ by a factor of almost 2; the larger of these results is assumed to be the predistilled analysis and is used for the intake assessment. An assumed chronic intake from April 10 through 25 (last date of incident) yields an intake of 71.2 µCi/d for a total intake of 1,070 µCi. The resulting dose is 72 mrem.

A chronic intake from April 10 to June 15 yields an intake rate of 8.84 µCi/d for a total intake of 581 µCi (39 mrem).

Because there are few samples and the results follow no specific pattern, there is little difference between the fits. Therefore, the acute intake is assigned because it yields the larger dose.

Case H

He came in contact with the possible source of tritium on [date redacted].

No other information is included in the report. The conclusion in the report is:

> *It is expected that, as a result of a review of his work history and urinalysis data, a dose assignment of less than 3 rem will be made.*

However, no follow-up information is available. Because the only available information indicates that an intake would have occurred on April 6, an acute intake was modeled. The resulting intake is 1,240 µCi with a dose of 84 mrem.

The best estimates for the five reviewed cases are summarized in the Table 5-12 below.

Table 5-12. Summary of RFP tritium dose estimates.

Case	Intake (µCi)	Dose (mrem)
A	38.7	2.6
B	28.1	1.9
C	21.3	1.4
D	1070	72
H	1240	84

The tritium contamination was associated with plutonium scrap material. Therefore, the largest assessed dose of 84 mrem should be assigned to all individuals who were monitored for plutonium in 1973.

5.3.5.3 Post-1973 Unmonitored Tritium Exposure

For the assessment of tritium exposures at RFP after 1973, an analysis of NOCTS data from 1974 and 1975 was performed. There are 38 individuals with tritium data in 1974 and 37 in 1975. ORAUT-OTIB-0075, *Use of Claimant Datasets for Coworker Modeling*, provides justification and guidance (ORAUT 2009).

When assessing tritium intakes for most sites, it is assumed that intake potential exists only during tritium bioassay monitoring because monitoring is cheap, easy, and requires only spot samples, and therefore presents less of a burden than other forms of bioassay on both the employer and the employee. Because tritium was not of primary concern at RFP and was present only as a potential contaminant on equipment, a particular individual was not placed on a routine sampling program. Instead, a program was established in which one-tenth of the collected urine samples for plutonium analysis were also analyzed for tritium content (Bowman 1974) as well as the collection of samples when there was a particular concern. Samples available in NOCTS for these two years indicate that analyses were performed throughout the year, with most individuals being sampled only once.

For the coworker 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 coworker study for 1974 to 1975 yielded doses of 0 mrem for everyone.

For the years after 1975, there are 11 or fewer individuals in NOCTS with tritium data; this is insufficient for performing a coworker study. Results for these years are consistent with those from

the previous years and show a general decreasing trend. The intake rate from the 1974 to 1975 coworker study (i.e., 0 mrem) applies to these years. Therefore, no additional unmonitored tritium dose should be assigned after 1973.

5.3.5.4 Reporting Levels, Minimum Detectable Activities, 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 [34]. 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 [35]. The MDA for tritium should be assumed to be 600 pCi/L for all tritium bioassay (RFETS 1998c, p. 176).

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 Activity 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 <u>Americium and Plutonium</u>

5.4.1.1 Methods, Units, Isotopes, and Interferences

Before April 1997, 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 [36]. In addition, the activity of ^{241}Am was stated as a fraction of

the MPLB, which was 14.7 nCi (Falk 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. This interference is most troublesome to dose reconstruction for workers with residual lung depositions of plutonium and americium who subsequently worked in DU operations [37]. The second interference is the contribution of count from ^{241}Am not in the lungs (e.g., contributions from contamination on the skin, from material being cleared from the upper respiratory system, or from ingested material). A positive detection of ^{241}Am did not necessarily indicate an intake (especially one that resulted in a deposition to the alveolar-interstitial region of the lungs) of the plutonium-americium mixture, especially for a lung count in response to an incident [38].

5.4.1.2 Reporting Levels, Minimum Detectable Activities, and Uncertainties

Reporting levels are not easily defined because quantification was preceded by verification counts and professional judgments. In addition, before 1974, the practice was not to quantify a positive detection of ^{241}Am unless the deposition could be associated with a known incident with a known ppm ^{241}Am. Affected workers were classified as positive unknowns or some variation. Starting in 1974, the practice was changed to quantify the plutonium depositions for positive unknowns by assuming a default value of 1,000 ppm ^{241}Am on the date of the most probable intake or on the date of the first positive lung count. The ppm ^{241}Am was then calculated for the date of the lung count to account for the ingrowth of ^{241}Am from the nuclear transformation of ^{241}Pu and the radioactive decay of the initial ^{241}Am [39].

In general, this quantification was not applied retroactively to earlier positive lung counts. Once a lung deposition of plutonium had been quantified for a worker, the deposition continued to be quantified for all subsequent lung counts (except screening counts for new intakes), regardless of the result of the subsequent lung count (including negative values), until each of the last three results was less than the decision level for the count and the average of the last three results was within 1 standard deviation of 0.00 nCi plutonium [40].

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 [41]. The decision level for 1995 and later was calculated by ABACOS-Plus for a probability of a Type I (false positive) error of 5% (RFETS 2000b, p. 90). The decision level was used as a reporting level from 1995 to early 1997.

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.

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

Period[a]	Detector system	Index	MDA (nCi) for ^{241}Am[b]			
			Minimum system		Standard system	
			Half time	Full time	Half time	Full time
1964–1968	NaI(Tl) 4 × 4	0.90	1.7	1.5	1.3	1.2
		1.35	2.8	2.5	2.1	1.9
		1.80	4.6	4.1	3.5	3.2
1969→	NaI(Tl) 4 × 4	0.90	–	–	0.8	0.76
		1.35	–	–	1.3	1.3
		1.80	–	–	2.2	2
1973→	Phoswich	0.90	–	–	1.2	1.2
		1.35	–	–	2.0	2.
		1.80	–	–	3.3	3.2
1976–1978	Ortec Arrays (High-purity Ge)	0.90	0.26	0.18	0.2	0.14
		1.35	0.48	0.32	0.37	0.25
		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
		1.35	0.37	0.25	0.28	0.19
		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
		1.35	0.4	0.27	0.31	0.21
		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
		1.35	0.31	0.21	0.24	0.16
		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
		1.35	0.4	0.28	0.31	0.21
		1.80	0.74	0.5	0.57	0.39
1985→	PGT Organ Pipe Ge Detectors	0.90	–	–	0.15	0.11
		1.35	–	–	0.26	0.18
		1.80	–	–	0.46	0.32
1991→	EG&G Organ Pipe Ge Detectors	0.90	–	–	0.14	0.1
		1.35	–	–	0.26	0.18
		1.80	–	–	0.48	0.33
1995→	Ortec 2 Organ Pipe Ge Detectors	0.90	–	–	–	0.14
		1.35	–	–	–	0.3
		1.80	–	–	–	0.6

a. In overlapping years the more favorable MDAs should be assumed.
b. – = not applicable.

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 [42]. 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 1995 and later. (The MDA values are reported on report forms from 1995 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 if height and weight (or index) data for the worker are not available [43]. The DR should assume the MDA is twice the decision level for 1995 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. Dose reconstructors need to establish the date of the intake and the initial ppm ^{241}Am. If that information is not apparent in the

available records, an approach that is favorable to claimants is to assume the initial ppm ^{241}Am to be 100 [44].

The assumption of the intake date is not straightforward and should balance maximizing the plutonium lung deposition (intake date is close to the date of the lung count) and maximizing the accrued lung dose (intake date is far from the date of the lung count). In addition, the choice of intake date for the lung count data should be coordinated with that for the associated urine data [45].

Dose reconstructors must choose the value of the initial mass fraction of ^{241}Pu. At the RFP lung counter, 0.005 was historically used as the initial mass fraction of ^{241}Pu and is a realistic choice for intakes that occurred in the 1950s to June 1976. The fraction 0.0036, based on the isotopic composition for RFP stream plutonium in the mid-1970s, should be used for intakes that occurred from July 1976 to 1989. For intakes after 1989, the initial fraction of ^{241}Pu should be reduced to account for the aging (radioactive decay) of the ^{241}Pu [46].

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 [47]. With the advent of ABACOS-Plus in 1995, the percent error at 1 standard deviation was reported for all identified nuclides. 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 (RFETS 2000b, p. 89).

The major uncertainty for the calculation of the plutonium lung deposition is the ppm ^{241}Am in the plutonium in the lungs at the time of the lung count. Factors in the uncertainty are the intake date, the value of the initial ppm ^{241}Am, the initial fraction of ^{241}Pu, and the degree of association of the americium with the plutonium while in the lungs. An underlying assumption is that the americium remains associated with the plutonium particles in the lungs until the particles are dissolved or removed from the lungs. The degree of validity of this assumption has not been determined [48].

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 early 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 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).

The interference is ^{238}U in natural uranium. Unless there is a reported activity for ^{234}U that is approximately equal to that reported for ^{238}U, dose reconstructors should use the assumption (favorable to claimants) that the ^{238}U activity is all from occupational exposure to DU [49].

5.4.2.2 Reporting Levels, Minimum Detectable Activities, 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.

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 [50]. (As noted in Section 5.4.1.2 for americium and plutonium, MDA values are reported on forms for 1995 and later, but are not worker-specific. Dose reconstructors should disregard these MDA values.) The DR should assume the MDA is twice the decision level for 1995 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 [51]. 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.

The standard deviation of the net count rate is reported through 1995 but includes only the contribution of counting statistics. To estimate the uncertainty of a ^{238}U or DU activity from the net count rate, dose reconstructors can divide the worker-specific MDA by 3.3 [52].

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.

Wound count information is largely irrelevant to dose reconstruction [53]. The relevant items are the urinalysis data, the identification of the mode and date of intake, and whether there was residual plutonium at the wound site. Guidance on assessing wound intakes is provided in Technical Information Bulletin: Guidance on Wound Modeling for Internal Dose Reconstruction (ORAUT 2005)

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 (RFETS 1998d, p. 62).

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

- 20 dpm/sample, for (gross alpha, liquid scintillation) routine nasal samples;
- 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 (L_c), 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 [54]. 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) [55]. 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 [56].

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 [57]. 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 [58].

- 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 (L_c) 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 [59].

- 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. If the batch failed, every sample in the batch was conditionally failed pending further evaluation [60].

- 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 [61].

- 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 [62].
- 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 [63]. 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 (L_c) 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 [64].

- The VF is the volume fraction, the fraction of the sample that was analyzed. A VF 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 [65].

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 (40 MLT means 40 minutes live time) [66].

- 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) [67].

- 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. Those data have little dosimetric use [68].

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 data set 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. A ratio of 1.20 or greater indicated probable detection of americium [69].
- 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. Although that information was captured occasionally, the data were not used because of the instability of the predicted background cpm [70].

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 Am, or the lung burden. Rather, it reports the cutoff, which is the decision level, and Normal if the DIFFERENCE is less than the cutoff [71].

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. If the results were not normal, the subject would have been recounted with a germanium detector system [72]. 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 [73]. 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) [74].

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, an obsolete concept since 1989 [75].

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 [76].

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 basis of the decision level for the plutonium isotopes is not obvious, but was probably the decision level for detecting the L X-rays. In any case, this decision level value does not apply and should be disregarded for the plutonium isotopes [77]. 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 and the value of the initial tabulated ppm ^{241}Am generally on one of the early lung count reports [78].

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. Otherwise, the date of the lung count should be used as the intake date [79].

Dose reconstructors should be aware that the lung counter detectors were also used for wound counts (Berger 1988b; RFETS 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 [80].

5.7 ATTRIBUTIONS AND ANNOTATIONS

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

Much of the information in this TBD, including the Attachments, was written by Roger B. Falk and is based on his insights, recollections, research and development activities, and administration in the radiation dosimetry and health effects programs at the Rocky Flats Plant.

[1] Falk, Roger B. Oak Ridge Associated Universities (ORAU) Team. Senior Life Scientist. June 2006.
The statements of the primary types of intakes and bioassay data are based on the observations by the author during his work at RFP in the internal dosimetry and health effects programs.

[2] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This statement is based on the observations of such notations on incident and lung count reports related to ZPPR materials.

[3] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
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.

[4] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Oxides of plutonium metal (air-oxidized and fire-oxidized) are classified as type S and most other plutonium compounds as type M by the ICRP (ICRP 1994a). In any case, dose reconstructors should use the solubility class that is most favorable to claimants.

[5] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Retention of plutonium in the lungs of workers exposed in the 1965 plutonium fire was observed to be more avid than would be predicted by the default ICRP type S model (ICRP 1994b), based on lung counts performed as part of the Former Radiation Worker Medical Surveillance Program at RFP, 28 to 38 years after intake.

[6] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Oxides (air-oxidized and fire-oxidized) of plutonium metal are classified as type S and most other plutonium compounds as type M by the ICRP (ICRP 1994a). In any case, dose reconstructors should use the absorption type that is most favorable to claimants.

[7] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The recommendation is favorable to claimants when intakes are assessed from airborne plutonium data and is essentially neutral when intakes are assessed from urine or lung count data.

[8] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The source of the americium is only from the decay of ^{241}Pu. No other americium isotopes are involved.

[9] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Spot urine samples for plutonium were rarely requested and then usually associated with a significant incident, especially an incident with chelation (DTPA) treatment follow-up. Such exceptions should be easily discernible in the documentation of the incident in the worker's health physics file, especially starting in 1990, the period of interest for this recommendation. In addition, many of the urine sample result reports include the volume of the sample [see Figures C-8 to C-12. (Although some of these examples are for uranium analytes, the format of the report is the same for plutonium analytes.)]. These reported volumes can be used to normalize the result to a 24-hr sample when appropriate.

[10] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The original 2003 version of this sentence was stated incorrectly. The factor of 1.0264, when multiplied by the intake assessed from ^{239,240}Pu urine data, would yield the intake for ^{238,239,240}Pu, not the ^{238}Pu component of the intake. In addition, the factor was based on a slightly different isotopic composition from that stated in Table 5-1. The ^{238}Pu component of the intake is obtained correctly by multiplying the intake assessed from ^{239,240}Pu urine data by 0.0235, a value 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.

[11] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Interferences that add to the value of the analyte are always favorable to claimants. Therefore, the recommendation was made to use the results as found in the record unless dose reconstructors have generic instructions, outside the purview of this TBD, to do otherwise.

[12] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
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.

[13] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This statement, based on informal discussions in the 1980s and early 1990s with [name redacted], the [position redacted] at Rocky Flats starting in [date redacted], was included here for completeness.

[14] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The recommendation for dose reconstructors to be wary of, and generally not use, urine data that was flagged with Code 1 is based on good science and common sense. Although chelation-enhanced urine data might be favorable to claimants, use of such data without modification in standard models that are based on unenhanced data is not scientifically sound. It is also not sound to use data that did not pass quality standards in real time.

[15] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The information in this paragraph is based on observations and deductions of the author from review of original urine data logs and individual urine data reports in preparation of this TBD section and Attachment A, as well as from personal involvement in the development of improved urinalysis reporting protocols at RFP in the 1980s and early 1990s.

[16] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This statement is based on an undocumented conversation with [name redacted], [position redacted] at RFP in the summer of 2003.

[17] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The recommendation to use the values in Table 5-6 seemed to be the only viable option.

[18] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This information is based on the author's recollections of the implementation of these upgrades at RFP.

[19] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This statement is based on the observation that the same reporting format used for plutonium results was used for americium results.

[20] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This information is based on informal conversations in the early 1990s with [name redacted], the [position redacted] at Rocky Flats starting in [date redacted].

[21] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This credibility check is presented to dose reconstructors as optional guidance and does not preclude the use of americium urine data for dose reconstructions if deemed appropriate, even if the data do not pass this credibility test. Note also that the maximum ingrowth of americium in virgin WG plutonium (with 0.5% by weight ^{241}Pu) is less than 5,000 ppm. Plutonium with 10,000 ppm ^{241}Am or greater would be credible only for a process that enhanced the americium concentrations, such as those processes listed.

[22] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This recommendation was based on the plutonium-to-americium activity ratio, which is considerably greater than 2 for WG plutonium, and the problem of the thorium interferences in americium urinalyses. These two factors make the plutonium urine data set the better choice to determine plutonium intakes.

[23] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This recommendation extrapolates a value favorable to claimants forward to the next point in time for which the reporting level was determined.

[24] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This statement is based on the observation that the same reporting format used for plutonium results was used for americium results.

[25] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
If there are interferences that contribute to the magnitude of the results, considering those interferences as EU results in a higher than actual outcome and is therefore favorable to claimants.

[26] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
It is not clear in the urine data logs for 1964 to 1971 which electroplated uranium samples were for EU and which were for DU. EU operations were discontinued in this period although some urine sampling for EU could have occurred for workers involved in decontamination activities.

[27] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The statements about the units of the reported urine data are based on observations of numerous urine data reports.

[28] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This recommendation implies dividing the intake assessed from the assumed ^{238}U urine data by 0.89 (the ^{238}U fraction of total DU activity) to calculate the total DU intake. This approach is favorable to claimants by about 12% if the activities of the ^{234}U and ^{235}U were actually included in the reported uranium urine results in the 1980s.

[29] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This recommendation extrapolates the earlier practice for a period when the actual practice is not known. This approach is favorable to claimants if the earlier values would result in a higher intake assessment, as in this case.

[30] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
No documented date was found about when gross alpha analyses were discontinued. The year 1973 was estimated following a review of HSDS urinalysis reports for a sampling of workers previously sampled for gross alpha, with a finding of no analysis code G after 1972.

[31] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Urinalysis code G was observed in HSDS urinalysis reports to 1972. Code G correlates with the gross alpha B_2 analysis code on the Urinalysis Record Card (see, for example, Figure C-2).

[32] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The term “favorable to claimants” is used interchangeably with the “upper bounds.”

[33] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The effect of interferences is not included in the estimate of the standard deviation because it is not really a random variable but rather an intermittent bias of unknown magnitude.

[34] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Liquid scintillation was used as the counting method for tritium in the 1980s, based on the author's personal observations. No documentation has been noted about the earlier systems at RFP, but it seems reasonable to consider that the same method was used in the 1970s.

[35] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Numerous tritium results at these levels have been observed by the author in the worker's health physics files at the cited levels.

[36] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
These modifications were implemented by the author in 1973 at RFP. Figure C-18 is an example of the implementation of these modifications.

[37] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This interference, though troublesome in real time, is favorable to claimants whose americium lung count results were enhanced by count from ^{234}Th.

[38] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Interferences, especially contamination on the worker's chest, occasionally caused a false positive lung count. This statement was intended to alert dose reconstructors to this possibility. The sentence is modified in recognition of the fact that an intake could have occurred without resulting in a deposition in the alveolar-interstitial region of the lung.

[39] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
These modifications were implemented by the author in 1974 at RFP.

[40] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This modification was implemented by the author in the early 1980s at RFP.

[41] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
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.

[42] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The worker-specific MDA for the americium in vivo measurement depends on the worker's index and the calibration factor *K* for that index for the detector system used for the worker's lung count. A relatively easy method is to normalize the MDA value for the 1.35 index by the ratios of the calibration factors (given in Attachment B for each detector system) for index 1.35, and the worker-specific index is:

$$MDA_{\text{worker}} = MDA_{1.35}\frac{K_{1.35}}{K_{\text{worker}}} \quad (5\text{-}1)$$

An easy method to derive the worker-specific MDA from values listed in Table 5-13 is to use a spreadsheet to plot the MDA values for the three indices, for the detector system of interest, and to determine the equation for an exponential trend line. This equation, in the form $y = \#.\#\#\#\# \, e^{\#.\#\#\#\# x}$, where *x* is the worker's index and *y* is the worker-specific MDA, can then be used to calculate the worker-specific MDA.

[43] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The recommendation to use the median value of the MDA is consistent with the generic approach of the program. Except for some workers with lung counts only in the 1960s, this situation is expected to be rare.

[44] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The recommendation to use an initial value of 100 ppm Am, if the actual value for an intake is not documented, is based on freshly purified plutonium (within 0 to 5 months depending on the efficiency of the purification process).

[45] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This guidance is based on the consideration that significant intakes of plutonium at RFP were acute intakes, albeit sometimes a set of intermittent acute intakes. After implementation of the body counter in 1965 and as the sensitivity of the system improved, the assignment of the intake date to newly detected depositions, but not from a new intake, was problematic. Dose reconstructors might have generic instructions, especially for efficiency methods, for assigning the intake scenario.

[46] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This guidance is presented to support the calculation of the ingrowth of ^{241}Am after the date of an acute intake or after the start of a chronic intake. After the end of plutonium production activities in 1989, the initial fraction of ^{241}Pu in RFP plutonium was a decreasing variable based on the age of the plutonium since blending.

[47] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
These modifications were implemented by the author in 1976 and 1981 at RFP. Figures C-20 and C-21 are examples of the implementation of these modifications.

[48] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The validity of this assumption, listed as an uncertainty, seems to be supported by observations that americium lung count measurements for many RFP workers with confirmed lung depositions of plutonium-americium mixtures have remained relatively constant or have slightly increased at decades after the initial short-term clearance period of several years. Recent U.S. Transuranium and Uranium Registries autopsy data for RFP cases also indicate the retention of americium in the lungs that is consistent with this assumption.

[49] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
No documented date was found about when gross alpha analyses were discontinued. The year 1973 was estimated after a review of HSDS urinalysis reports for a sampling of workers previously sampled for gross alpha, with a finding of no analysis code G after 1972.

[50] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Dividing the ^{238}U MDA by 0.89 accounts for the contribution to the DU MDA from activities of the other uranium isotopes.

[51] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
If a super-equilibrium 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.

[52] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This method suggested to estimate the uncertainty of the activity from its MDA is the same as the method suggested in Sections 5.3.1.2, 5.3.3.1.2, and 5.3.4.2 and is equally applicable.

[53] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Because dose reconstructors are likely to find numerous wound count reports in files of workers assigned to plutonium areas, this statement helps to focus the attention of dose reconstructors on the most relevant data for quantifying internal doses to organs. The relevant data are cited in the next sentence. The actual wound count and contamination data might be relevant if the cancer site coincided with the site of the wound, an occurrence not yet noted by the author.

[54] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The interpretations are those of the author, either gleaned from using the data as an internal dosimetrist at RFP or as the designer and implementer of the reports as part of the technical staff supporting RFP internal dosimetry programs.

[55] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
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.

[56] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
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.

[57] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Exceptions to this practice have been observed in the HSDS urinalysis reports for some workers with documented chelation therapy, especially for americium results from analysis of a urine sample also analyzed for plutonium (and the plutonium result was correctly coded with Code 1).

[58] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This circumstance is evident in the example reports in Figures C-2, C-3, and especially C-5.

[59] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This is presented for information only. The decision of whether to disqualify the result was the call of the laboratory quality assurance officer who reviewed the data and signed the report. The Batch Val code V is the primary indicator of a valid result.

[60] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This is presented for information only. In practice, the batch evaluation review would have occurred before the release of the Analytical Report. The Batch Val code V is the primary indicator of a valid result.

[61] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
It is prudent not to use a result that was invalidated based on failure to meet quality standards.

[62] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The original 2003 version of this sentence was stated incorrectly. The factor of 1.0264, when multiplied by the intake assessed from ^{239,240}Pu urine data, would yield the intake for ^{238,239,240}Pu, not the ^{238}Pu component of the intake. In addition, the factor was based on a slightly different isotopic composition from that stated in Table 5-1. The ^{238}Pu component of

the intake is obtained correctly by multiplying the intake assessed from ^{239,240}Pu urine data by 0.0235, a value 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.

[63] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
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 if such a situation occurs.

[64] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This is presented for information only. The time of the end of the excretion period is not critical for retrospective dose reconstructions. IMBA, which is used by Project dose reconstructors, has a default sample time of 12:00 a.m.

[65] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The units for fecal and nasal smear sample results are well known to the author from his experience as internal dosimetrist at RFP. In addition, the units are probably stated explicitly on other reports in the health physics files of affected workers.

[66] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The time of day of the count is not critical for retrospective dose reconstruction. Later lung count reports usually recorded the time of day in this field and noted the count time only if it was different from the standard count for the era, through the 1970s (see Attachment B for the standard count times). Electronically generated lung count reports, starting circa 1981, record the count time used for that count (see Figures C-21 to C-31). The count time would be useful to dose reconstructors mainly to calculate an MDA for a given lung count, if needed.

[67] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
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, f_2 = 1, and ε = 53 (from ICRP Publication 2, Table 5, and based on a relative biological effectiveness = 10).

[68] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The main reason the measurements obtained by the detector over the gut/liver/below-sternum area are not dosimetrically useful is that a calibration factor was not developed in real time to convert the signal to activity. It was a subjective measurement; i.e., was it normal or high?

[69] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This ratio was the subjective rule-of-thumb used by the author in real time at RFP as a supplemental method to discern possible low-level depositions of the plutonium-americium mixtures for lung counts performed with the NaI detector system.

[70] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The variability in the background in the L X-ray region of the spectrum prevented the establishment of a stable calibration factor for the direct measurement of plutonium via L X-rays. The author was directly involved in this effort.

[71] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
See, for example, Figure C-22.

[72] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This statement reflects the practice to perform a follow-up lung count with a better resolution detector system (i.e., a germanium detector system) when action levels for a count with a poorer resolution, NaI scintillation detector system, such as the phoswich detectors, were exceeded.

[73] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
See, for example, Figure C-26.

[74] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
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.

[75] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
DOE Order 5480.11, implemented in 1989, shifted the basis from the ICRP Publication 2 approach to control the dose to a critical organ to the ICRP Publication 26 and 30 approach of assessing the committed dose equivalent to organs from intakes (ICRP 1959, 1977, 1979; DOE 1988). The quality factor for alpha radiation was increased from 10 to 20.

[76] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This statement is the result of direct observation of information in Figure C-29.

[77] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
Not only is the plutonium decision level inoperative because the decision is based on the detection of americium, but also the decision level is never operative for follow-up measurements of a confirmed deposition – there is no decision to be made. A decision level is operative only if the null hypothesis is operative. The null hypothesis is not operative in this example.

[78] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
This information was provided just in case dose reconstructors want to determine the value of the parts per million of the ^{241}Am used in the calculation.

[79] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
ABACOS-Plus used the date of the count as the default intake date unless an intake date was specifically input for the count. This statement should not be interpreted as guidance to dose reconstructors to use that default date in dose reconstructions.

[80] Falk, Roger B. ORAU Team. Senior Life Scientist. June 2006.
The only exception to this statement found by the author 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.

[81] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
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 2011).

[82] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
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.

[83] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This information was discerned from comparing records in the urine data logs with entries on Urinalysis Record Cards.

[84] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The tolerance levels were noted as the Working MDL 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.

[85] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
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.

[86] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This statement was based on the examination of the urinalysis records of a number of workers affected by this practice.

[87] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
These reporting levels were discerned from the lowest values recorded in the americium urine data logs.

[88] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
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.

[89] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This is a summary of the observations of the recovery determined from the batch spike versus a standard recovery value, based on calculations to reproduce the result in the urine data logs. Additional discussions are provided for the analytes later in the document.

[90] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The adjustment of the volume in this manner could have occurred earlier. However, urine data logs for 1955 to 1959 were not found.

[91] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This information was provided by [name redacted], the [position redacted] at Rocky Flats starting in [date redacted], in an interview with the author in 1992.

[92] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
Notations indicating detectors with 40% efficiency started to appear in the urine data logs in August 1964.

[93] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This statement is based on the author's direct experience and on discussions with [name redacted], who was the [position redacted] at Rocky Flats starting in [date redacted] and also [position redacted] during the cited period.

[94] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The author was directly involved with making this change.

[95] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The author was directly involved with a committee of Radiological Health and Analytical Laboratory personnel in 1993 to implement these changes. Upon further review, the author observed that the count time was increased to 1,440 minutes (24 hours) rather than the stated 2,000 minutes. This correction is now made.

[96] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This information was provided by [name redacted], the [position redacted] at Rocky Flats starting in [date redacted], in an interview with the author in 1992; it was verified during examination of the urine data logs for the early 1960s.

[97] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
These values were based on observations made in the urine data logs concerning when the count results were converted to the activity of EU in the sample. Apparently, that decision was based on the count uncorrected by volume, for which the minimum reported activity was 20 dpm/24-hr sample. When a volume adjustment was made, higher minimum reported values up to 28 dpm/24-hr sample were observed.

[98] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
These values were obtained through calculations by the author to duplicate the results stated in the urine data logs.

[99] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The urine data logs for 1964 to 1971 do not distinguish explicitly which samples were for workers in EU areas versus those for workers in DU areas.

[100] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This information was discerned by the author from examination of the urine data logs for fluorimetric measurements.

[101] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This information was discerned by the author from examination of the urine data logs for electroplating measurements.

[102] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This information was provided by [name redacted], the [position redacted] at Rocky Flats starting in [date redacted], in an interview with the author in 1992.

[103] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
A better way to indicate the generosity of the nonspecificity of the gross alpha result if applied to a specific radionuclide is to use the term “favorable to claimants” rather than “upper bounds.”

[104] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
To have a coherent data set, only background count data for samples counted for 150 minutes were extracted from the urine data logs.

[105] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
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.

[106] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This exception was made for americium because the detector backgrounds for the 1950s did not apply.

[107] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This statement is a summary of the observations of the author during the review of the urine data logs.

[108] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This approach is consistent with the consideration stated in the fourth bullet in the subsection headed Assessment of MDA.

[109] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This subjective observation of the similarities of the recovery values in the preceding table was interesting to the author but was not used to determine any MDA value.

[110] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The distribution of volumes for routine 24-hour urine samples was determined from the volumes recorded in the urine data logs for gross alpha analyses for 1967 and 1971, a data set of 1,437 values. The author chose the gross alpha samples as the sample set least likely to include special samples that could have had an excretion period of less than 24 hours.

[111] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The distribution of volumes for routine 24-hour urine samples was determined from the volumes recorded in the urine data logs for gross alpha analyses for 1967 and 1971, a data set of 1,437 values. The author chose the gross alpha samples as the sample set least probably to include special samples that could have had an excretion period of less than 24 hours.

[112] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
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.

[113] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
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.

[114] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This information was discerned from the urinalysis data logs by the author.

[115] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The decision by the author to assess the MDA based on one aliquant was based on the observation of the data logs that the decision of detection for the overwhelming majority of the samples was based on only one aliquant. Occasionally, the decision was based on the average of two aliquants. Because the MDA for one aliquant is higher than that for two aliquants, this decision is consistent with the consideration stated at the beginning of the Assessment of MDA subsection, fourth bullet.

[116] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
This introduction summarizes the information presented in more detail in the body of this TBD. Most of this information is based on the direct experience of the author, who provided technical support to the operations and developments of the in vivo lung-counting systems at

RFP starting in 1970 and extending into the mid-1980s and also from 1989 to 1992. The author also provided second-level management of the dosimetry programs from 1986 to 1989.

[117] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The dates for the start of routine operations of the three counting rooms were determined from the references [Room A (Boss and Mann 1967) and Room C (Falk et al. 1979)] or from the author discerning the year that the room was first recorded on a worker's lung count report (Room B).

[118] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The information summarized in this section was discerned by the author from (1) notations on lung count reports for workers counted in that era, (2) a transition briefing from [name redacted], the [position redacted] providing technical support to the Rocky Flats [position redacted] from [date redacted] to [date redacted], and (3) reports cited in the reference section.

[119] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The information in this section describes the lung-counting program as found by the author in 1970. The transitions in 1969 were verified by the author in preparation of this report by observations of notations in worker lung count reports. The transitions did not generally take place exactly on January 1, 1969, but usually sometime in 1968, which was a transition year. However, for the purpose of determining MDAs, the old practice was considered to be extended through the year, and no credit was taken for the new practice until it was in effect for the entire year.

[120] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The date of the first entry in the logbook for routine ppm ^{241}Am determinations for incident samples was observed by the author to be January 3, 1969. This statement does not preclude earlier special ppm ^{241}Am determinations. Indeed, a special ppm ^{241}Am determination was made for the October 15, 1965, plutonium fire incident.

[121] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The information in this section describes the systems based on the direct involvement of the author and verified by the author from observations of the body count result reports as needed.

[122] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The information in this section describes the systems as observed by the author, either directly or indirectly.

[123] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The upgrades in this period were implemented by [name redacted] and [name redacted], technical staff supporting the [position redacted] program at Rocky Flats in the [date redacted] into the [date redacted], with discussions with the author, who at that time was a customer of the in vivo measurements program as the internal dosimetrist for the medical monitoring program for former radiation workers at RFP. This era is well documented in RFETS (2000b).

[124] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The count time, which is the duration of the lung count, was observed by the author from lung count reports for workers in this era.

[125] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
For this equation and for subsequent equations of the calibration factor *K*, the calibration factors were normalized to LLNL phantom [also called the Lawrence Livermore Torso Phantom and described in RFETS (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."

[126] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
See the subsection headed 1969 to 1976 in Section B.3. In addition, the information in this section describes the lung-counting program as found by the author in 1970. The transitions in 1969 were verified by the author in preparation of this report by observations of notations in worker lung count reports. The transitions did not generally take place exactly on January 1, 1969, but usually sometime in 1968, which was a transition year. However, for the purpose of determining MDAs, the old practice was considered to be extended through the year, and no credit was taken for the new practice until it was in effect for the entire year.

[127] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The author personally determined the resolution of the Ortec and Princeton Gamma Tech (PGT) detectors for the 59.5-keV photopeak of the ^{241}Am gamma and noted the degradation in the resolution. This was expected because the collection efficiency of the charge induced in the active part of the detector diminishes with an increasing volume of that active part (based on the author's recollection of solid-state physics).

[128] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The author personally determined the resolution of the Ortec and Princeton Gamma Tech (PGT) detectors for the 59.5-keV photopeak of the ^{241}Am gamma and noted the degradation in the resolution. This was expected because the collection efficiency of the charge induced in the active part of the detector diminishes with an increasing volume of that active part (based on the author's recollection of solid-state physics).

[129] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
See Table 5-1 in the body of this TBD. The value of 0.0049 had been historically used at the RFP body counter as the rounded value of 0.005.

[130] Falk, Roger B. ORAU Team. Senior Life Scientist. July 2006.
The method of determining the ppm ^{241}Am from samples representative of the plutonium mixtures involved in possible inhalation incidents, starting in 1969, involved the ratio of the L X-ray photopeaks and the ^{241}Am 59.5-keV photopeak as measured by a NaI(Tl) detector. The ppm ^{241}Am determined by this method was highly uncertain for values less than 100 ppm and greater than 10,000 ppm because of the counting statistics. Although one might consider zero ppm ^{241}Am to be the true lower bound for freshly purified plutonium, a zero value is not practical to use in Equation B-17 (i.e., division by zero is not allowed). The value of 100 ppm ^{241}Am is also supported by its rank at the 10th percentile in the low-to-high ranking of 442 values of the incident ppm recorded in the logbook for January 1969 to September 1972. The value also represents the ingrowth of americium in freshly purified plutonium (within 0 to 5 months, depending on the efficiency of the purification process).

[131] Arno, Matthew. ORAU Team. Dose Reconstructor. June 18, 2007.
Lognormal distributions typically provide the best fit to the available data and are a distribution suitable for input into the Interactive RadioEpidemiological Program (IREP).

[132] Arno, Matthew. ORAU Team. Dose Reconstructor. June 18, 2007.
The error associated with individual bioassay results is normally distributed because the dominant source of uncertainty is the counting statistics. Although the underlying group statistics are normally distributed, each result was treated as if it were normally distributed to match what is done for analysis of an individual's bioassay data and because the lognormal distribution of the data is addressed by analyzing both the 50th- and 84th-percentiles of the data.

[133] Arno, Matthew. ORAU Team. Dose Reconstructor. June 18, 2007.
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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. See *minimum detectable 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 activity (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 (WB) 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 ACTIVITY FOR URINALYSIS METHODS AT ROCKY FLATS

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MINIMUM DETECTABLE ACTIVITY FOR URINALYSIS METHODS AT ROCKY FLATS (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 located 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):

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

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 $\alpha = 0.05$ and $\beta = 0.05$, $k = 1.645$)
- s_0 = the standard deviation in the net count of a sample with no additional analyte:

$$s_0 = \sqrt{s_{B1}^2 + \frac{s_{B0}^2}{m^2}} \qquad \text{(A-2)}$$

- T = the standard counting time for the procedure

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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 [81]. 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 [82].

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

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 [83].

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

Analyte	Method code
DU	A, B_1 (starting 5/1/64)
EU	B_1
Plutonium	C_1, C_2
Gross alpha	B_2, B_3, D

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 exceptions. Table A-3 lists the values of the tolerance levels [84]:

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 [85].

For plutonium, the reporting and recording level was ≥0.2 dpm/24 hr for 1962 to April 6, 1970. 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 [86].

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 [87].

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

$$Activity\left(\text{dpm/24-hr sample}\right) = \frac{\left(C/T - B_{\text{Det}} - B_{\text{Blk}}\right)\left(V/A\right)}{\varepsilon R} \qquad \text{(A-3)}$$

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)

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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 [88]. 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). In addition, the recovery values from the spikes usually were not used to customize the standard value of *R* for samples in the batch [89].

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.

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

$$Mass\left(\text{µg/24-hr sample}\right) = \frac{S - B_{\text{Blk}}}{K} \qquad \text{(A-4)}$$

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.]

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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 [90].

1961 to 1962
Starting on December 13, 1961, a thenoyltrifluroacetone (TTA) extraction step was added to the carrier precipitation method to improve the specificity of the process to isolate plutonium [91]. 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 [92].

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 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 [93]. In 1990, the acceptable recovery range was changed to 0.35 to 1.10 [94]. 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

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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 beyond
Upgrades to procedures occurred in 1993 to achieve a process MDA less than or equal to 0.020 dpm/sample [95]. 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 beyond)
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 [96]. 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 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 [97].

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) [98].

EU operations were phased out at RFP from 1962 to 1963, although some workers were still monitored for EU intakes through 1971 [99].

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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 [100].

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) [101].

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 [102].

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. In either case, the results should be considered upper bounds because of the nonspecificity of the analysis [103].

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.

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- 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 for routine 24-hour urine samples.

Percentile	Volume (mL)
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 of 150 minutes [104]:

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

	Detector background (cpm)			
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

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americium) and for all sample count times [105]. For americium, the values for the 1960s are used because the americium process was not implemented in the 1950s [106].

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 the 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. In practice, high blank values (a subjective decision) were generally ignored, and suspect (unexpectedly high) sample results were either confirmed or overruled by recounting, resampling, or analyzing another aliquant [107].

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.

Analyte	Period	Blank count (cpm)	
		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:

$$s_{Blk} = \frac{\sqrt{B_{Blk}T}}{T} \quad \text{(A-5)}$$

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

$$B = T\left(B_{Det} + B_{Blk}\right) \quad \text{(A-6)}$$

$$s_{B0} = T\sqrt{s_{Det}^2 + s_{Blk}^2} \quad \text{(A-7)}$$

$$s_{B1} = \sqrt{B} \quad \text{(A-8)}$$

$$s_o = \sqrt{s_{B1}^2 + s_{B0}^2} \quad \text{(A-9)}$$

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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_o for the average of two aliquants is equal to the value of s_o 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 [108].

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. Recoveries used in MDA assessments.

		Recovery	
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

The recovery values are based on incomplete data sets 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. In general, values for all the processes are remarkably similar, except for americium from 1965 to 1971 [109].

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The volume adjustment factor V/A is incorporated into the calibration factor K as the reciprocal $1/(V/A)$, so 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 [110].

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 [111].

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. Fraction of alphas absorbed in residue.

Analyte	Period	95th percentile	
		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 [112]. For the purpose of the MDA assessment, dose reconstructors should use the value of F_a for the evaporation process.

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:

$$K = \varepsilon R V_f F_a \qquad \text{(A-10)}$$

The values of Δ_B and Δ_K are considered to be zero [113].

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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_o is equal to the value of s_{B0} multiplied by the square root of 2:

$$s_o = s_{B0}\sqrt{2} \quad \text{(A-11)}$$

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-4). 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 [114]. 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.

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-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.

The value of s_o 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.

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

Period	Values of the variables					MDA (dpm/24-hr sample)
	s_o	ε	R	V_f	F_a	
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. Values of variables and MDA for plutonium for extreme conditions.

Period	Values of the variables					MDA (dpm/24-hr sample)
	s_o	ε	R	V_f	F_a	
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

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

MDA (dpm/24-hr sample) for one extreme condition				
Period	s_o	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_o, R	s_o, V_f	s_o, 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

MDA (dpm/24-hr sample) for three extreme conditions				
Period	s_o, R, V_f	s_o, R, F_a	s_o, 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

Enriched 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.

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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_o is calculated for one aliquant, and for 1960 to 1971 the value of s_o 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	Values of the variables					MDA (dpm\24-hr sample)
	s_o	ε	R	V_f	F_a	
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.

The value of s_o 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-15. Values of variables and MDA for EU for extreme conditions.

Period	Values of the variables					MDA (dpm\24-hr sample)
	s_o	ε	R	V_f	F_a	
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

Depleted Uranium

The MDA for DU is assessed for two processes: fluorimetric mass measurements from 1952 to April 30, 1964, and electrodeposition and alpha-counting measurements from May 1, 1964, to 1971.

For the fluorimetric mass measurements, the MDA is assessed for one aliquant because the decision for detection was based on one aliquant, even though quantification was based on the average of two aliquants [115]. In Table A-17, the MDA at the extreme condition is based on the 95th-percentile volume.

For the electrodeposition and alpha-counting measurements, the MDA values tabulated for EU for 1964 to 1971 apply also to DU.

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

MDA (dpm/24-hr sample) for one extreme condition				
Period	s_o	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_o, R	s_o, V_f	s_o, 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_o, R, V_f	s_o, R, F_a	s_o, 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.

	Values of the variables			MDA (µg/24-hr sample)	
Period	s_{B0}	**Median** K	**Extreme** K	**Median**	**Extreme**
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-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.

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

Period	Values of the variables					MDA (dpm/24-hr sample)
	s_o	ε	R	V_f	F_a	
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

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

Period	Values of the variables					MDA (dpm/24-hr sample)
	s_o	ε	R	V_f	F_a	
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_o 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_o	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_o, R	s_o, V_f	s_o, 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_o, R, V_f	s_o, R, F_a	s_o, 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

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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.

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

Period	Values of the variables					MDA (dpm/24-hr sample)
	ε	R	V_f	F_a	ε	
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	Values of the variables					MDA (dpm\24-hr sample)
	s_o	ε	R	V_f	F_a	
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_o 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.

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

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Table A-23. Values of the MDA for gross alpha measurements for one, two, or three extreme conditions.

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

MDA (dpm/24-hr sample) for two extreme conditions						
Period	s_o, R	s_o, V_f	s_o, 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

MDA (dpm/24-hr sample) for three extreme conditions				
Period	s_o, R, V_f	s_o, R, F_a	s_o, 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

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,

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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.

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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. [116]

B.2 MDA METHODOLOGY

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

$$MDA = \frac{\left(1+\Delta_K\right)\left(2\Delta_B B + 2ks_o + 3\right)}{KT} \tag{B-1}$$

where:

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

$$s_o = \sqrt{s_{B1}^2 + \frac{s_{B0}^2}{m^2}} \tag{B-2}$$

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

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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.

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:

- 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 [117]

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 [118]
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 REMAB phantom from Alderson Research Laboratories. No adjustment was made for CWT.

1969 to 1976 [119]
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:

1. The ROI of the 59.5-keV photopeak of ^{241}Am was expanded.

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2. 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.

3. The standard count time was changed to 2,000 seconds (1,000 seconds for expedited counts).

4. 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):

1. Subjects were characterized by an index *I* equal to the ratio of the subject's weight (*W*, pounds) divided by twice the subject's height (*H*, inches).

2. A population of at least 20 known cold (unexposed) subjects of a diversity of indices was counted to generate a data set of net count rate versus index.

3. A curve fit to the data set generated a prediction equation with the index as the variable.

4. 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 [120], 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 [121]
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,

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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.

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:

1. 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 ($CWT = 2I - 0.1$).

2. Calibration was accomplished using a Masonite phantom from 1976 to 1978.

3. Calibration was accomplished using the LLNL phantom starting in 1979.

4. 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 [122]
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
1991 EG&G Ortec organ pipe detectors, 20 cm^2 per detector

No other significant changes were made during this period.

1995 to 2005 [123]
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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Another significant change (RFETS 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
	Index = 1.35	1.26
	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

The discontinuity factors for the *CWT* can be calculated by any index *I* using:

$$CWT\ Discontinuity\ Factor = 0.5364e^{0.635I} \qquad \text{(B-5)}$$

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

Count time T = 20 MLT or 40 MLT [124]

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_o 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_o = 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 [125] and incorporating the discontinuity factors (Equation B-5, 1.30, and 1.45) is given by:

$$K = \frac{55.13e^{-0.2359(2I-0.1)}}{e^{0.635I}} \qquad \text{(B-6)}$$

The calibration factor for the system with only one detector over the left portion of the chest is given by Equation B-6 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

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 + Δ_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

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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_o = 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, 1969 – FOR NAI(TL) AND PHOSWICH DETECTOR SYSTEMS

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 seconds or 2,000 seconds [126]

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 seconds
B = 2,200 for T = 2,000 seconds
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_o 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 seconds
R = 1,667 for T = 2,000 seconds
s_{B1}^2 = Total subject count + $R = B + 2R$
= 2,767 for T = 1,000 seconds
= 5,533 for T = 2,000 seconds

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$s_{B0}^2 = (0.1B)^2$
= 12,100 for T = 1,000 seconds
= 48,400 for T = 2,000 seconds
m = 1
s_o = 121.9 for T = 1,000 seconds
= 232.2 for T = 2,000 seconds

The ^{241}Am calibration factor K for two detectors, normalized to the calibration with the LLNL phantom and incorporating the discontinuity factors (Equation B-5 and 1.30) is given by:

$$K = \frac{79.94e^{-0.2359(2I-0.1)}}{e^{0.635I}} \quad \text{(B-7)}$$

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, 1976 – FOR ORTEC GERMANIUM DETECTOR SYSTEMS

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 seconds or 2,000 seconds

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.

$\Delta_B = 0$
m = 8

For eight detectors:

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

For five detectors:

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

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

For eight detectors:

T = 1,000 seconds:
s_{B1} = 6.53 s_{B0} = 18.5 s_o = 6.93

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T = 2,000 seconds:
s_{B1} = 9.23 s_{B0} = 26.1 s_o = 9.79

For five detectors:

T = 1,000 seconds:
s_{B1} = 5.17 s_{B0} = 14.6 s_o = 5.48
T = 2,000 seconds:
s_{B1} = 7.30 s_{B0} = 20.7 s_o = 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-5 and 1.30 (for pre-1979 systems)] is given by:

$$K = \frac{24.12e^{-0.3398(2I-0.1)}}{e^{0.635I}} \quad \text{(B-8)}$$

and, for Ortec systems 1979 and after:

$$K = \frac{31.36e^{-0.3398(2I-0.1)}}{e^{0.635I}} \quad \text{(B-9)}$$

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 for the Ortec germanium detector system.

	Eight-detector calibration factor (K)	
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, 1978 – FOR PGT I GERMANIUM DETECTOR SYSTEMS

The PGT I germanium system is basically the same as the Ortec germanium system. The main difference is that the PGT I detectors had a larger surface area but a poorer resolution [127].

Δ_B = 0
m = 4

For eight detectors:

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

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For five detectors:

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

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

For eight detectors:

T = 1,000 seconds:
s_{B1} = 7.75 s_{B0} = 15.5 s_o = 8.67

T = 2,000 seconds:
s_{B1} = 10.95 s_{B0} = 21.9 s_o = 12.2

For five detectors:

T = 1,000 seconds:
s_{B1} = 6.12 s_{B0} = 12.2 s_o = 6.84

T = 2,000 seconds:
s_{B1} = 8.66 s_{B0} = 17.3 s_o = 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-5 and 1.30 (for pre-1979 systems)] is given by:

$$K = \frac{34.09e^{-0.3292(2I-0.1)}}{e^{0.635I}} \quad \text{(B-10)}$$

and, for PGT I systems 1979 and after:

$$K = \frac{44.318e^{-0.3292(2I-0.1)}}{e^{0.635I}} \quad \text{(B-11)}$$

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

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

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

	Eight-detector calibration factor (K)	
Index	**Pre-1979**	**1979 →**
0.90	11.00	14.30
1.35	6.15	7.99
1.80	3.43	4.46

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

The PGT II germanium system is basically the same as the Ortec and PGT I systems. The main difference is that the PGT II detectors, again, had a larger surface area but a poorer resolution [128].

$\Delta_B = 0$
$m = 4$

For eight detectors:

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

For five detectors:

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

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

For eight detectors:

$T = 1{,}000$ seconds:
$s_{B1} = 8.26$ $s_{B0} = 16.5$ $s_o = 9.23$

$T = 2{,}000$ seconds:
$s_{B1} = 11.7$ $s_{B0} = 23.4$ $s_o = 13.1$

For five detectors:

$T = 1{,}000$ seconds:
$s_{B1} = 6.53$ $s_{B0} = 13.1$ $s_o = 7.31$

$T = 2{,}000$ seconds:
$s_{B1} = 9.23$ $s_{B0} = 18.5$ $s_o = 10.3$

The ^{241}Am calibration factor K for two arrays with a total of eight detectors (incorporating Equation B-5), is given by:

$$K = \frac{38.65e^{-0.3579(2I-0.1)}}{e^{0.635I}} \qquad \text{(B-12)}$$

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.

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

Index	Eight-detector calibration factor (*K*)
0.90	11.88
1.35	6.47
1.80	3.52

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

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

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

The ^{241}Am calibration factor K for two arrays with a total of six detectors (incorporating Equation B-5) is given by:

$$K = \frac{34.32e^{-0.2946(2I-0.1)}}{e^{0.635I}} \qquad \text{(B-13)}$$

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

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

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

ATTACHMENT B
MINIMUM DETECTABLE ACTIVITY FOR IN VIVO LUNG COUNTS AT ROCKY FLATS (continued)

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

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.

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

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

The ^{241}Am calibration factor K for two arrays with a total of six detectors, incorporating Equation B-5, is given by:

$$K = \frac{42.36e^{-0.3708(2I-0.1)}}{e^{0.635I}} \quad \text{(B-14)}$$

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

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

Index	**Six-detector 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:

$$\varepsilon = a_1^{a_2 CWT} \quad \text{(B-15)}$$

ATTACHMENT B
MINIMUM DETECTABLE ACTIVITY FOR IN VIVO LUNG COUNTS AT ROCKY FLATS (continued)

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:

$$K = 35.9^{-0.41CWT} \quad \text{(B-16)}$$

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:

$$MDA\ Conversion\ Factor = \frac{1\times10^6 - ppm\ ^{241}\text{Am}}{48.2 ppm\ ^{241}\text{Am}} \quad \text{(B-17)}$$

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 [129]. 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. The value of 100 ppm ^{241}Am can be taken as the lower bound [130] and represents freshly purified plutonium.

ATTACHMENT B
MINIMUM DETECTABLE ACTIVITY FOR IN VIVO LUNG COUNTS AT ROCKY FLATS (continued)

Table B-10. Americium ingrowth in RFP plutonium.

	Am-241 ppm at time (yr) after intake					
	Initial fraction Pu-241 = 0.0036			Initial fraction Pu-241 = 0.0050		
Years	100	1,000	10,000	100	1,000	10,000
1	270	1,200	10,200	340	1,200	10,200
2	430	1,300	10,300	560	1,500	10,400
4	730	1,600	10,600	980	1,900	10,800
6	1,000	1,900	10,800	1,400	2,200	11,100
10	1,500	2,400	11,200	2,000	2,900	11,700
20	2,300	3,200	11,900	3,100	4,000	12,700
30	2,800	3,600	12,200	3,800	4,700	13,200
40	3,000	3,900	12,300	4,200	5,000	13,500
50	3,200	4,000	12,300	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:

$$A = L_1 P_0 \left(e^{-\lambda_{Pu241} T} - e^{-\lambda_{Am241} T} \right) + \frac{1 \times 10^6 A_0 L_2}{A_0 L_2 + e^{-\lambda_{Pu239} T}} \quad \text{(B-18)}$$

where

A = ppm ^{241}Am at time T (years)
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 years) = 0.0481
λ_{Am241} = decay constant for ^{241}Am (half-life = 433 years) =0.00160
A_0 = initial ppm ^{241}Am

$$L_2 = \frac{e^{-\lambda_{Am241} T}}{1 \times 10^{-6} - A_0} \quad \text{(B-19)}$$

λ_{Pu239} = decay constant for ^{239}Pu (half-life = 24,100 years) = 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.

Table B-11. Americium MDA for in vivo lung counts at RFP.[a]

Period	Detector system	Index[b]	MDA (nCi) for Am-241[c,d]				Comments
			Minimum system		Standard system		
			Half time[e]	Full time	Half time[e]	Full time	
1964–1968	NaI(Tl) 4×4	0.90	1.7	1.5	1.3	1.2	Full time = 40 MLT. Minimum system is one detector over the left chest. Standard system is two detectors, over right and left chests.
		1.35	2.8	2.5	2.1	1.9	
		1.80	4.6	4.1	3.5	3.2	
1969–		0.90	–	–	0.80	0.76	Full time = 2,000 s. Standard system is two detectors, over right and left chests.
		1.35	–	–	1.3	1.3	
		1.80	–	–	2.2	2.0	
1973–	Phoswich	0.90	–	–	1.2	1.2	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.
		1.35	–	–	2.0	2.0	
		1.80	–	–	3.3	3.2	
1976–1978[f]	Ortec Arrays (High-purity Ge)	0.90	0.26	0.18	0.20	0.14	Full time = 2,000 s. Standard system is eight detectors in two arrays. Minimum system is five detectors in two arrays.
		1.35	0.48	0.32	0.37	0.25	
		1.80	0.86	0.59	0.66	0.45	
1979–		0.90	0.20	0.14	0.16	0.11	
		1.35	0.37	0.25	0.28	0.19	
		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	
		1.35	0.40	0.27	0.31	0.21	
		1.80	0.71	0.49	0.55	0.38	
1979–		0.90	0.17	0.12	0.13	0.09	
		1.35	0.31	0.21	0.24	0.16	
		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	
		1.35	0.40	0.28	0.31	0.21	
		1.80	0.74	0.50	0.57	0.39	
1985–	PGT Organ Pipe (OP) Ge Detectors	0.90	–	–	0.15	0.11	Standard system = six detectors. Standard count time = 2,000 s. Occasionally, five or seven detectors were used.
		1.35	–	–	0.26	0.18	
		1.80	–	–	0.46	0.32	
1991–	EG&G Organ Pipe Ge Detectors	0.90	–	–	0.14	0.10	Standard system = six detectors. Standard count time = 2,000 s.
		1.35	–	–	0.26	0.18	
		1.80	–	–	0.48	0.33	
1995–	Ortec 2 Organ Pipe Ge Detectors	0.90	–	–	–	0.14	Standard system = four detectors. Standard count time = 2,000 s.
		1.35	–	–	–	0.3	
		1.80	–	–	–	0.6	

a. Americium-241 grows into the plutonium mixture from the nuclear transformation of ^{241}Pu. The initial weight fraction of ^{241}Pu 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 ^{241}Am, the ppm ^{241}Am 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. 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).

c. To convert the MDA for ^{241}Am to the MDA for ^{239,240}Pu, multiply the MDA for ^{241}Am by [(1 × 10^6 – ppm ^{241}Am) ÷ (48.2 × ppm ^{241}Am)], where ppm ^{241}Am is the parts per million by weight of the ^{241}Am in the plutonium mixture at the time of the lung count.

d. – = not applicable.

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

f. 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.

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 - Fluorimeter (Reported in micrograms/); B1: Electroplating; B2: Ether Extract (Reported in gr) hrs.; C2: Carrier Precipitation; C3: 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 AD	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	7.2		10		8.1		BK	15	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	19	12
	Count	11	10	BK		8.8	BK	BK	BK	BK	BK	BK	BK	BK	BK
	Tech.	A	A	A		A	A	A	A	A	A	A	A	A	A

Man No. [redacted] Name of Employee: [redacted] 1414-336 Badge No. 327-110 Area 3644

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 - Other 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												
	Count												
	Tech.												
13	Day								19		16	6	24
	Count								BK ✓		BK ✓	BK ✓	Bk ✓
	Tech.								B_2 ✓		B_2	B_2	B_2
14	Day	27	24	19	28	20	18	23	16	30	27		1 17
	Count	3H	BK	10	BK	BK	35	12	9.3	11.6	12.5		15 16
	Tech.	B ✓ RES ✓	B ✓	B ✓	B ✓	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
	Count	36.4	BK	8.7	8.5	BK	6.4	7.4	BK	BK	7.4	BK	7.6
	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
	Count		BK BK	BK	BK	15.4 8.7	BK	BK	BK	BK	BK	BK	BK
	Tech.		A	A	A	A	A	A	A	A	A	A	A

Man No. [redacted] Name of Employee: [redacted] Badge No. 144-200 Area 44

SPECIFY EMPLOYEE NO. [redacted]
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	14	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	4	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. [redacted] Area [redacted]

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	Dec Level	Aspec	DQO	BLANK Chem 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 Analyte Val		Recovery	Result (DPM)	Error
Routine Urine	8-SEP-1993 09:24	25747/1	0.016	0	AMN	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)

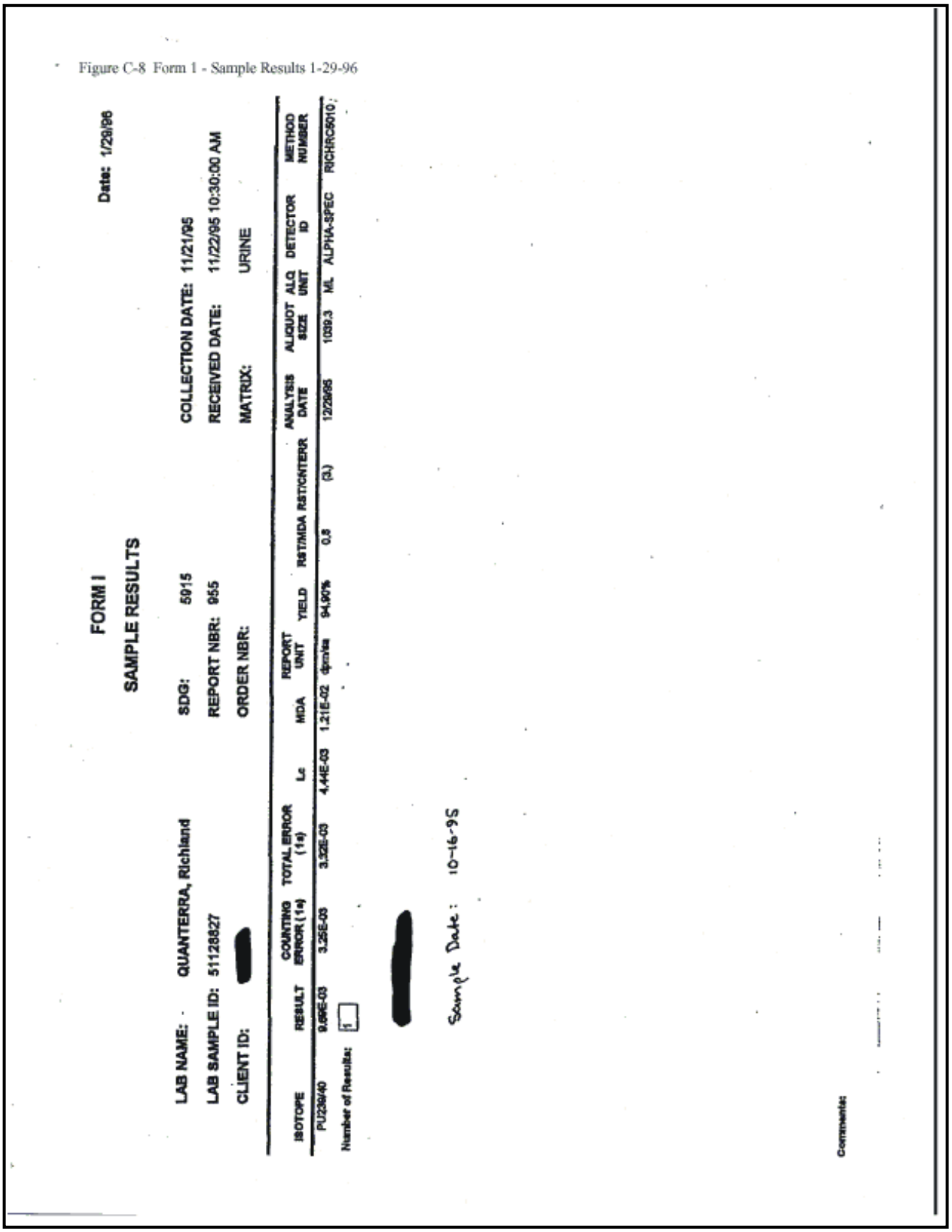


Figure C-8 Form 1 - Sample Results 1-29-96

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	RST/CNTERR	ANALYSIS DATE	ALIQUOT SIZE	ALQ. UNIT	DETECTOR ID	METHOD NUMBER
PU239/40	9.69E-03	3.25E-03	3.32E-03	4.44E-03	1.21E-02	dpm/sa	94.90%	0.8	(3.)	12/29/95	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 [redacted] 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: 80709501 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

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

0013

* (V=Valid; I=Invalid)

Comments:

(Validated by)

Date 8/3/98

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:

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 & GR

NAME [redacted]
MAN NO. [redacted]

DATE	Element	Type Sample	Type Anal	RESULTS	DATE	Element	Type Sample	Type Anal	RESULTS
1-8-74	H_3	PD	6	7990 ± 486					
2-5-74	"	"	150	21603 ± 5801					
8-25-76	"	"		1253 ± 426					
10-5-76	"	"	37	1320 ± 498					
4-11-78	PU	FECAL	52	32.3 ± 1.8					
"	AM	"	"	0.15 ± 0.13					
"		"	"	PPM AM 100					
5-17-78	PU	NOSE	93	1.0 ± 0.28					
"	AM	"	"	0.43 ± 0.20					
"		"	"	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~~
~~URM~~
Pers. File

HEALTH PHYSICS

BODY COUNTER INFORMATION

Name [redacted] Man No. [redacted] Date 12-8-65 Time 4045 CT

Reason for Counting: 76 Final Exposure minus Aby + Meth

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]
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	μg
#1 : 4"x4mm NaI Crystal	R chest	Am	60	52.2	34.9	17.3	0.06
#2 : 4"x4mm NaI Crystal	L chest	Am	60	46.6	34.5	11.7	0.4
#3 : 9"x4" NaI Crystal				____ nci Cs^{137} ; ____ g K			

(1) Gross c/m — Background

(2) Gross c/m — Bkgd. — Match subject

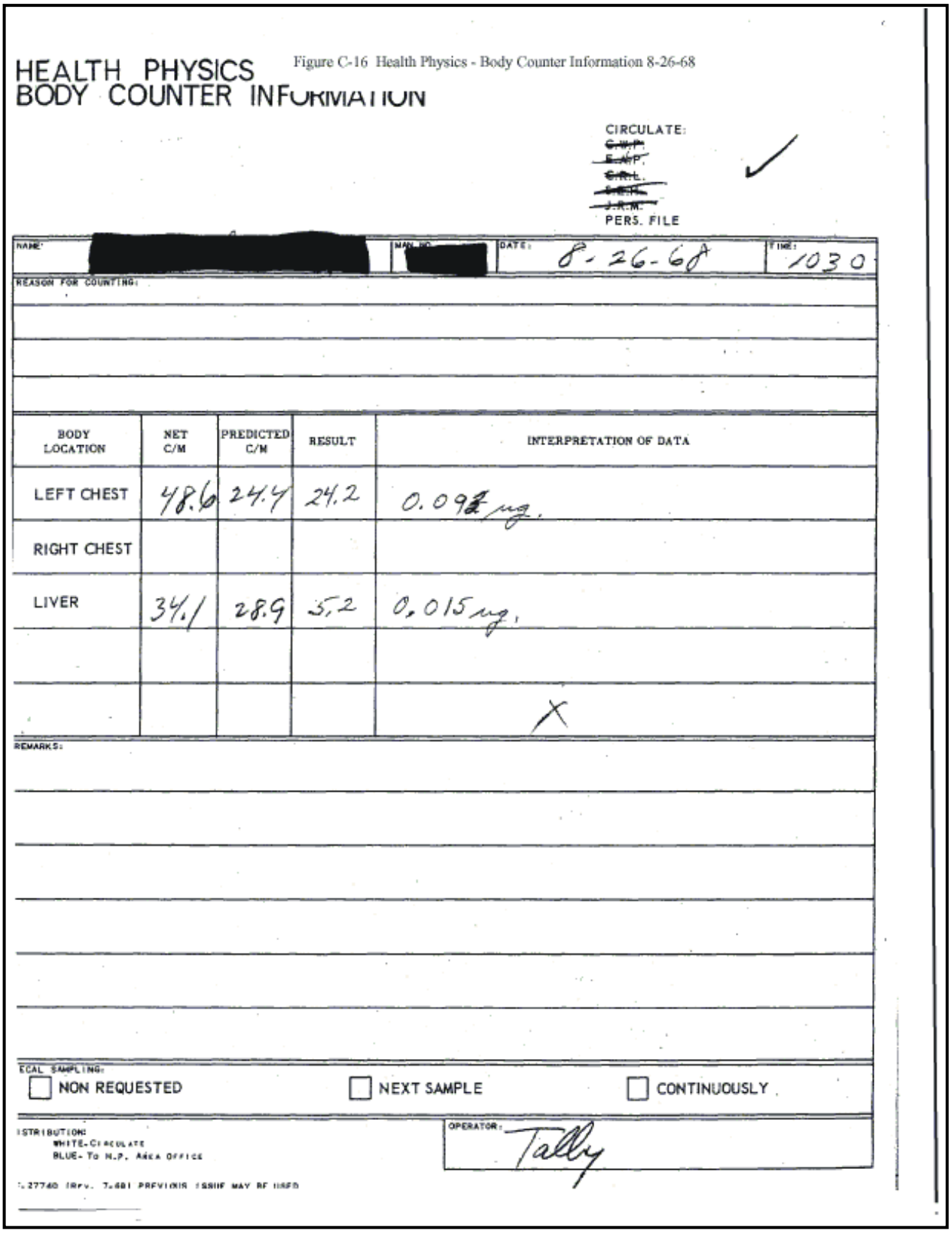
Tally
Operator

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

RF-22740 (Rev. 8-67) Prev. Issue May Be Used

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

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



HEALTH PHYSICS
BODY COUNTER INFORMATION

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

CIRCULATE:
~~C.W.P.~~
~~E.A.P.~~
~~C.R.L.~~
~~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	0.098 ug.
RIGHT CHEST				
LIVER	34.1	28.9	5.2	0.015 ug.
				X

REMARKS:

ECAL SAMPLING:
☐ NON REQUESTED ☐ NEXT SAMPLE ☐ CONTINUOUSLY

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

OPERATOR: Tally

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.R.H.
J.R.M.
PERS. FILE

NAME: [redacted] BADGE 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:
Indef 1.40

FECAL SAMPLING:
☐ NON REQUESTED ☐ NEXT SAMPLE ☐ CONTINUOUSLY

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

OPERATOR: Tolly

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: 1.6.5 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: R.W. Perchonok SUPERVISOR:

RST-285 (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: 5-30-25 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: RBF 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 8 0

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										
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 nCi Pu @ 1000 ppm Am; 12.67 @ 2585 ppm Am
Am: 6.25 c/m per nCi 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: [redacted] 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+ 0.968 + 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: -- PHOSWICH

60 & 63 KEV

GROSS CT/MIN =	33.78
BKG CT/MIN =	33.66
NET CT/MIN =	0.12
STD DEV =	1.42
CUTOFF =	2.34

93 KEV

GROSS CT/MIN =	33.66
BKG CT/MIN =	32.46
NET CT/MIN =	1.20
STD DEV =	1.41
CUTOFF =	2.32

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

RESULTS ARE NORMAL -- X NONE -- OVERNIGHT
URINE SAMPLING: -- 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

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

TH-234/U-238

	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: _____

APPROVED BY: _____

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
CAL 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	GROSS CT/MIN =	4.26	
BKG CT/MIN =	4.25	BKG CT/MIN =	4.78	
NET CT/MIN =	4.87	NET CT/MIN =	-0.52	
STD DEV =	0.55	STD DEV =	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: (X) NONE __ OVERNIGHT __ 24 HOUR __ THREE 24 HOUR
FECAL SAMPLING: (X) 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	=	4.71
BKG CT/MIN	=	4.51	=	4.76
NET CT/MIN	=	-0.67	=	-0.05
STD DEV	=	0.39	=	0.42
CUTOFF	=	0.63	=	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 # 8
COUNT TIME 2000 SEC INDEX #: 1.3615
BUILD #: [redacted] ROOM#: OPERATION:
REASON FOR COUNT: __ NEW __ RECOUNT __ ROUTINE ___ SEPARATION
__ POSSIBLE INHALATION (_____ PPM)
OTHER Recall
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	GROSS CT/MIN =	4.38	3.78
BKG CT/MIN	=	3.82	BKG CT/MIN =	4.12	3.41
NET CT/MIN	=	8.87	NET CT/MIN =	0.26	0.37
STD DEV	=	0.64	STD DEV =	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: KGJ

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:

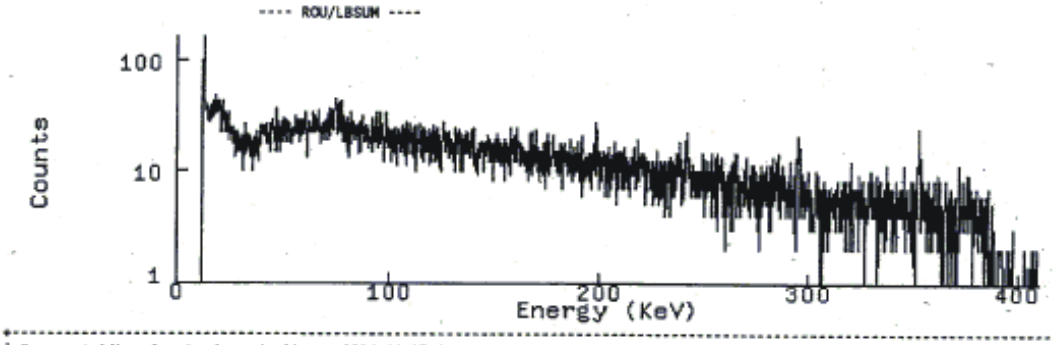
Identification #: [redacted]
Count Started: 6-MAR-1996 15:06:55.
Intake Date: 6-MAR-1996 15:06:55.
Frequency:
Operator: K.J. Nannie / K.J. Moul
Weight: 213.00
Date of Birth: [redacted]

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: GAMMAM V1.2,PEAKEFF V2.2,NID V3.1,WTMEAN 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

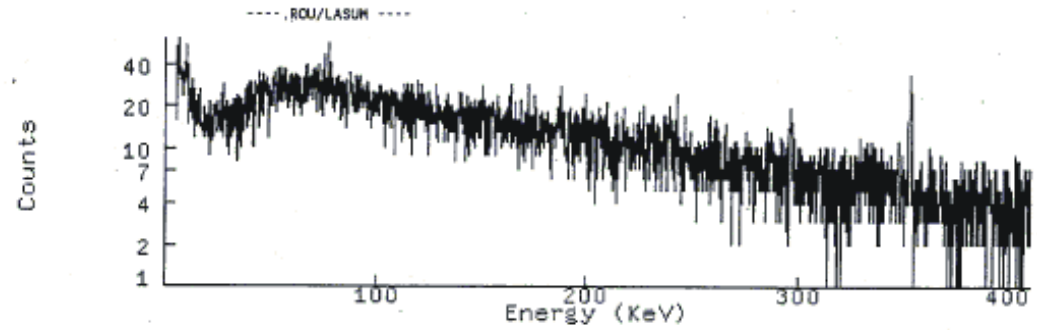
Arrangement/Geometry: ROU/LASUM
Elapsed Live Time: 0 00:30:00.00

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	-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. ROU/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] Identification #: [redacted]
Employer: [redacted] Count Started: 14-JUN-2001 11:05:57
Job Code: [redacted] Intake Date: 15-OCT-1965 00:00:00
Sex: Male Date of Birth: [redacted]
Height: 68.000 Weight: 175.00
CWT: 3.0738
Reason: OTHER
Comment: MEDICAL RECALL

Operator: JSM Initials: [redacted]

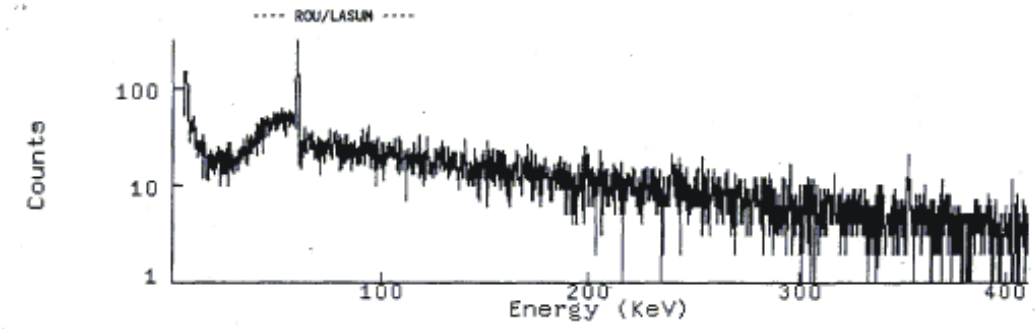
Counter: Room A Lung Arrangement/Geometry: ROU/LASUM
Analysis Limits (channels): 100 to 2000 Elapsed Live Time: 0 00:30:00.00
Detector Count Rate (gross cps): 4.062 3.619 3.782 3.563
Analysis Energies (keV) Am-241: 59.54 U-235: 143, 185, and 204 U-238: 63 and 93

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 COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT

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ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (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 coworker 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. Data analysis for 1989 and later data were performed by NIOSH (2006b).

D.1.2 Bioassay Data Selection

Urinalysis data for uranium and plutonium from 1952 to 1988 were extracted from the Comprehensive Epidemiology Data Resource (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).

The RFP HIS-20 database was obtained after the coworker analysis had been performed. A comparison of CEDR and HIS-20 was made. The databases are 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 coworker 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 Rocky Flats, 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.

In vivo ^{241}Am lung data from 1965 to 1988 were extracted from a Microsoft Access table named "RFFACW02_RFWB." There were just fewer than 80,000 ^{241}Am records in the lung database. From

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

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 data sets, 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 [131]. 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 data set. 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 COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

Table D-1. Summary of uranium urinary excretion rate analyses, 1953 to 1988.[a]

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)	Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
7/1/1953	**3.727**	**10.008**	5/15/1964	8.368	23.389
2/15/1954	3.866	10.362	8/15/1964	8.161	22.172
5/15/1954	4.161	11.472	11/15/1964	8.297	23.535
8/15/1954	3.732	10.074	**7/1/1965**	**7.823**	**20.789**
11/15/1954	3.409	9.389	**7/1/1966**	**7.432**	**18.360**
2/15/1955	3.225	9.019	**7/1/1967**	**7.445**	**18.440**
5/15/1955	3.333	9.487	**7/1/1968**	**7.430**	**18.459**
8/15/1955	3.434	9.406	**7/1/1969**	**7.509**	**18.518**
11/15/1955	3.442	9.875	**7/1/1970**	**7.440**	**18.275**
2/15/1956	3.310	9.039	**7/1/1971**	**7.421**	**18.131**
5/15/1956	3.497	9.843	**7/1/1972**	**7.316**	**18.176**
8/15/1956	3.635	10.213	**7/1/1973**	**7.403**	**18.059**
11/15/1956	3.302	9.121	**7/1/1974**	**7.388**	**18.084**
2/15/1957	3.460	9.894	**7/1/1975**	**7.378**	**18.104**
5/15/1957	3.492	10.173	**7/1/1976**	**7.418**	**18.037**
8/15/1957	3.655	10.781	**7/1/1977**	**0.172**	**0.538**
11/15/1957	3.700	10.996	**7/1/1978**	**0.893**	**2.355**
2/15/1958	4.089	12.575	**7/1/1979**	**0.444**	**2.037**
5/15/1958	3.739	10.593	**7/1/1980**	**0.241**	**1.049**
8/15/1958	3.907	11.266	**7/1/1981**	**0.178**	**1.109**
11/15/1958	4.705	14.316	2/15/1982	0.237	1.152
2/15/1959	4.381	13.159	5/15/1982	0.062	0.677
5/15/1959	5.518	17.908	8/15/1982	0.016	0.211
8/15/1959	5.544	16.566	11/15/1982	0.112	0.741
11/15/1959	5.887	19.134	2/15/1983	0.221	1.062
2/15/1960	8.806	33.071	5/15/1983	0.432	1.330
5/15/1960	6.856	22.227	8/15/1983	0.327	1.576
8/15/1960	7.476	24.214	11/15/1983	0.072	0.646
11/15/1960	6.602	23.668	2/15/1984	0.273	1.400
2/15/1961	5.944	20.258	5/15/1984	0.221	1.330
5/15/1961	5.722	18.628	8/15/1984	0.133	0.997
8/15/1961	5.574	18.290	11/15/1984	0.065	0.464
11/15/1961	6.598	22.669	2/15/1985	0.034	0.410
2/15/1962	5.862	20.451	5/15/1985	0.030	0.281
5/15/1962	4.692	15.380	8/15/1985	0.040	0.511
8/15/1962	5.654	16.742	11/15/1985	0.037	0.415
11/15/1962	4.397	13.827	2/15/1986	0.029	0.357
2/15/1963	4.166	13.230	5/15/1986	0.033	0.339
5/15/1963	4.175	13.154	8/15/1986	0.018	0.207
8/15/1963	3.841	12.283	11/15/1986	0.022	0.316
11/15/1963	3.601	11.507	**7/1/1987**	**0.057**	**0.467**
2/15/1964	6.354	18.506	**7/1/1988**	**0.059**	**0.412**

a. All results shown in **bold** are annual rather than quarterly averages.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

Table D-2. Summary of plutonium urinary excretion rate analyses, 1952 to 1988.[a]

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)	Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
7/1/1952	**2.514**	**8.198**	11/15/1972	0.028	0.168
7/1/1953	**0.716**	**1.046**	2/15/1973	0.024	0.145
7/1/1954	**0.575**	**1.053**	5/15/1973	0.033	0.180
7/1/1955	**0.469**	**0.919**	8/15/1973	0.067	0.305
7/1/1956	**0.615**	**1.264**	11/15/1973	0.061	0.268
7/1/1957	**2.610**	**12.006**	2/15/1974	0.060	0.224
2/15/1958	2.173	10.041	5/15/1974	0.049	0.189
5/15/1958	1.037	2.872	8/15/1974	0.033	0.144
8/15/1958	1.295	3.801	11/15/1974	0.016	0.109
11/15/1958	0.919	2.581	2/15/1975	0.021	0.104
2/15/1959	0.709	1.542	5/15/1975	0.019	0.095
5/15/1959	0.942	2.276	8/15/1975	0.022	0.200
8/15/1959	0.945	2.482	11/15/1975	0.015	0.097
11/15/1959	0.560	1.211	2/15/1976	0.016	0.144
2/15/1960	0.614	1.353	5/15/1976	0.021	0.102
5/15/1960	0.596	1.221	8/15/1976	0.015	0.104
8/15/1960	0.453	0.955	11/15/1976	0.043	0.184
11/15/1960	0.573	1.528	2/15/1977	0.083	0.262
2/15/1961	0.728	1.625	5/15/1977	0.092	0.245
5/15/1961	0.691	1.377	8/15/1977	0.072	0.190
8/15/1961	0.754	2.035	11/15/1977	0.062	0.188
11/15/1961	0.656	1.645	2/15/1978	0.095	0.307
2/15/1962	0.337	0.809	5/15/1978	0.060	0.199
5/15/1962	0.326	0.735	8/15/1978	0.056	0.201
8/15/1962	0.271	0.589	11/15/1978	0.033	0.134
11/15/1962	0.220	0.431	2/15/1979	0.062	0.237
2/15/1963	0.250	0.467	5/15/1979	0.013	0.100
5/15/1963	0.248	0.496	8/15/1979	0.013	0.087
8/15/1963	0.238	0.432	11/15/1979	0.029	0.139
11/15/1963	0.252	0.562	2/15/1980	0.017	0.106
2/15/1964	0.296	0.810	5/15/1980	0.017	0.064
5/15/1964	0.249	0.483	8/15/1980	0.013	0.061
8/15/1964	0.379	1.668	11/15/1980	0.004	0.035
11/15/1964	0.334	1.066	2/15/1981	0.006	0.037
2/15/1965	0.283	0.757	5/15/1981[b]	2.25E-04	0.006
5/15/1965	0.348	1.085	8/15/1981	0.005	0.036
8/15/1965	0.221	0.417	11/15/1981	0.008	0.056
11/15/1965	0.266	0.646	2/15/1982[b]	1.43E-04	0.007
2/15/1966	0.293	0.821	5/15/1982[b]	3.11E-04	0.011
5/15/1966	0.237	0.554	8/15/1982[b]	1.37E-04	0.004
8/15/1966	0.213	0.430	11/15/1982[b]	2.90E-04	0.006
11/15/1966	0.252	0.625	2/15/1983	0.001	0.017
2/15/1967	0.251	0.622	5/15/1983[b]	3.99E-04	0.008
5/15/1967	0.240	0.565	8/15/1983	0.002	0.016
8/15/1967	0.199	0.413	11/15/1983	0.004	0.029
11/15/1967	0.236	0.535	2/15/1984	0.008	0.050
2/15/1968	0.228	0.526	5/15/1984	0.053	0.222
5/15/1968	0.205	0.461	8/15/1984	0.011	0.071
8/15/1968	0.252	0.585	11/15/1984	0.054	0.196

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)	Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
11/15/1968	0.278	0.724	2/15/1985	0.010	0.080
2/15/1969	0.292	0.692	5/15/1985	0.025	0.100
5/15/1969	0.266	0.606	8/15/1985	0.014	0.081
8/15/1969	0.240	0.519	11/15/1985	0.017	0.100
11/15/1969	0.264	0.558	2/15/1986	0.005	0.033
2/15/1970	0.242	0.515	5/15/1986	0.004	0.038
5/15/1970	0.165	0.623	8/15/1986	0.007	0.038
8/15/1970	0.100	0.423	11/15/1986	0.008	0.042
11/15/1970	0.120	0.470	2/15/1987	0.004	0.030
2/15/1971	0.091	0.366	5/15/1987	0.005	0.036
5/15/1971	0.055	0.209	8/15/1987	0.008	0.051
8/15/1971	0.073	0.293	11/15/1987	0.008	0.050
11/15/1971	0.061	0.249	2/15/1988	0.003	0.032
2/15/1972	0.046	0.398	5/15/1988	0.002	0.033
5/15/1972	0.046	0.442	8/15/1988	0.005	0.034
8/15/1972	0.029	0.199	11/15/1988	0.006	0.038

a. All results shown in **bold** are annual averages rather than quarterly averages. 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.

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 [132]. A uniform absolute error of 1 was applied to all results, thus assigning the same weight to each result. IMBA requires 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) 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

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

Table D-3. Americium-241 lung count bioassay data for individualized ^{239}Pu Type S fits.[a]

Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)	Effective sample date	50th percentile (dpm/24 hr)	84th percentile (dpm/24 hr)
7/1/1972	**6.73E-05**	**0.003**	5/15/1981	0.016	0.140
2/15/1973	0.005	0.059	8/15/1981	0.016	0.138
5/15/1973	0.010	0.107	11/15/1981	0.016	0.136
8/15/1973	0.025	0.188	2/15/1982	0.013	0.126
11/15/1973	0.009	0.095	5/15/1982	0.011	0.111
2/15/1974	0.005	0.067	8/15/1982	0.010	0.102
5/15/1974	0.007	0.080	11/15/1982	0.009	0.081
8/15/1974	0.007	0.079	2/15/1983	0.006	0.066
11/15/1974	0.007	0.079	5/15/1983	0.002	0.031
2/15/1975	0.017	0.150	8/15/1983	0.005	0.055
5/15/1975	0.027	0.180	11/15/1983	0.008	0.063
8/15/1975	0.039	0.244	2/15/1984	0.005	0.058
11/15/1975	0.048	0.278	5/15/1984	0.006	0.058
2/15/1976	0.043	0.261	8/15/1984	0.005	0.054
5/15/1976	0.044	0.254	11/15/1984	0.008	0.067
8/15/1976	0.017	0.133	2/15/1985	0.004	0.042
11/15/1976	0.012	0.111	5/15/1985	0.005	0.051
2/15/1977	0.010	0.097	8/15/1985	0.003	0.035
5/15/1977	0.008	0.082	11/15/1985	0.003	0.037
8/15/1977	0.007	0.061	2/15/1986	0.004	0.049
11/15/1977	0.004	0.051	5/15/1986	0.007	0.054
2/15/1978	0.008	0.083	8/15/1986	0.005	0.057
5/15/1978	0.007	0.070	11/15/1986	0.004	0.043
8/15/1978	0.007	0.066	2/15/1987	0.008	0.072
11/15/1978	0.004	0.045	5/15/1987	0.005	0.051
7/1/1979	**0.012**	**0.108**	8/15/1987	0.009	0.091
2/15/1980	0.026	0.195	11/15/1987	0.009	0.072
5/15/1980	0.020	0.159	2/15/1988	0.006	0.061
8/15/1980	0.021	0.171	5/15/1988	0.008	0.073
11/15/1980	0.027	0.207	8/15/1988	0.005	0.043
2/15/1981	0.018	0.151	11/15/1988	0.004	0.042

a. All results shown in **bold** are annual averages rather than quarterly averages.

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.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

D.2.2 Bioassay Fitting

IMBA was used to fit the bioassay results to a series of inhalation intakes. Data from 1952 through 1988 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 1988 was divided into multiple chronic intake periods.

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 solubility, this approach was used where it was 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 1988. Excretion data are shown in Table D-1. The solid lines in Figures D-1 and D-2 show the individual fits to the 50th-percentile excretion rates for type F material. Figure D-3 is the combined fit for all the intake periods. Figure D-4 shows the overall fit to the 84th-percentile excretion rates for type F material. The same intake periods were applied for both percentiles because the values followed a similar pattern. Similarly, Figures D-5 and D-6 show the individual fits to the 50th-percentile excretion rates for type M material. Figure D-7 is the combined fit for all the intake periods. Figure D-8 shows the overall fit to the 84th-percentile excretion rates for type M material. Figures D-9 to D-13 and D-14 to D-18 show the individual fits to the 50th-and 84th- percentile excretion rates for type S material, respectively. Figures D-19 and D-20 show the 50th- and 84th-percentile predicted excretion rates, respectively, from all type S intakes. Table D-7 tabulates the derived intake rates for Types F, M, and S materials at both the 50th- and 84th-percentile levels along with the associated geometric standard deviations (GSDs). Data for 1989 and later were from NIOSH (2006b).

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. 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 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 both Type M and S 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-21 to D-24 and D-25 to D-28 show the individual fits to the 50th- and 84th-percentile excretion rates for type M materials, respectively. The same intake periods were applied for both percentiles because the values followed a similar pattern. Figures D-29 and D-30 show the 50th- and 84th-percentile predicted excretion rates, respectively, from all type M intakes. In addition, intake rates for Type M plutonium based on lung-counting measurements of the associated americium-241 were also derived. The plutonium urinalysis results were determined to be more limiting and thus were used for the final values. Table D-8 lists the 50th- and 84th-percentile intake rates along with the associated GSD determined from plutonium urinalysis. For comparison, the intake rate determined from the americium lung counts at the 50th percentile level are also given. Data for 1989 and later were from NIOSH (2006b).

Plutonium Type S— The solid lines in Figures D-31 to D-35 and D-36 to D-40 show the individual fits to the 50th- and 84th-percentile excretion rates for type S materials, respectively. The same intake periods were applied for both percentiles because the values followed a similar pattern. Figures D-41 and D-42 show the 50th- and 84th-percentile predicted excretion rates, respectively, from all type S intakes. Figures D-43 to D-45 and D-46 to D-48 show the individual fits to the 50th- and 84th-percentile ^{241}Am lung count data. 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. Data for 1989 and later were from NIOSH (2006b).

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. Data for 1989 and later were from NIOSH (2006b). In 1993, the Secretary of Energy formally announced the end of nuclear production at Rocky Flats. Remediation was completed at the RFP in late 2005. Coworker intakes 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. Table D-4 summarizes 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 material and Type S material. Table D-5 summarizes the 95th-percentile plutonium intake rates derived from the fits for Type M material.

For types S and Super S material, Table D-6 provides the urinalysis and lung-count based intakes rates to be used.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

Table D-4. Derived uranium intake rates, 1953 to 2005.

Period	95th percentile (dpm/d)		
	Type F material	Type M material	Type S material
1953–1958	74.2	303	5,266
1959	130	763	20,322
1960	212	763	20,322
1961	135	502	11,600
1962	163	502	11,600
1963	84.7	502	11,600
1964	161	516	7,391
1965–1976	118	516	7,391
1977–1988	8.52	11.5	458
1989–1993	5.21	21.8	426
1994–2005	1.64	6.72	101

Table D-5. Derived Type M plutonium intake rates, 1952 to 2005.

Period	95th percentile (dpm/d) Type M material
1952–1961	718
1962–1969	190
1970–1979	75.6
1980–1988	26.7
1989–1993	47.6
1994–2005	2.21

The Table D-6 intake rates should be used as follows:

- For doses to systemic organs, use the systemic intake rates in Table D-6 in accordance with the guidance in ORAUT-OTIB-0049, *Estimating Doses for Plutonium Strongly Retained in the Lung* (ORAUT 2010).
    - Type Super S coworker doses to systemic organs should be calculated using urinalysis-based intakes for all periods.
- Doses to the lungs and thoracic lymph nodes, gastrointestinal tract, and extrathoracic regions (nonsystemic organs) should be based on the nonsystemic intake rates in Table D-6 in accordance with the guidance in ORAUT-OTIB-0049 (ORAUT 2010).
    - Type Super S coworker doses to nonsystemic organs should be calculated based on the lung count-based intakes for 1972 through 1988.
    - Type Super S coworker doses to nonsystemic organs should be calculated based on the urinalysis-based intakes for 1952 through 1971 and for 1989 through 2005.
- For all coworker type Super S adjustments, the date of the last bioassay sample should be assumed to be equivalent to the intake period end date in Table D-6. Note that type Super S adjustments should be made separately for each intake period in Table D-6.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

Table D-6. Derived Type S plutonium intake rates, 1952 to 2005.

Systemic intake rates			Nonsystemic intake rates		
Intake period start date	Intake period end date	intake rate, dpm/d ^{239}Pu	Intake period start date	Intake period end date	intake rate, dpm/d ^{239}Pu
1/1/1952	12/31/1961	11,243 [a]	1/1/1952	12/31/1961	11,243 [a]
1/1/1962	12/31/1969	3,368 [a]	1/1/1962	12/31/1969	3,368 [a]
1/1/1970	12/31/1979	1,168 [a]	1/1/1970	12/31/1971	1,168 [a]
1/1/1980	12/31/1993	385 [a]	1/1/1972	12/31/1976	953 [b]
1/1/1994	12/31/2005	34.2 [a]	1/1/1977	12/31/1982	863 [b]
			1/1/1983	12/31/1988	419 [b]
			1/1/1989	12/31/1993	385 [a]
			1/1/1994	12/31/2005	34.2 [a]

a. Urinalysis-based intake rates.
b. Lung count-based intake rates.

D.3.2 Dose Assignment

Doses to be assigned to individuals are calculated from the 95th-percentile intake rates [133]. 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 COWORKER DATA FIGURES

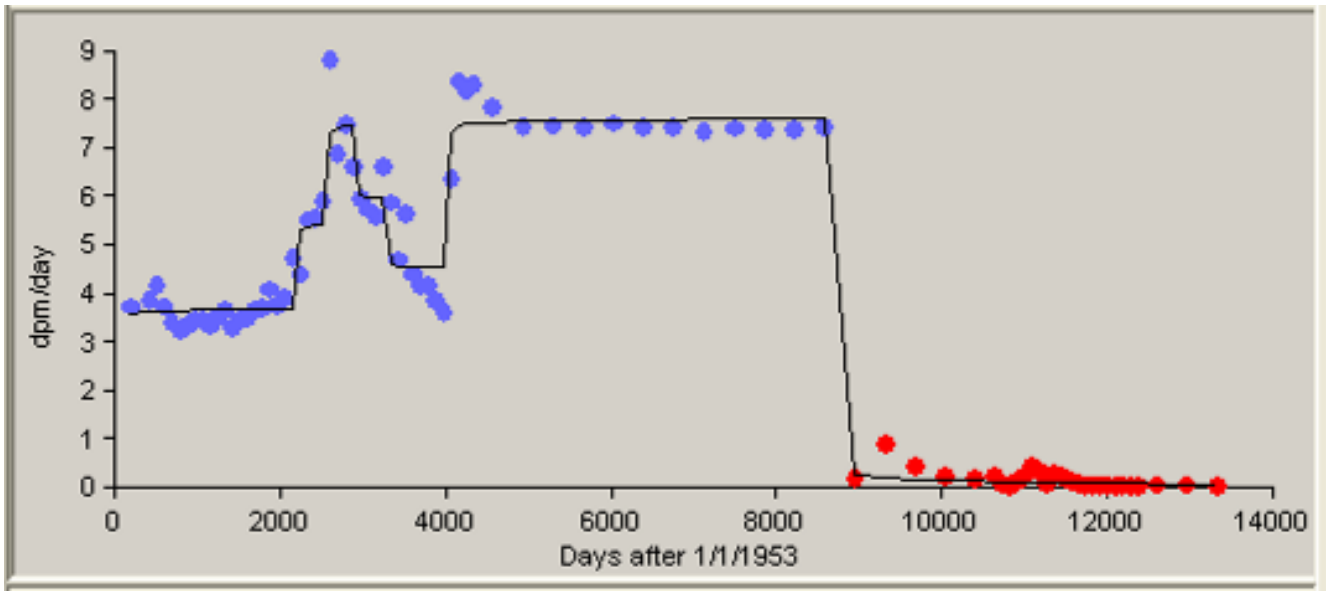

Figure D-1. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1976, 50th-percentile, Type F.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

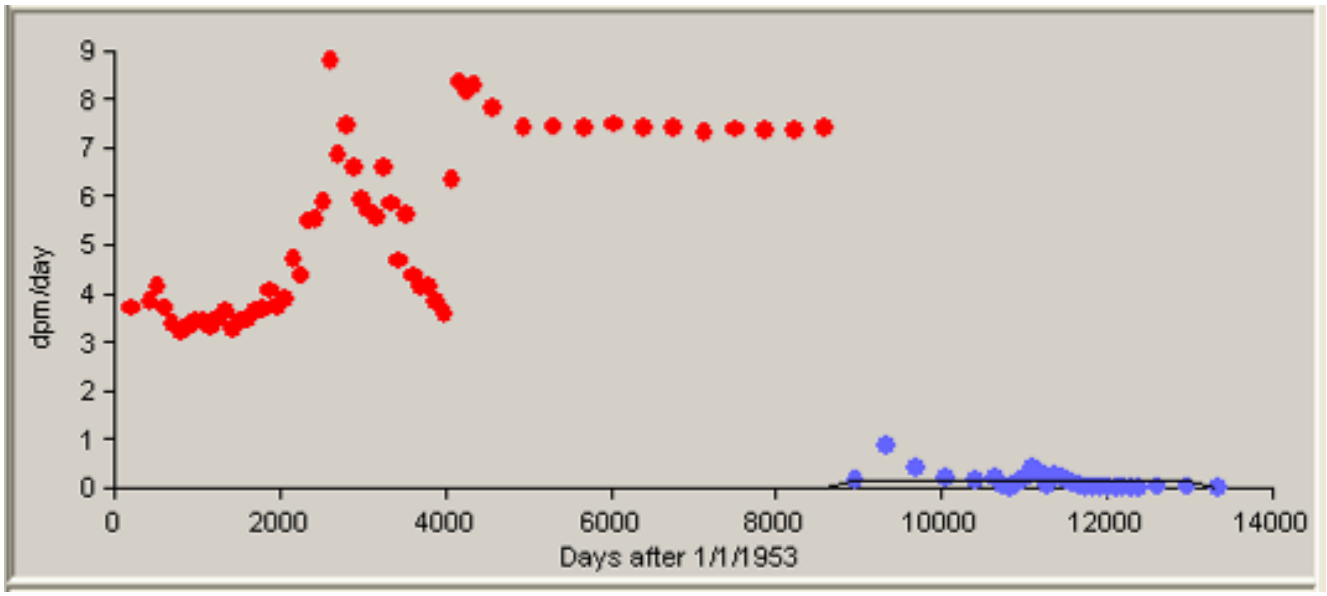

Figure D-2. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1977 to 12/31/1988, 50th-percentile, Type F.

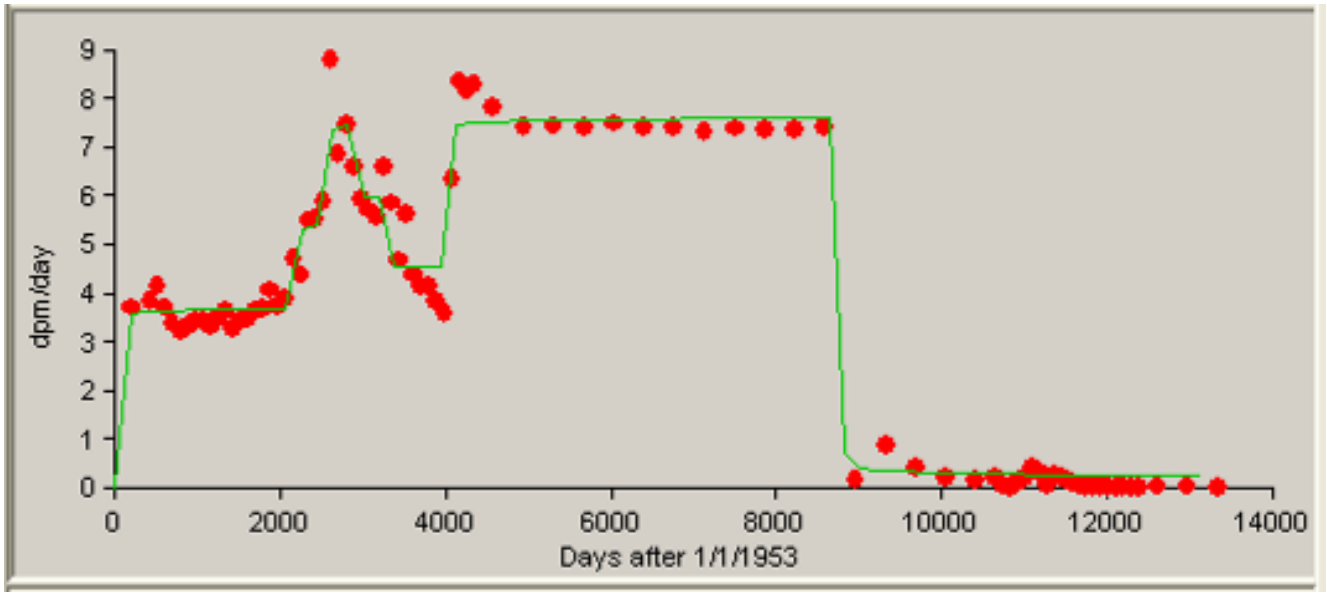

Figure D-3. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots) from all intakes 1/1/1953 to 12/31/1988, 50th-percentile, Type F.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

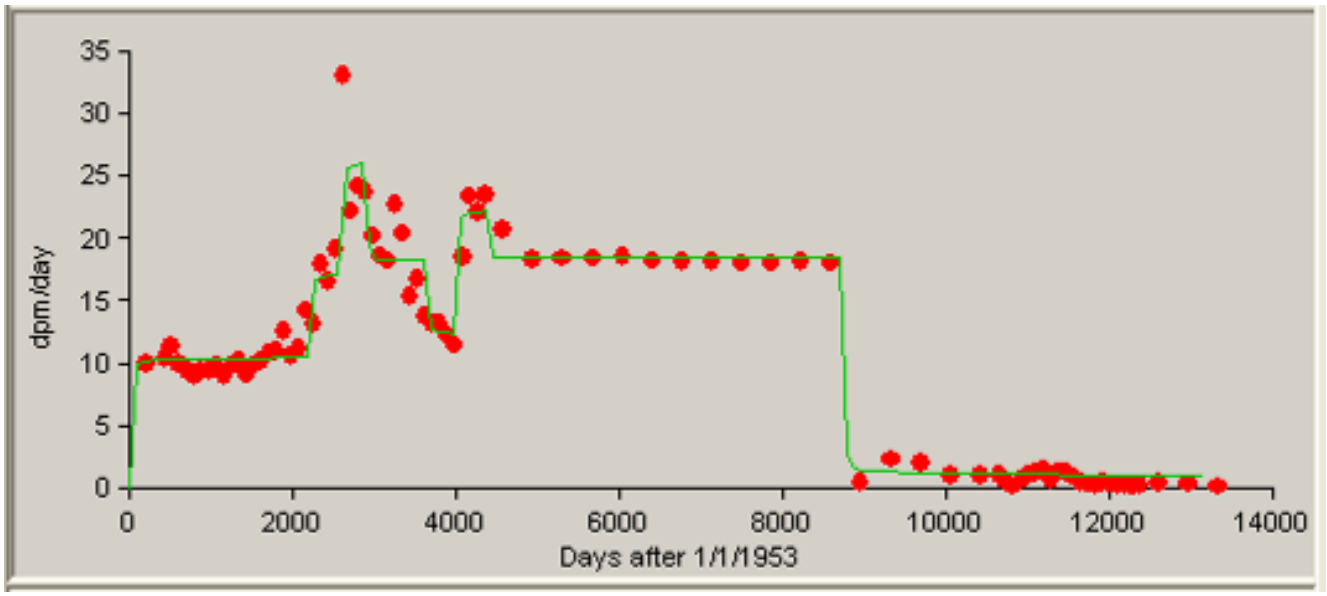

Figure D-4. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1988, 84th-percentile, Type F.

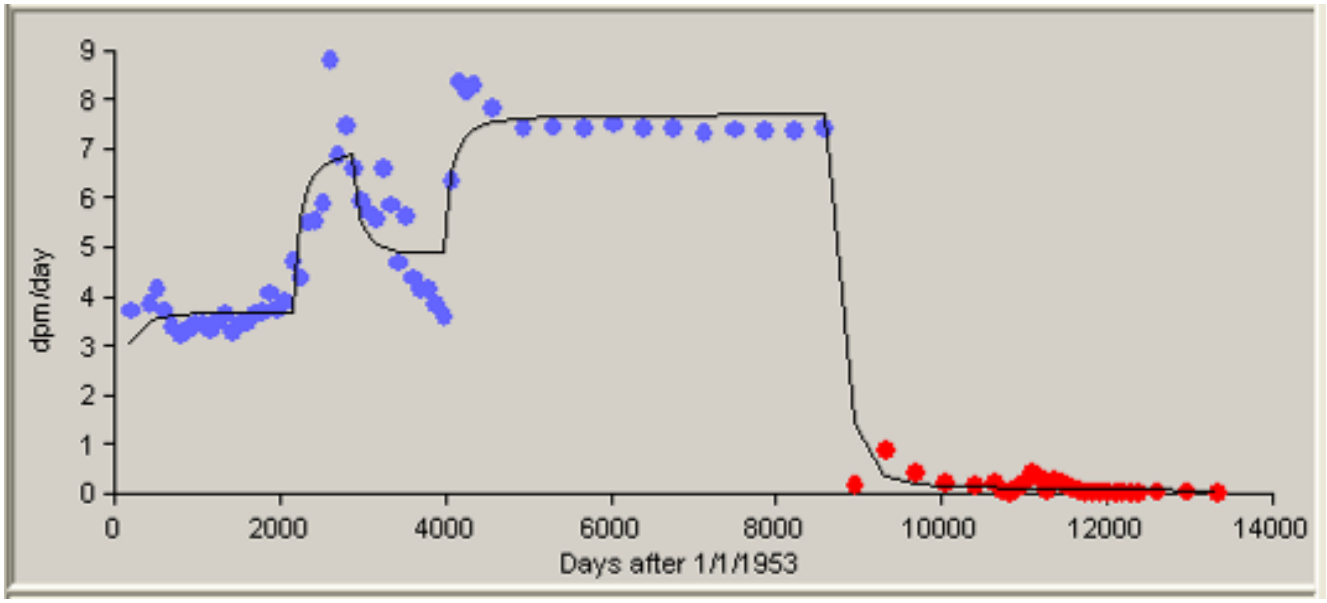

Figure D-5. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1976, 50th-percentile, Type M.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

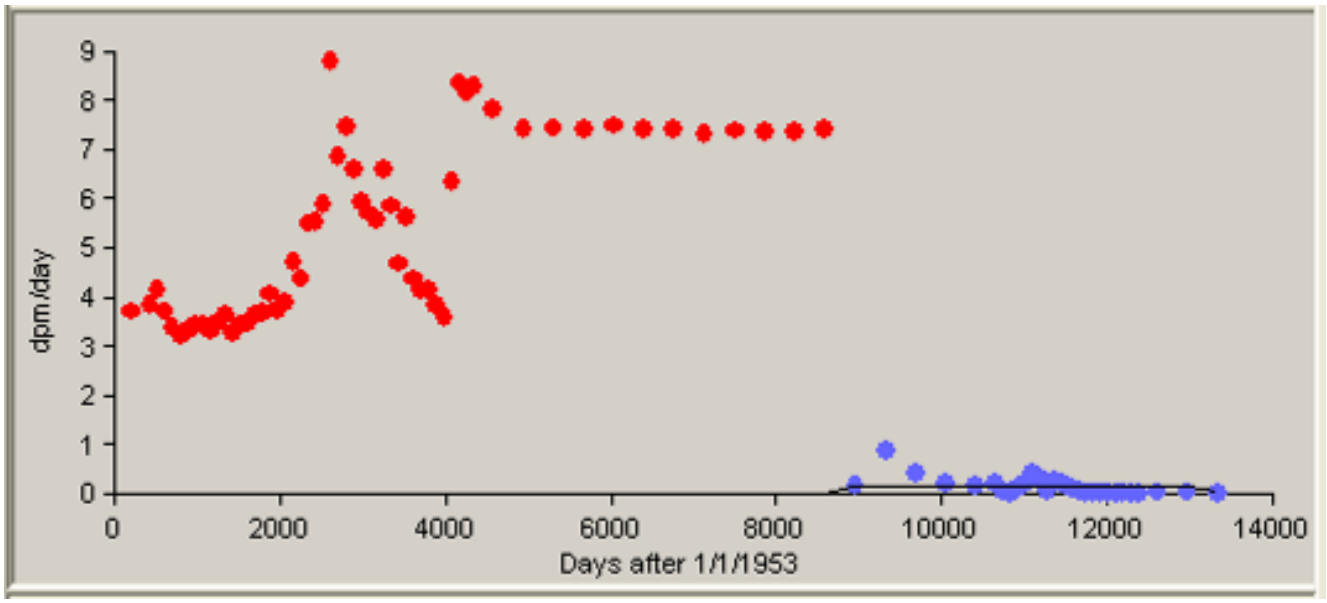

Figure D-6. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1977 to 12/31/1988, 50th-percentile, Type M

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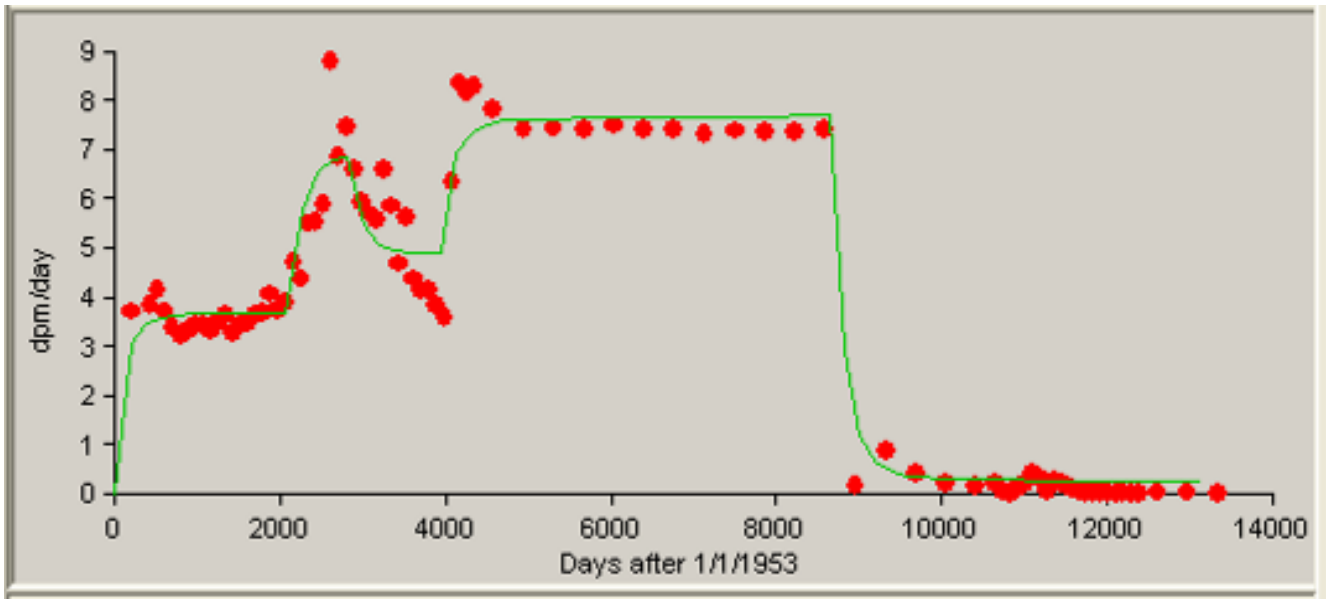

Figure D-7. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots) from all intakes 1/1/1953 to 12/31/1988, 50th-percentile, Type M.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

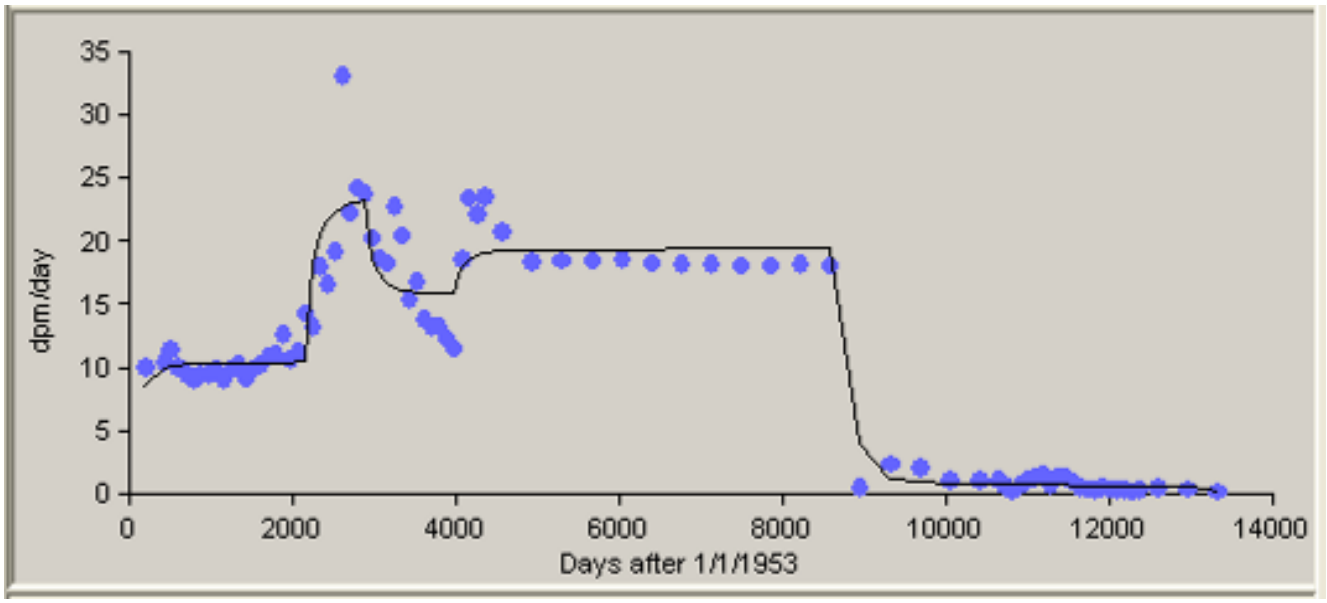

Figure D-8. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1988, 84th-percentile, Type M.

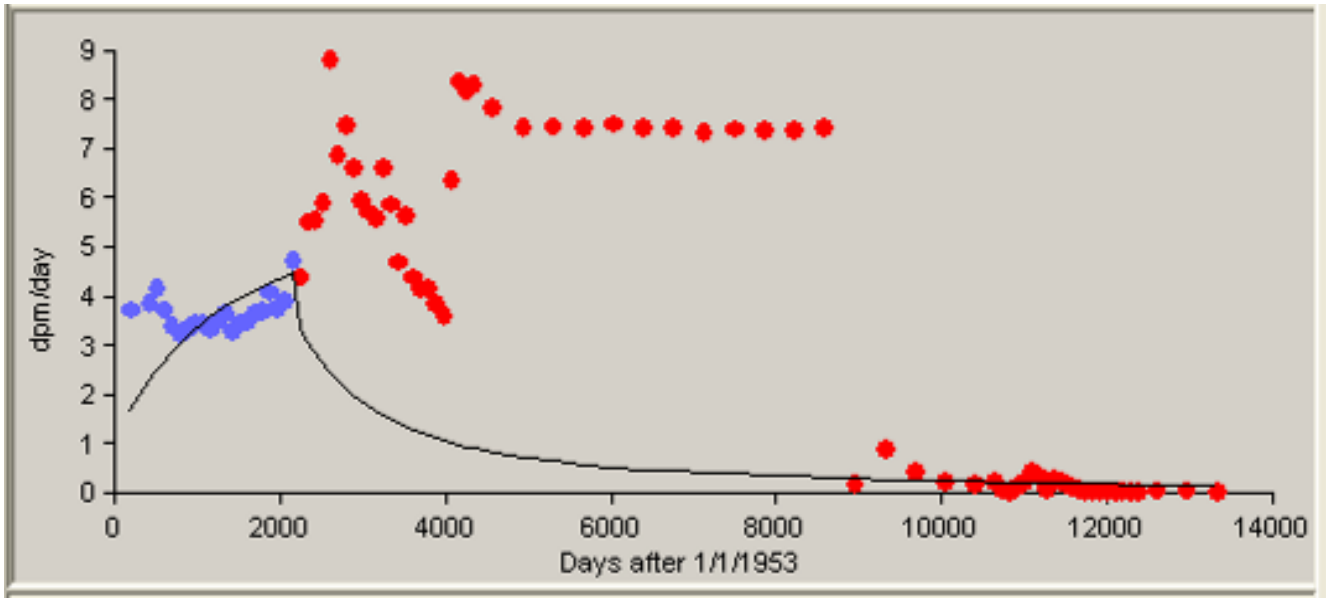

Figure D-9. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1958, 50th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

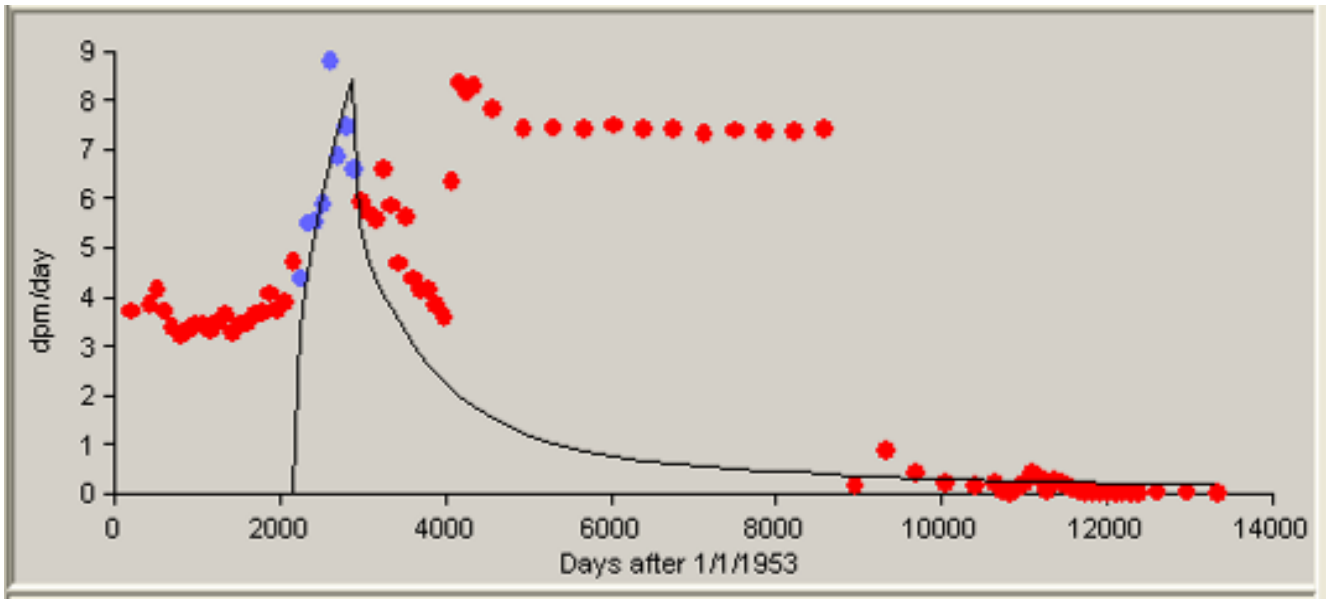

Figure D-10. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1959 to 12/31/1960, 50th-percentile, Type S.

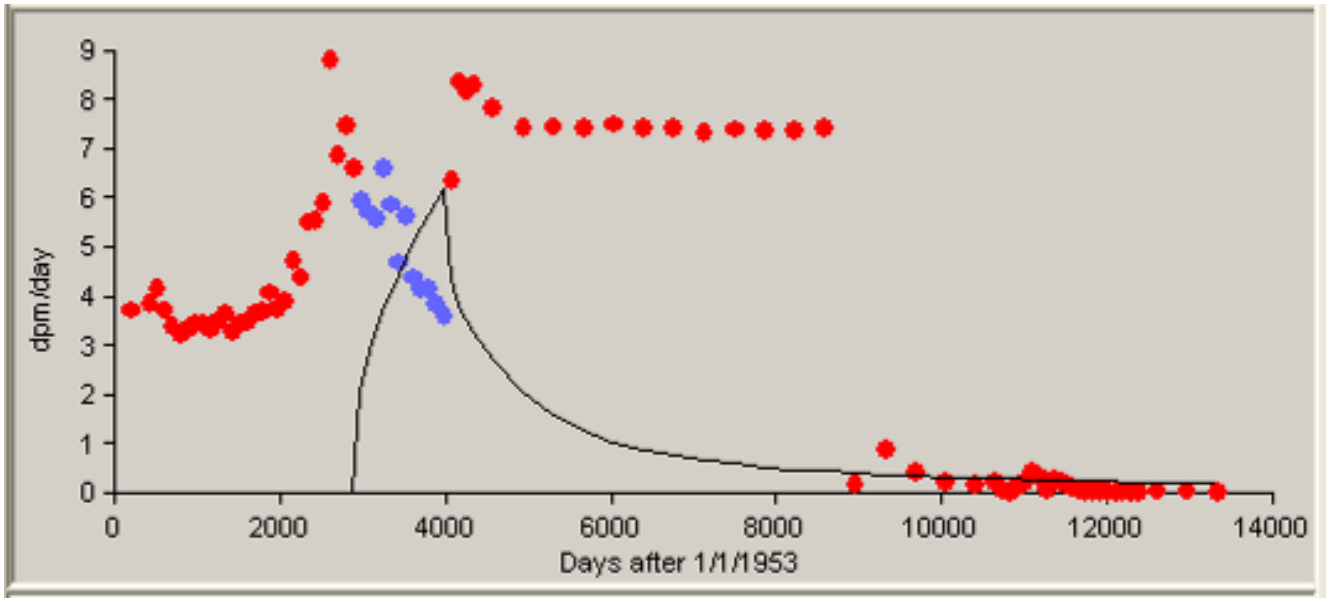

Figure D-11. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1961 to 12/31/1963, 50th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

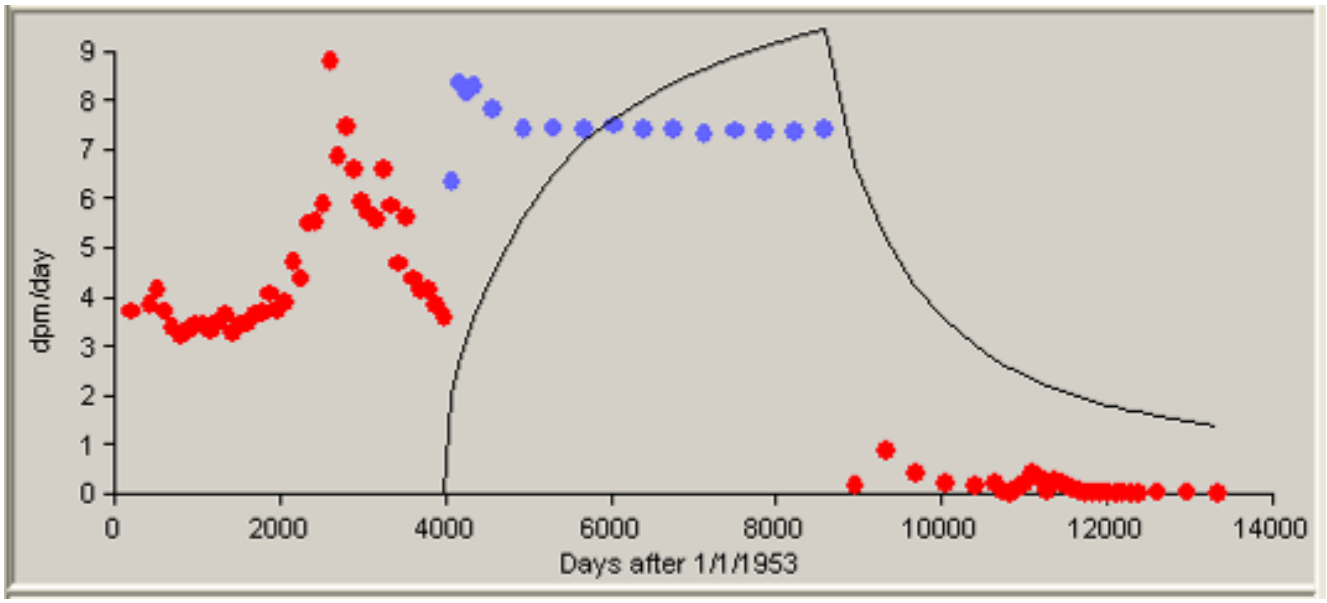

Figure D-12. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1964 to 12/31/1976, 50th-percentile, Type S.

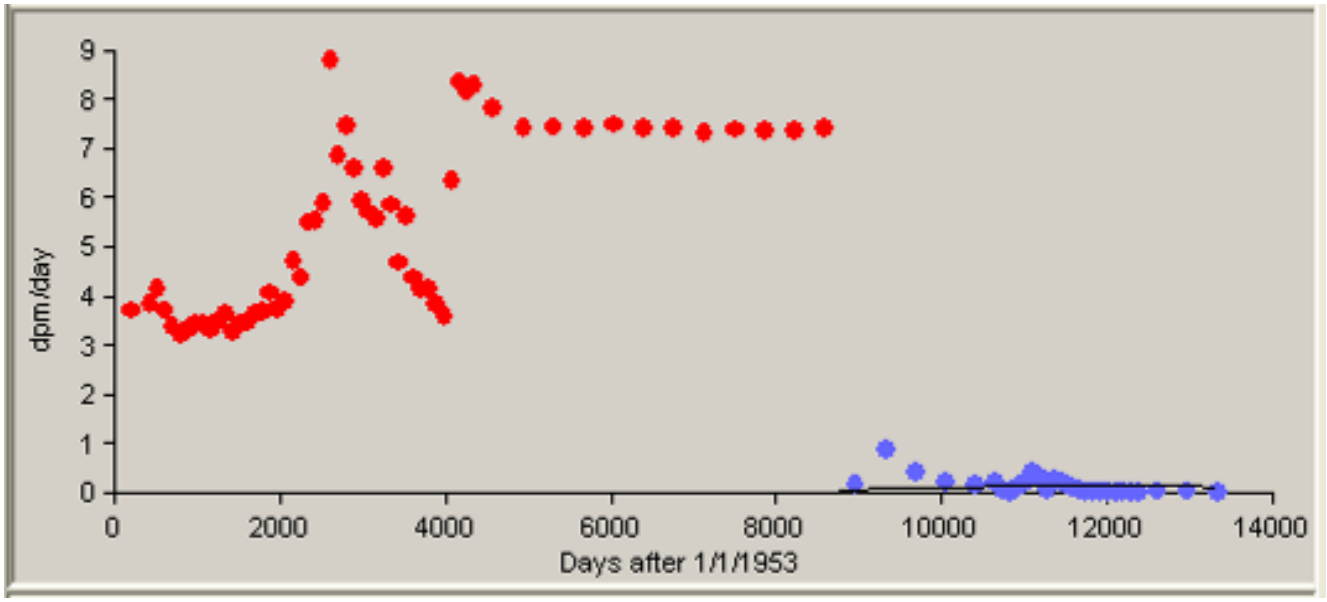

Figure D-13. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1977 to 12/31/1988, 50th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

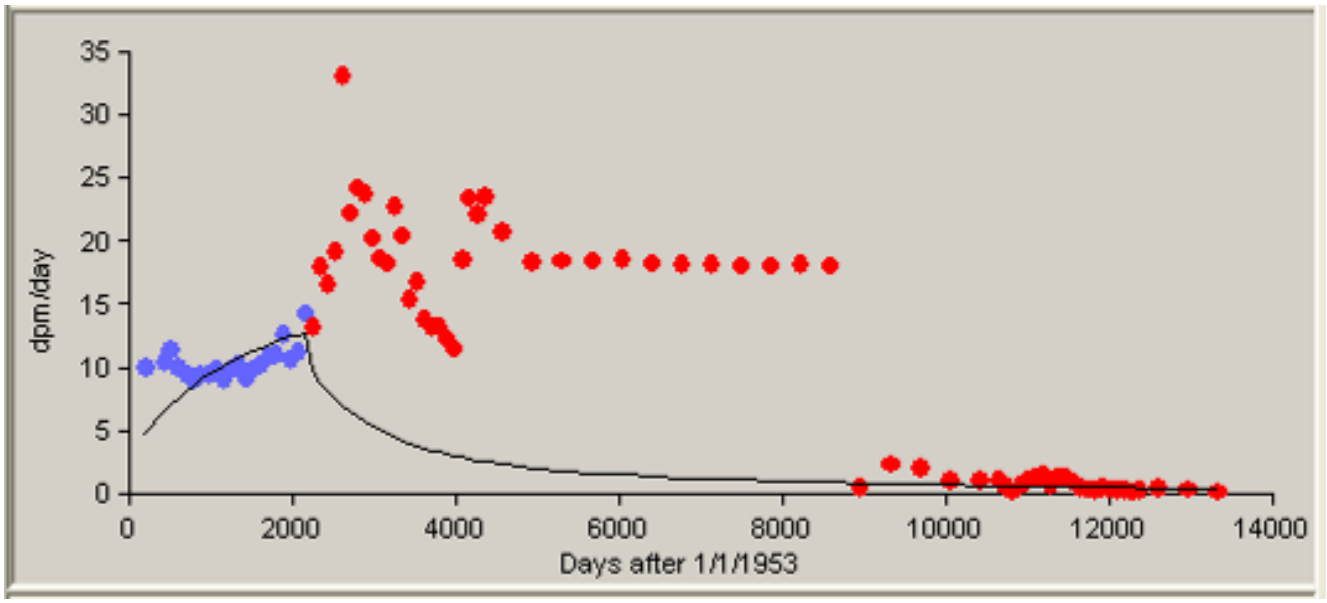

Figure D-14. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1953 to 12/31/1958, 84th-percentile, Type S.

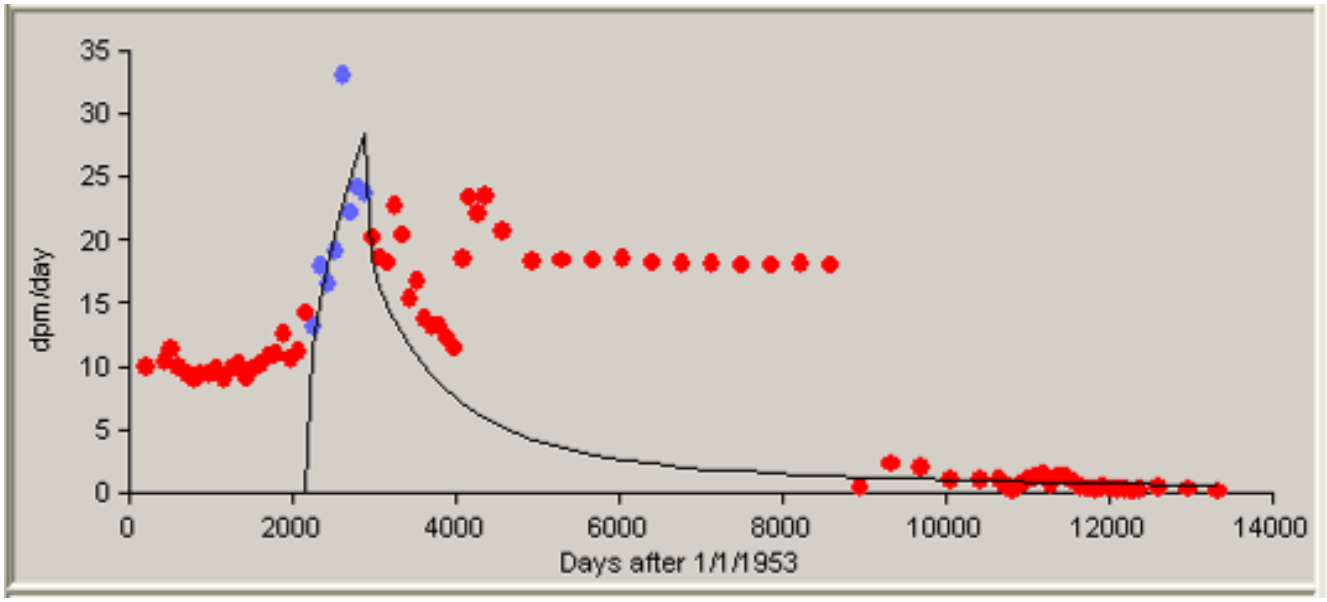

Figure D-15. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1959 to 12/31/1960, 84th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

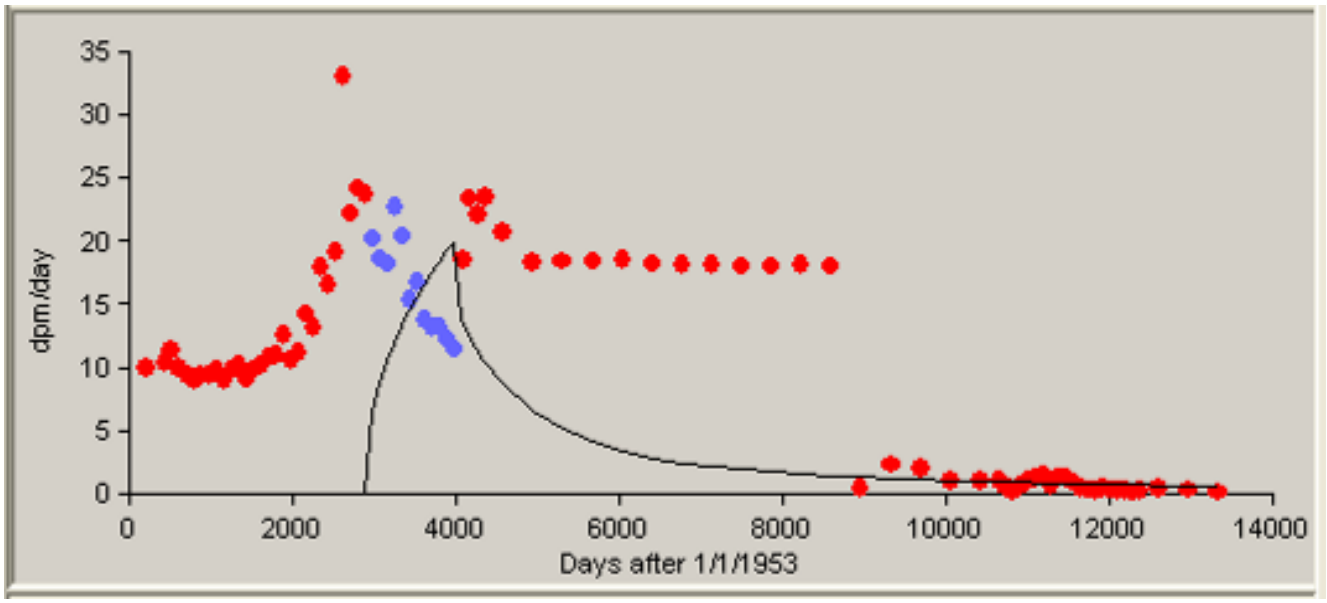

Figure D-16. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1961 to 12/31/1963, 84th-percentile, Type S.

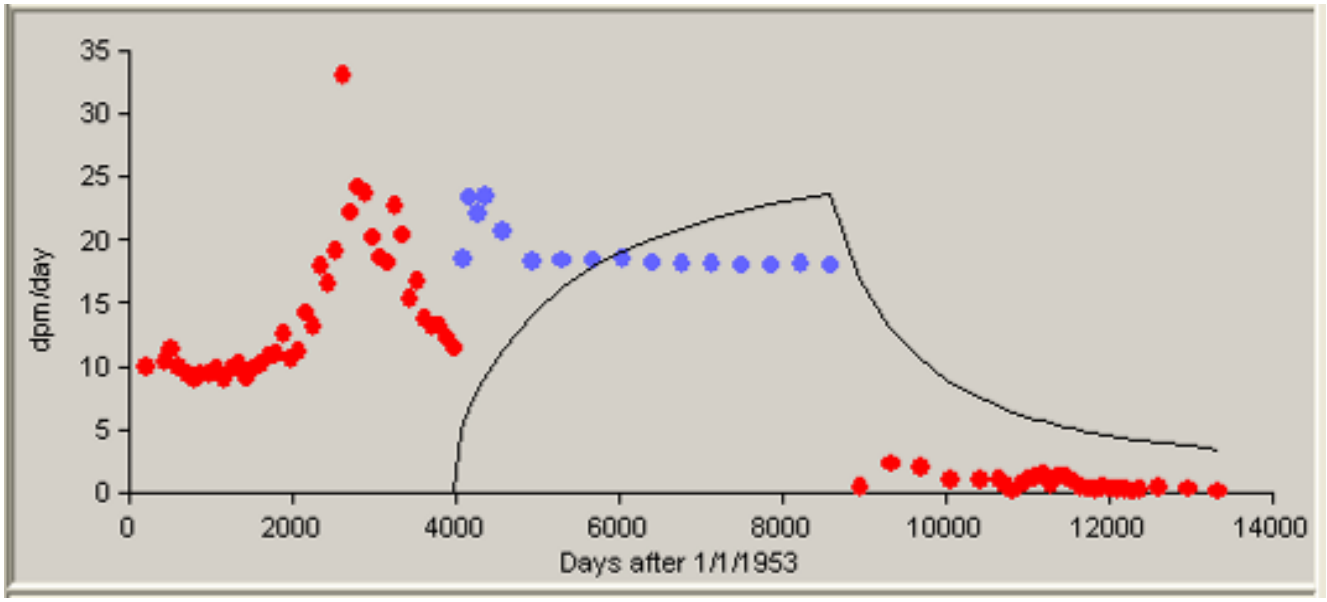

Figure D-17. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1964 to 12/31/1976, 84th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

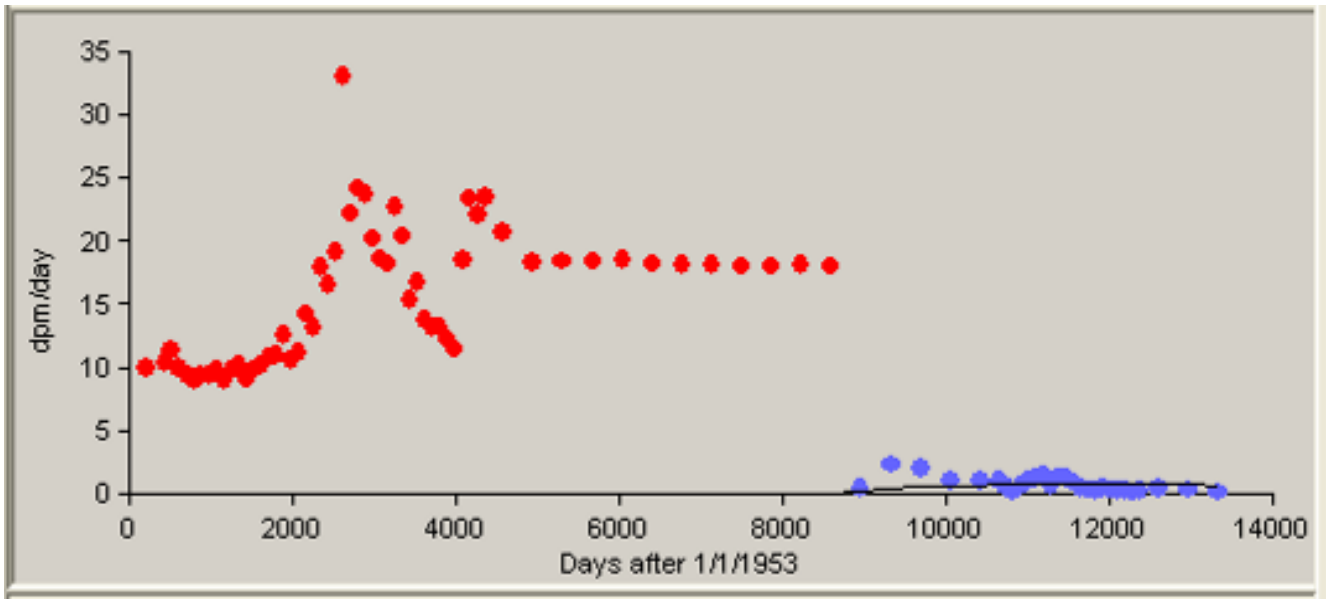

Figure D-18. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots), 1/1/1977 to 12/31/1988, 84th-percentile, Type S.

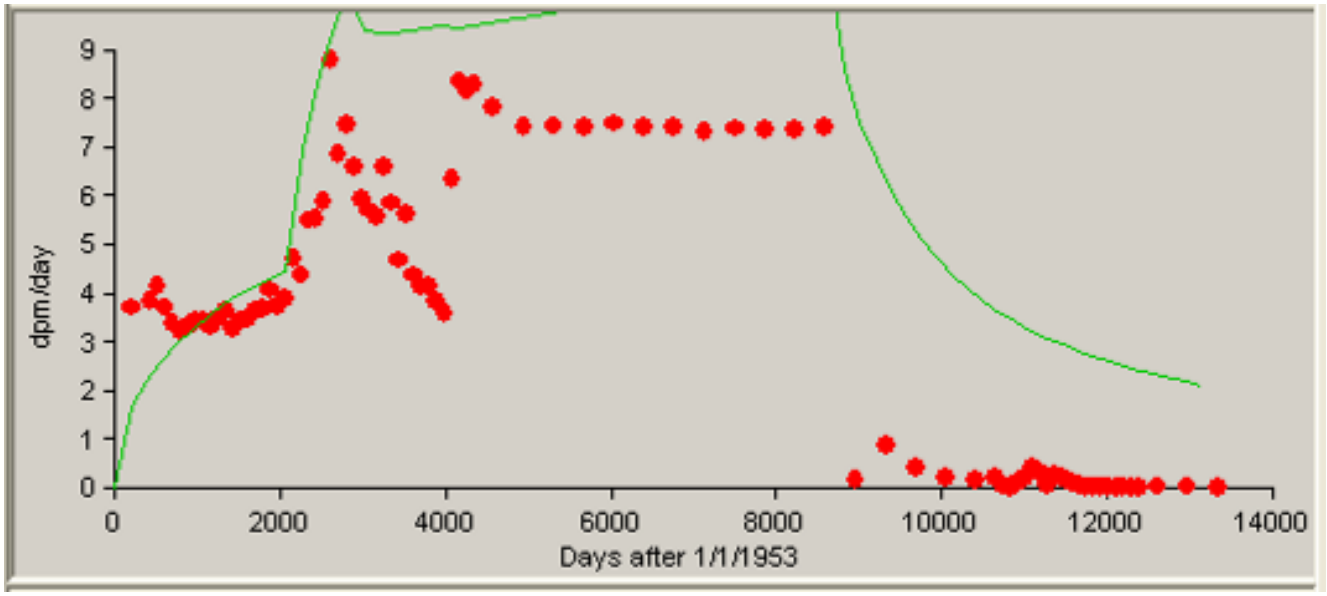

Figure D-19. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots) from all intakes 1/1/1953 to 12/31/1988, 50th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

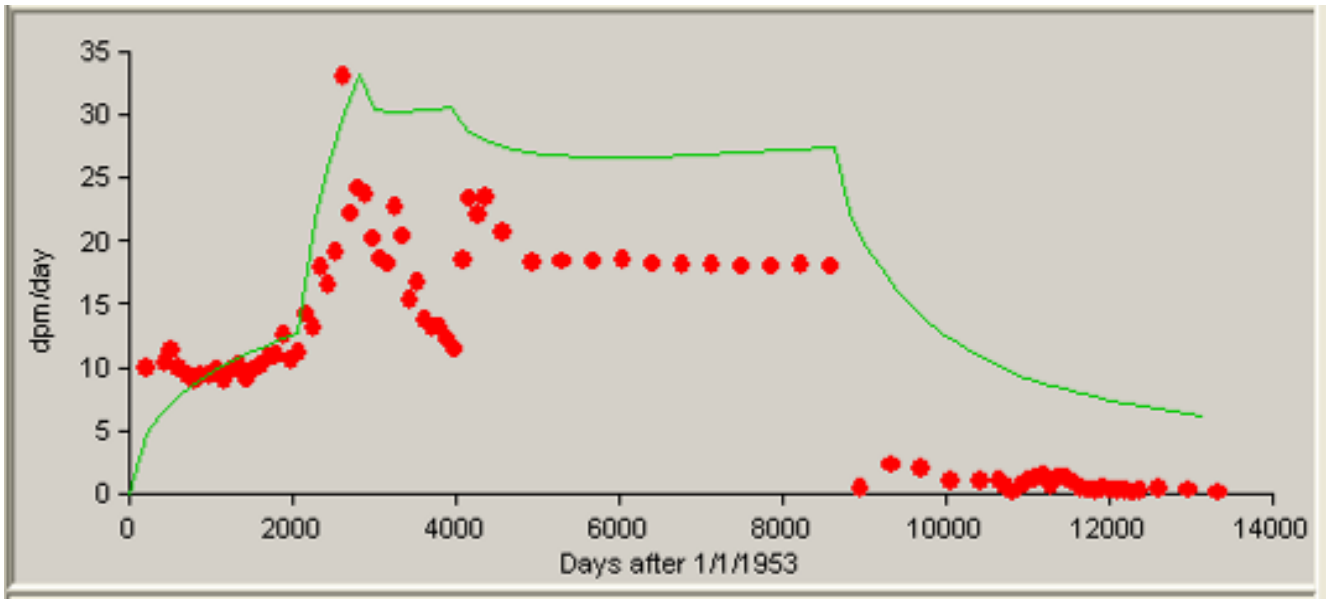

Figure D-20. Predicted uranium bioassay results (line) calculated using IMBA-derived uranium intake rates compared with measured uranium-in-urine results (dots) from all intakes 1/1/1953 to 12/31/1988, 84th-percentile, Type S.

Table D-7. IMBA-derived uranium intake rates (dpm/day).

Years	Type F			Type M			Type S		
	50%	84%	GSD	50%	84%	GSD	50%	84%	GSD
1953–1958	13.37	37.91	2.84	54.75	154.8	2.83	936.9	2,676	2.86
1959	19.7	61.99	3.15	102.7	347.5	3.38	2,768	9,300	3.36
1960	27.23	94.74	3.48	102.7	347.5	3.38	2,768	9,300	3.36
1961	21.62	65.97	3.05	71.85	234.2	3.26	1,680	5,438	3.24
1962	16.27	65.97	4.05	71.85	234.2	3.26	1,680	5,438	3.24
1963	16.27	44.36	2.73	71.85	234.2	3.26	1,680	5,438	3.24
1964	27.26	80.39	2.95	112.8	284.2	2.52	1,630	4,086	2.51
1965–1976	27.26	66.26	2.43	112.8	284.2	2.52	1,630	4,086	2.51
1977–1988	0.597	3.004	5.03	2.443	6.263	2.56	28.6	154.3	5.40

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

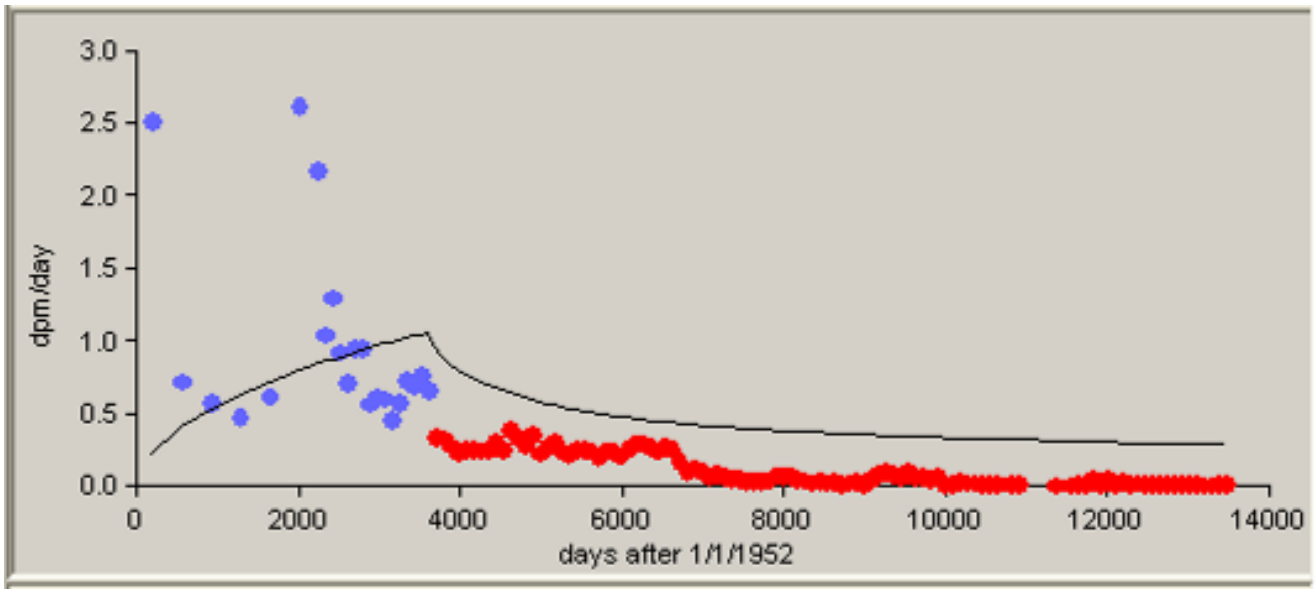

Figure D-21. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/1961, 50th-percentile, Type M.

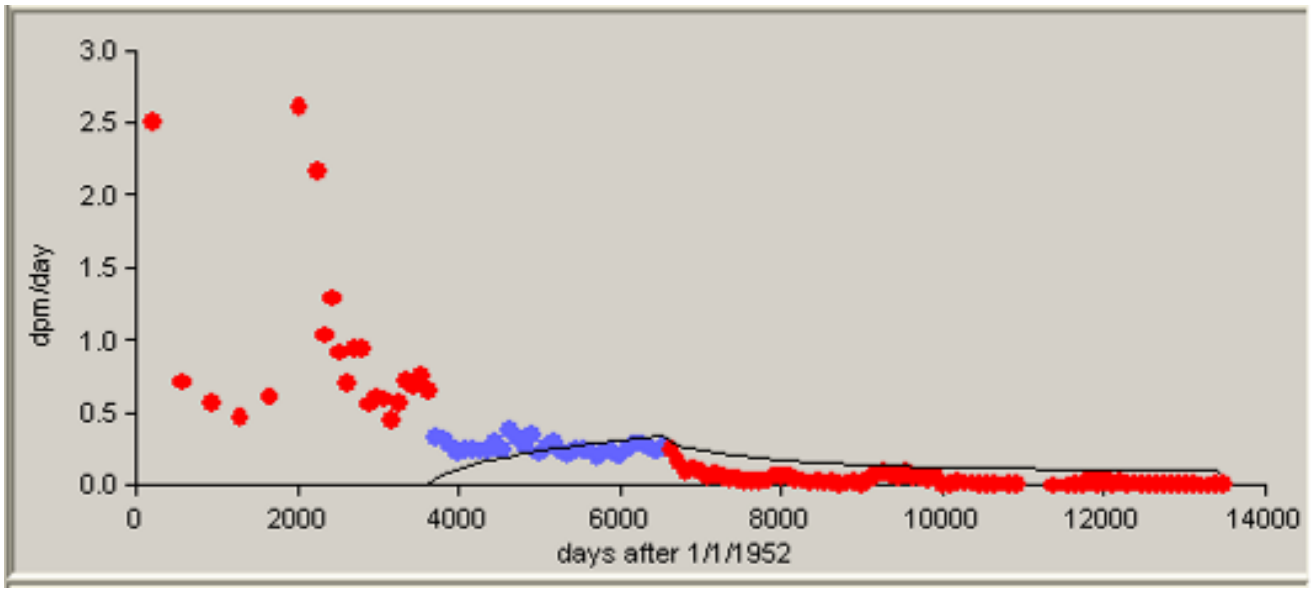

Figure D-22. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1962 to 12/31/1969, 50th-percentile, Type M.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

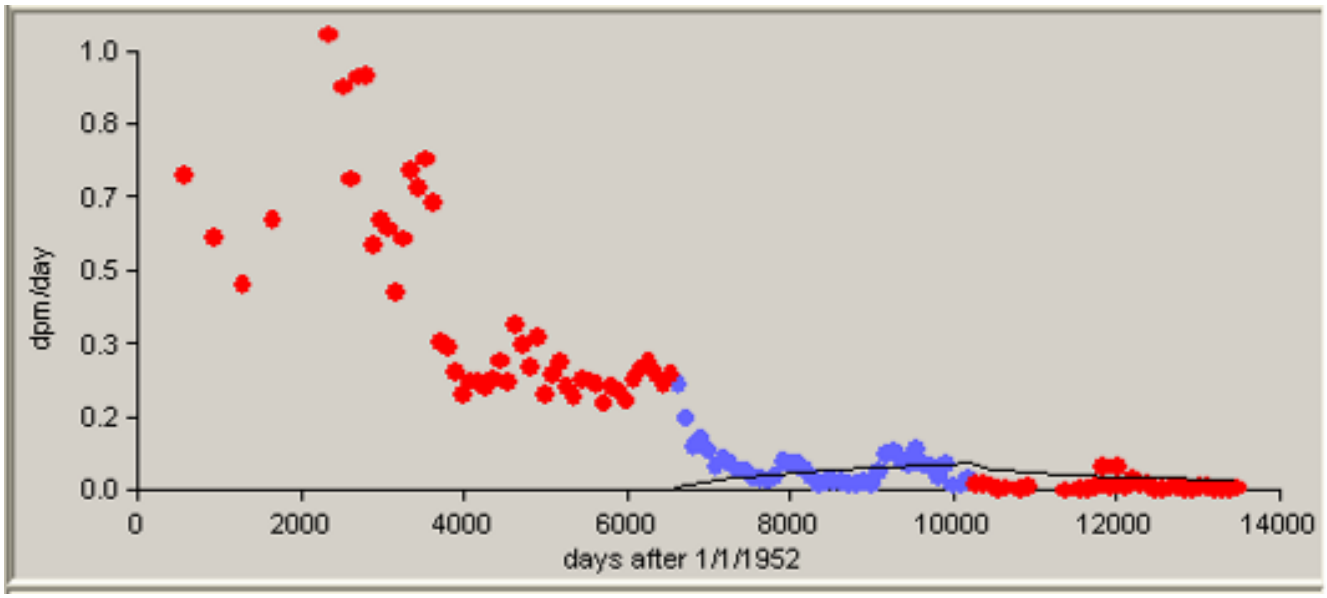

Figure D-23. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1970 to 12/31/1979, 50th-percentile, Type M.

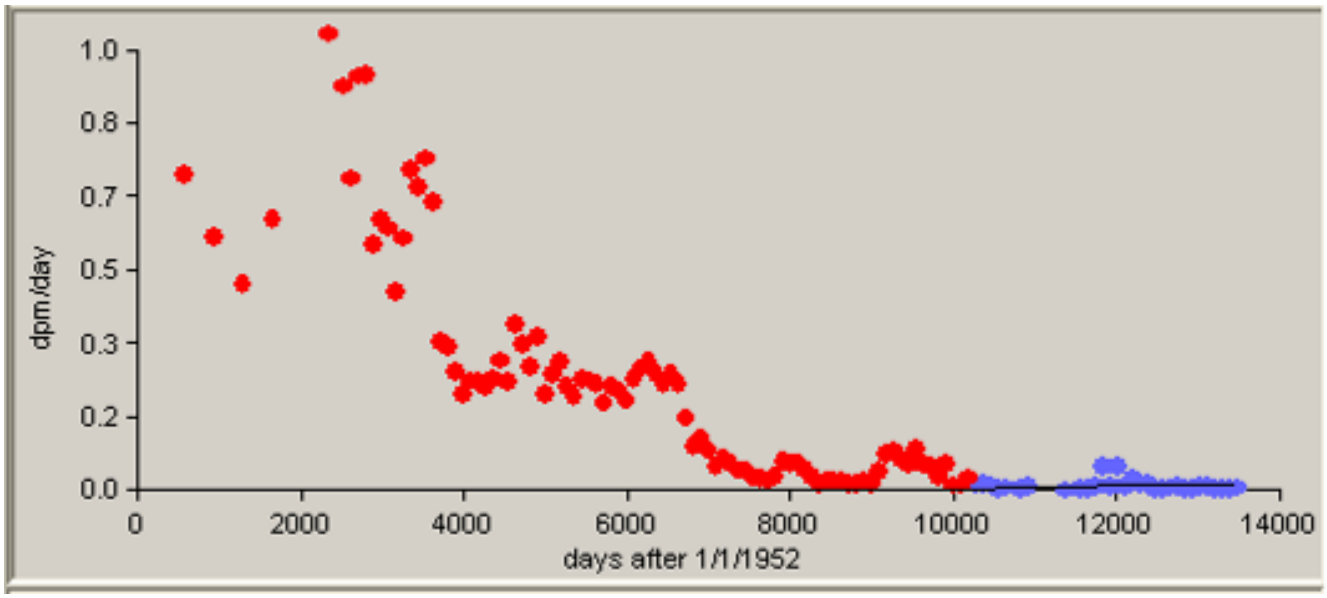

Figure D-24. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1980 to 12/31/1988, 50th-percentile, Type M.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

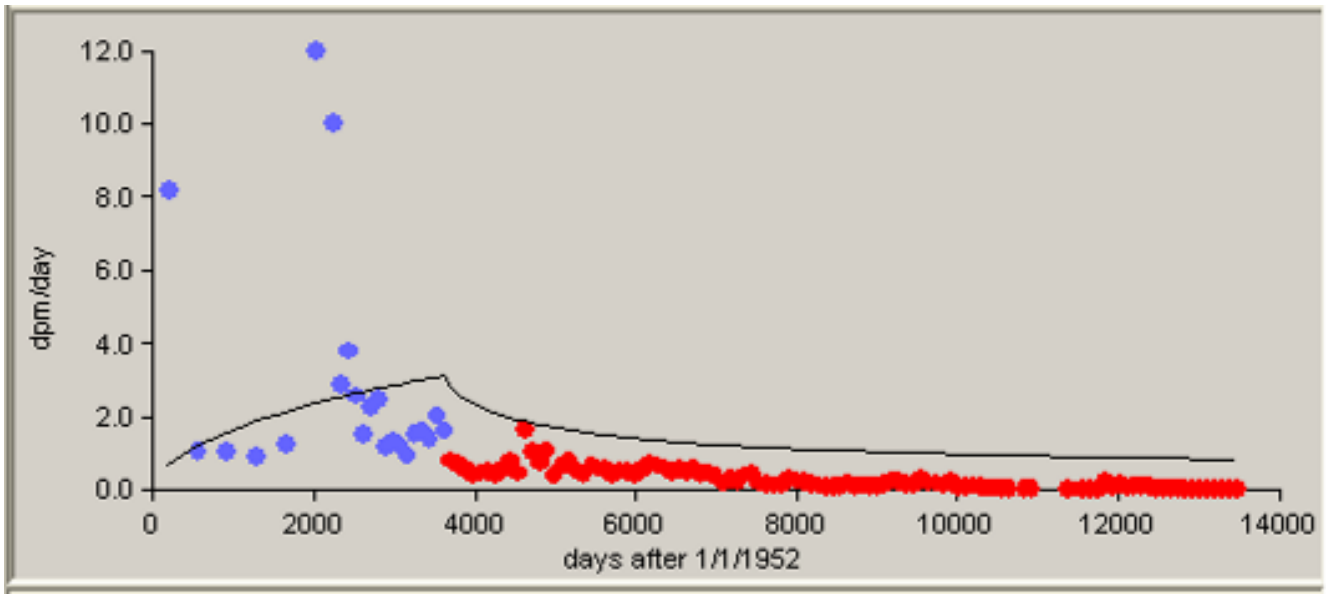

Figure D-25. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/1961, 84th-percentile, Type M.

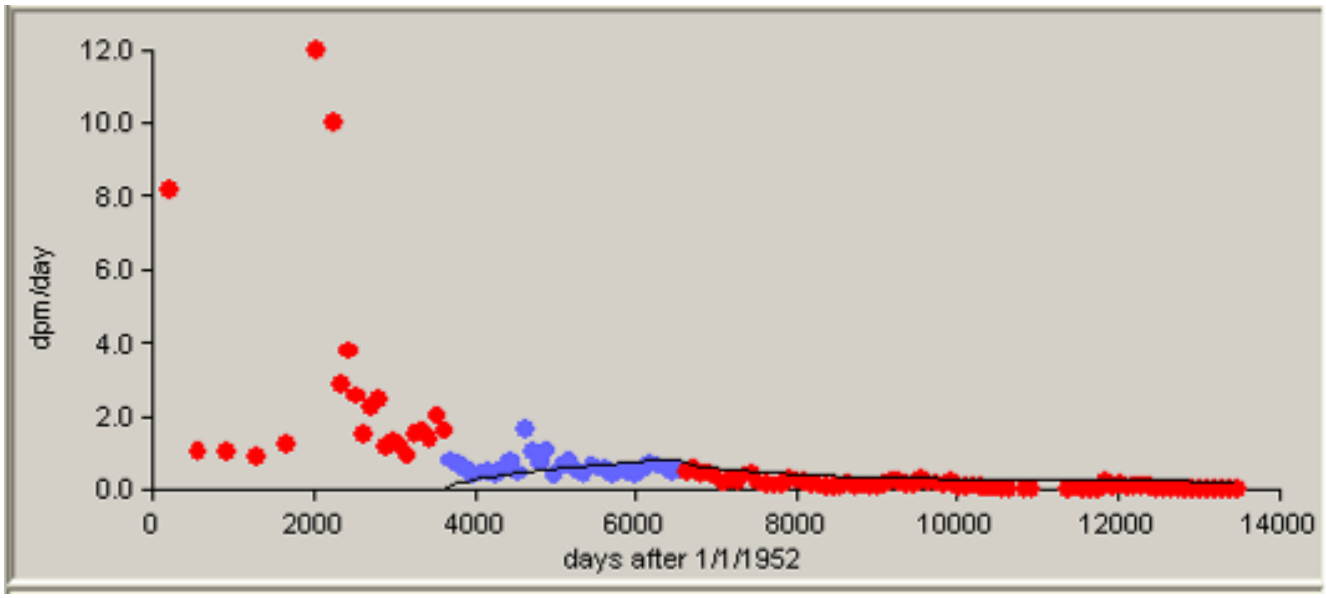

Figure D-26. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1962 to 12/31/1969, 84th-percentile, Type M.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

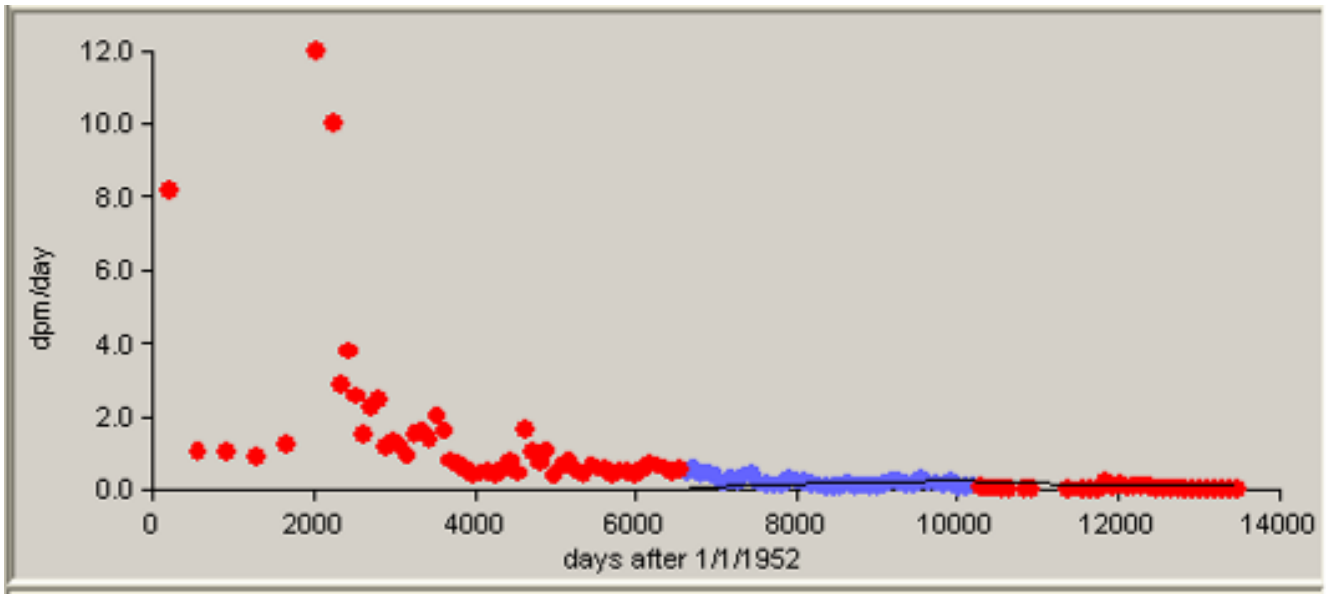

Figure D-27. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1970 to 12/31/1979, 84th-percentile, Type M.

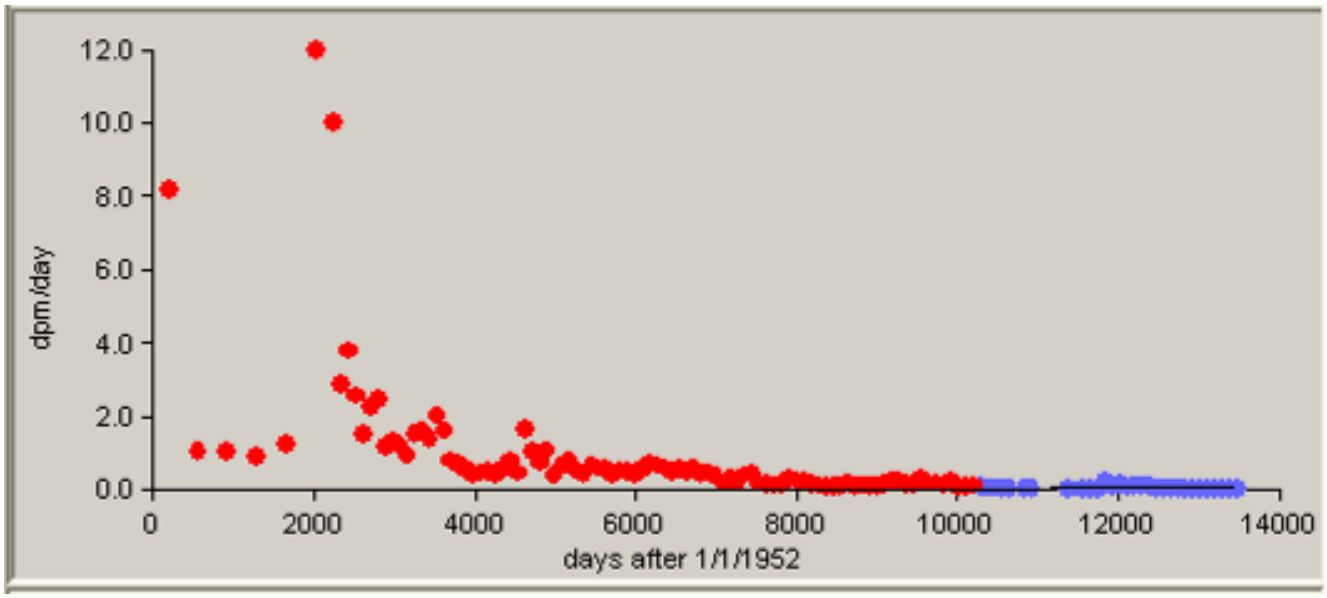

Figure D-28. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1980 to 12/31/1988, 84th-percentile, Type M.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

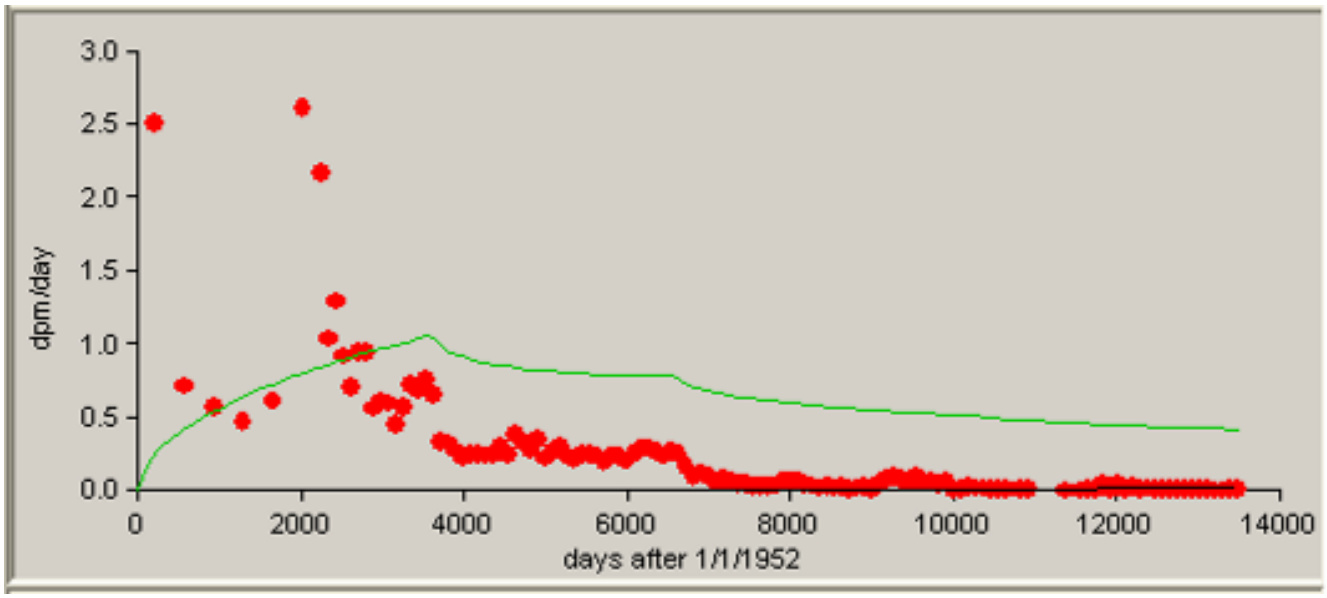

Figure D-29. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), from all intakes 1/1/1952 to 12/31/1988, 50-percentile, Type M.

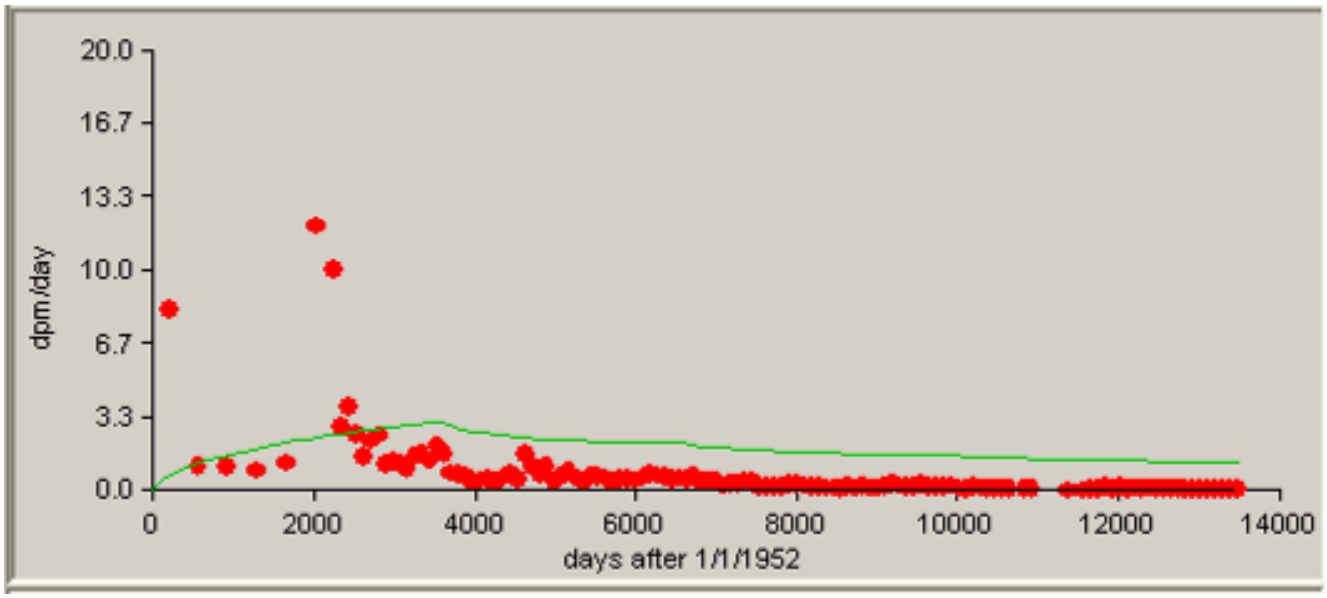

Figure D-30. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), from all intakes 1/1/1952 to 12/31/1988, 84th-percentile, Type M.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

Table D-8. IMBA-derived plutonium/americium intake rates, Type M.

Year	Plutonium urinalysis-based results, dpm/day			Americium lung count-based results, 50th percentile	
	Pu 50%	Pu 84%	GSD	Am, pCi/day	Pu, dpm/day
1952–1961	121	357.3	2.95		
1962–1969	43.5	106.5	2.45		
1970–1971	7.05	29.82	4.23		
1972–1976	7.05	29.82	4.23	0.387	178
1977–1979	7.05	29.82	4.23	0.280	129
1980–1982	1.622	8.907	5.49	0.280	129
1983–1988	1.622	8.907	5.49	0.124	57.3

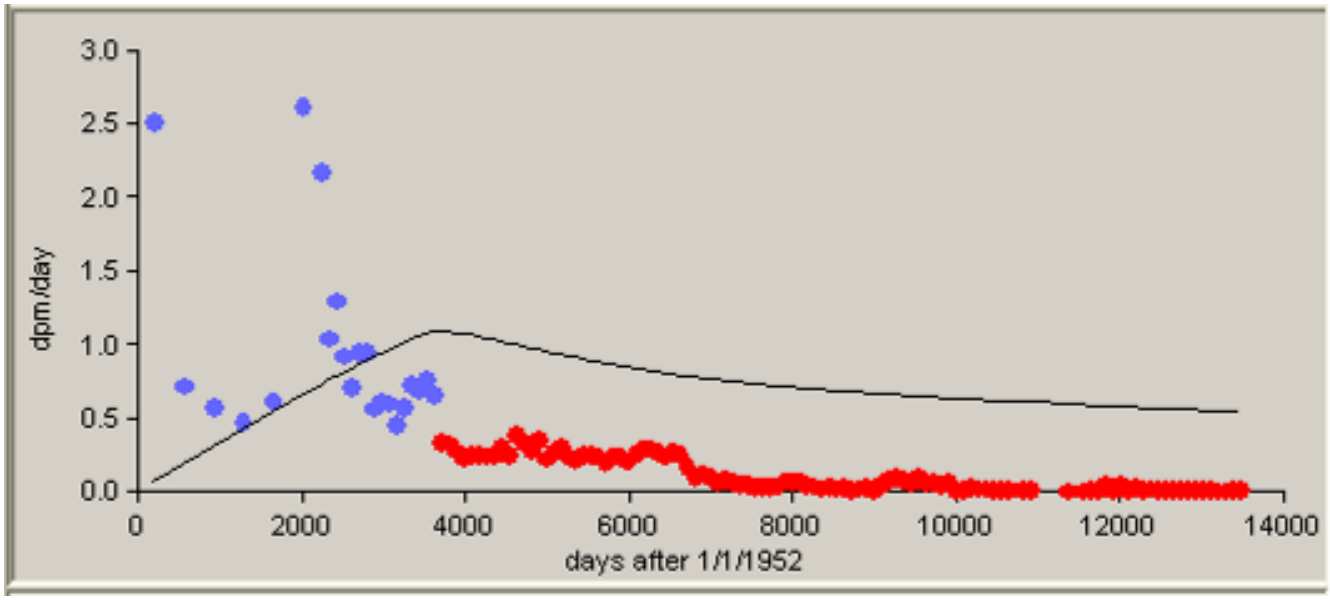

Figure D-31. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/1961, 50th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

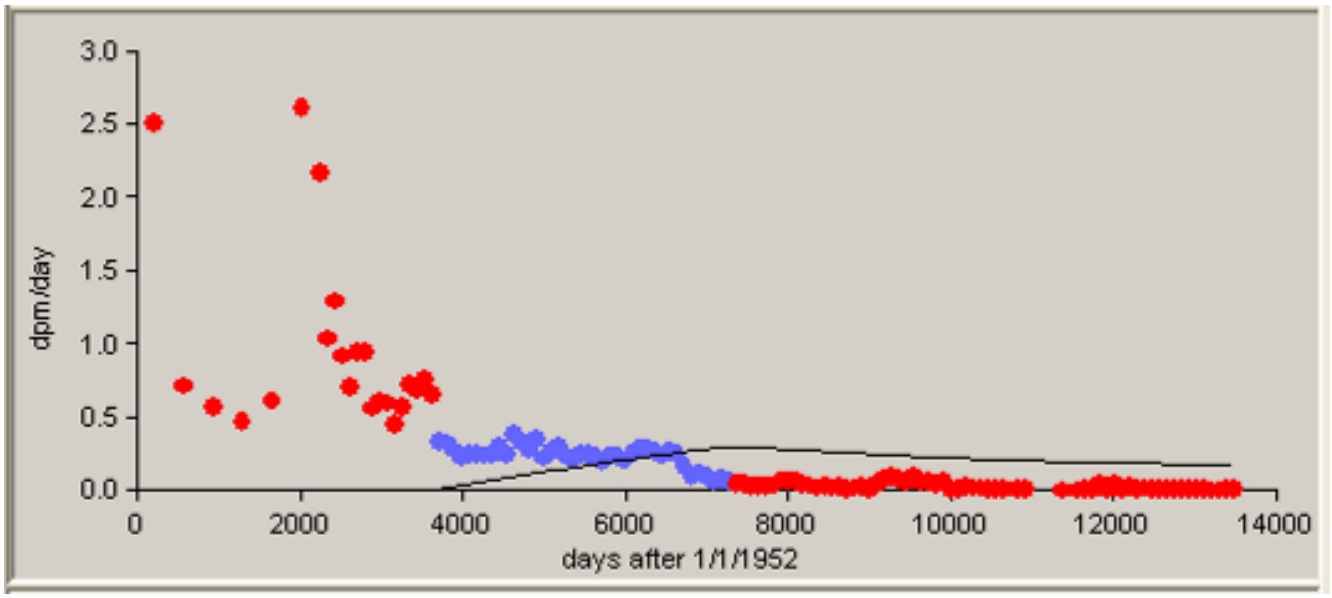

Figure D-32. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1962 to 12/31/1971, 50th-percentile, Type S.

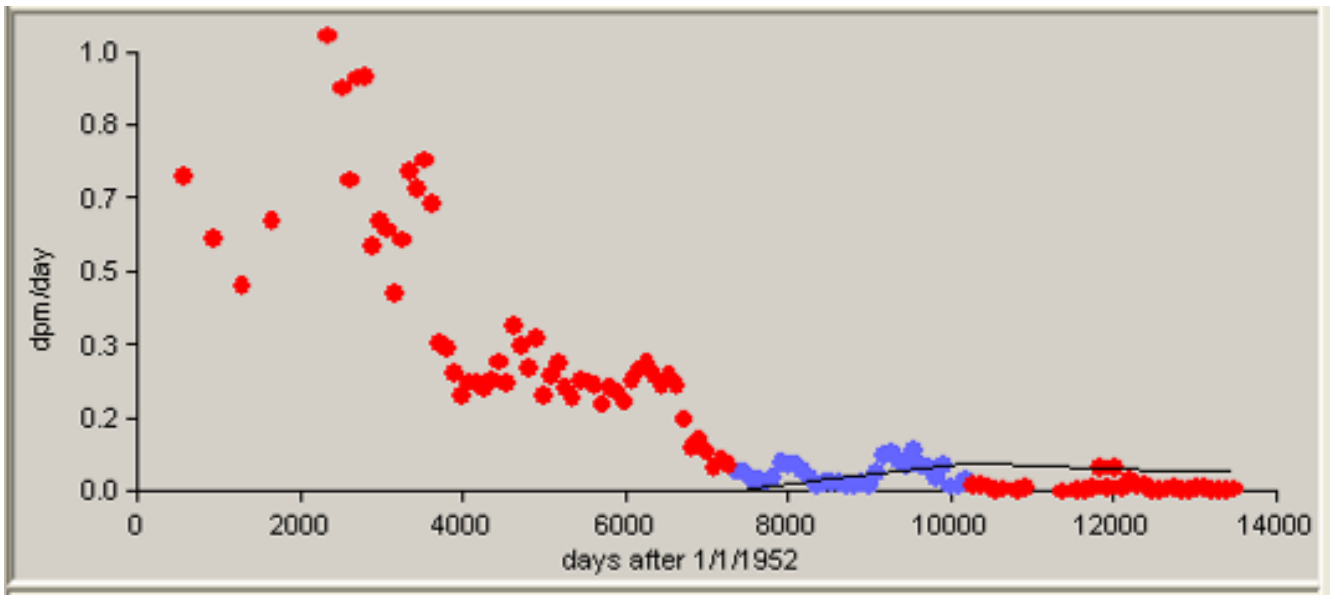

Figure D-33. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1972 to 12/31/1979, 50th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

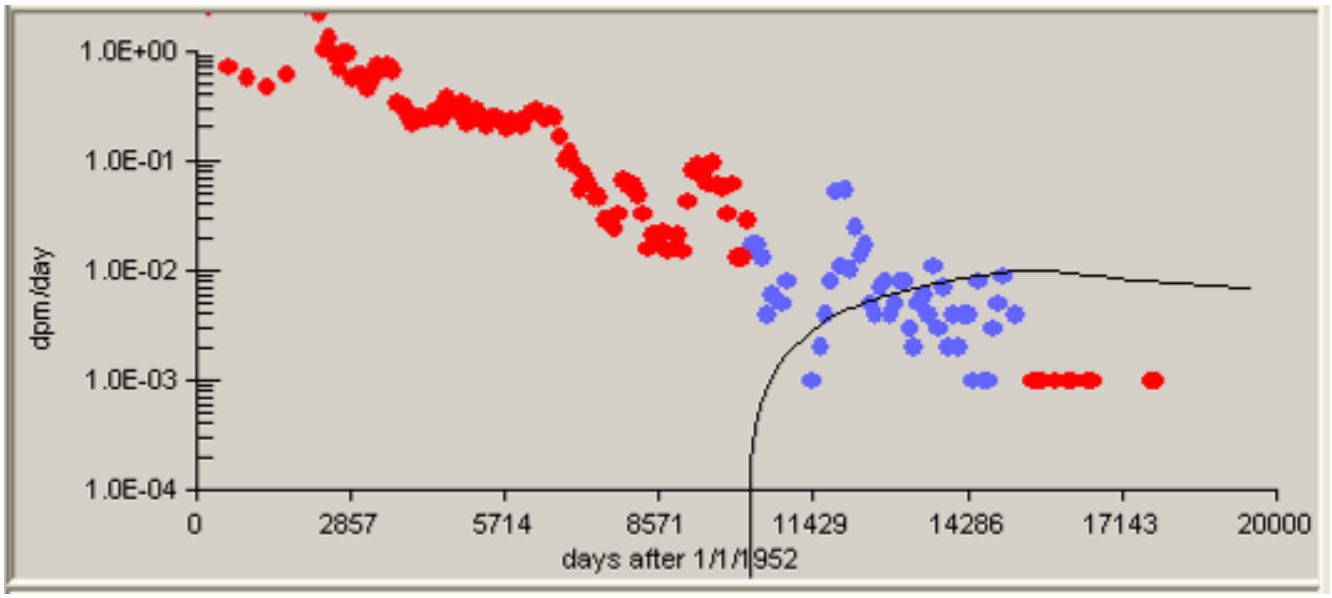

Figure D-34. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1980 to 12/31/1993, 50th-percentile, Type S.

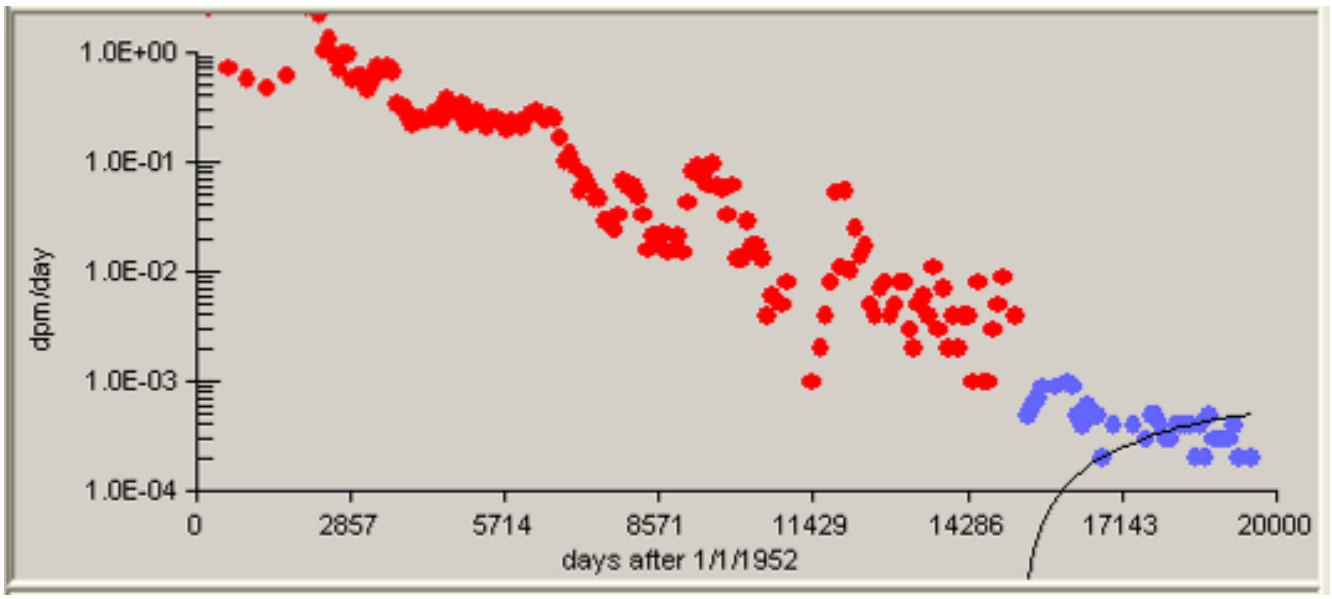

Figure D-35. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1994 to 12/31/2005, 50th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

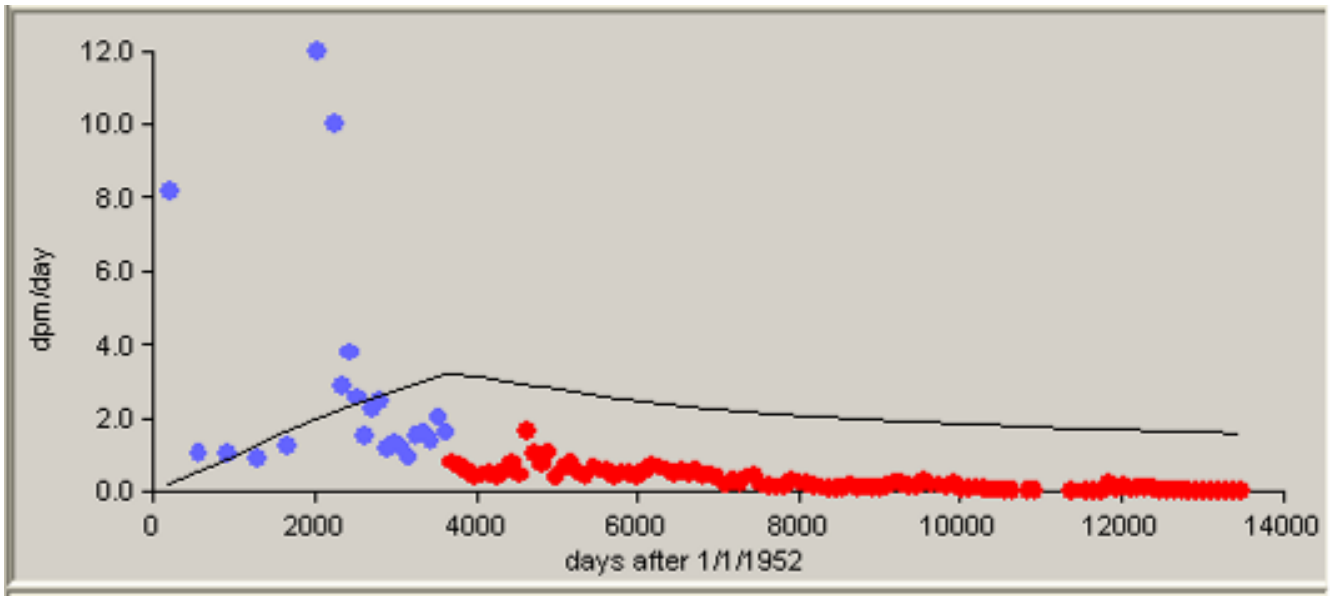

Figure D-36. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/1961, 84th-percentile, Type S.

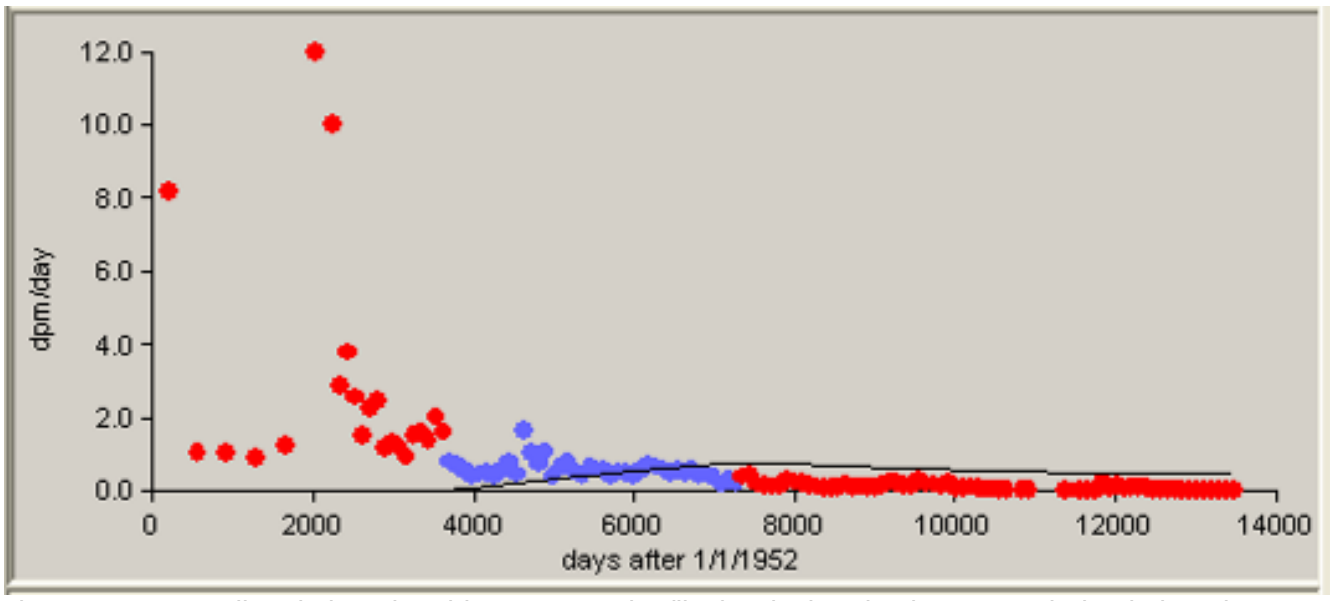

Figure D-37. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1962 to 12/31/1971, 84th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

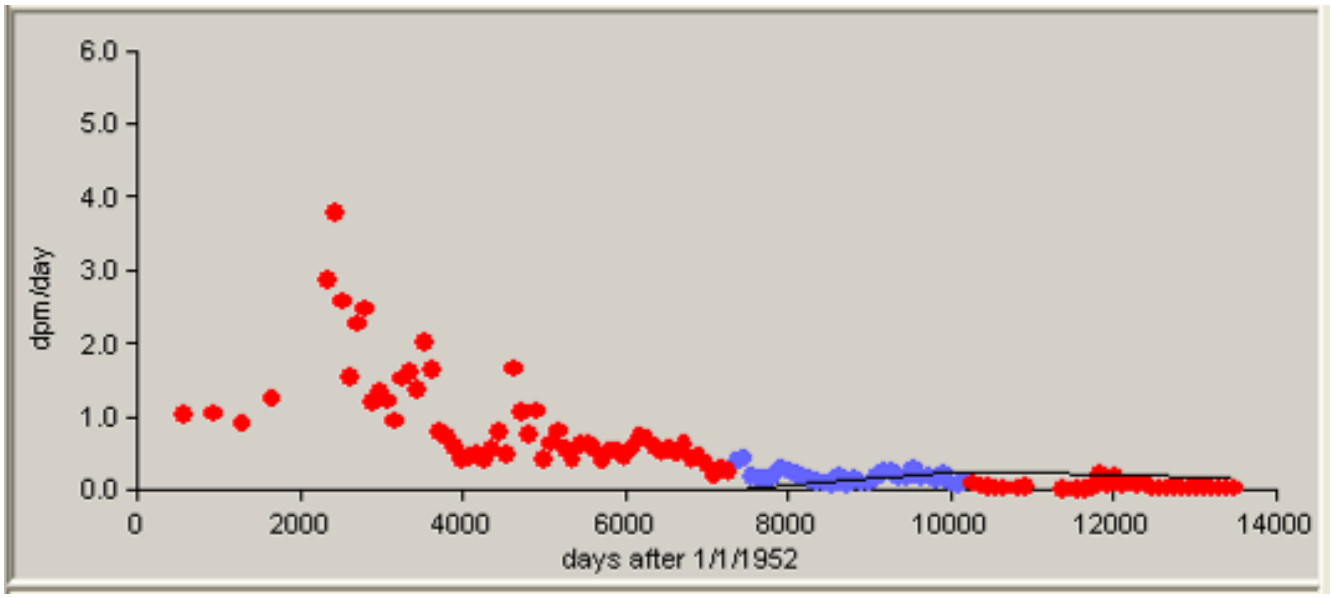

Figure D-38. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1972 to 12/31/1979, 84th-percentile, Type S.

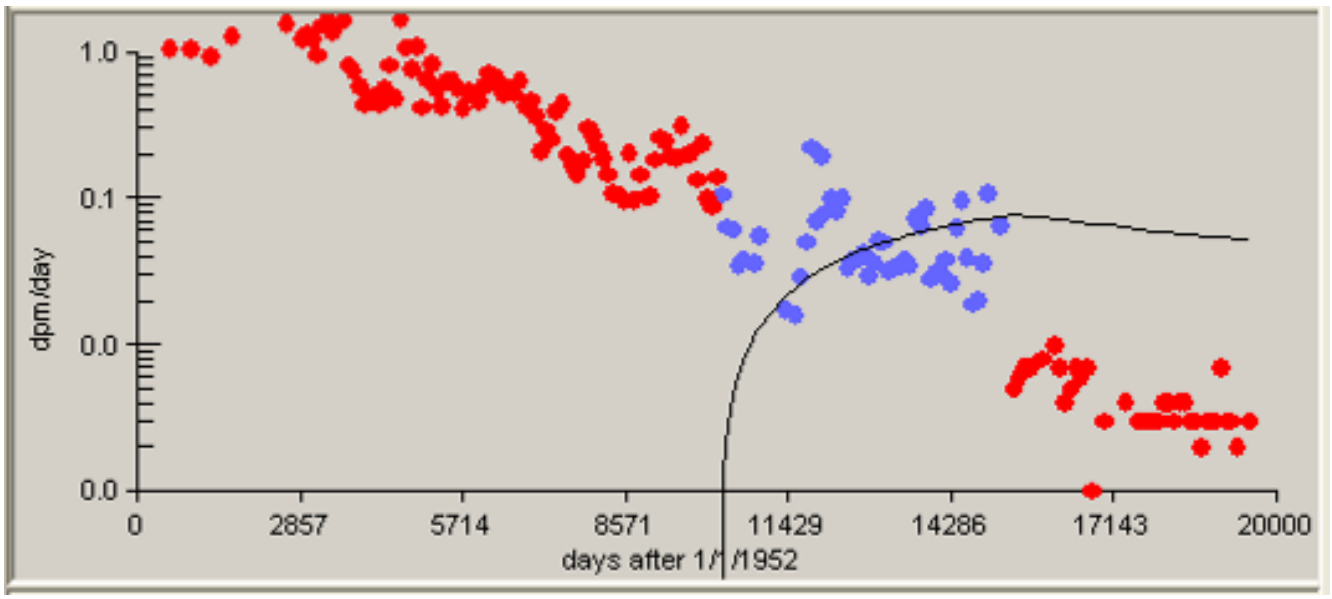

Figure D-39. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1980 to 12/31/1993, 84th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

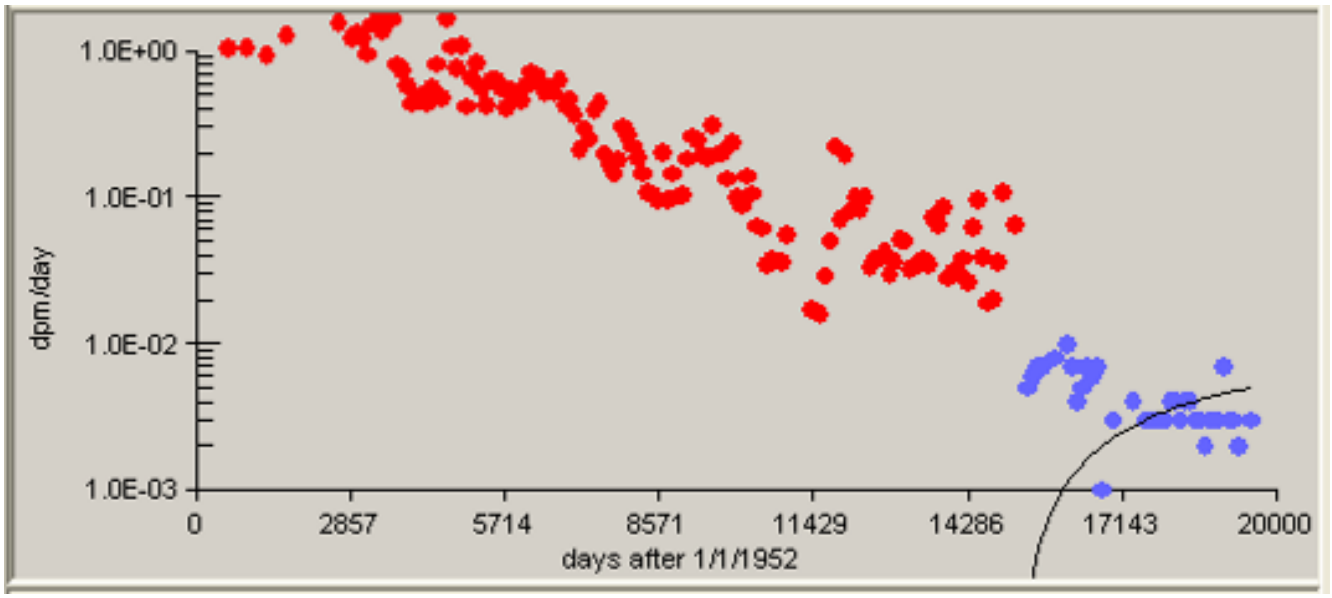

Figure D-40. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1994 to 12/31/2005, 84th-percentile, Type S.

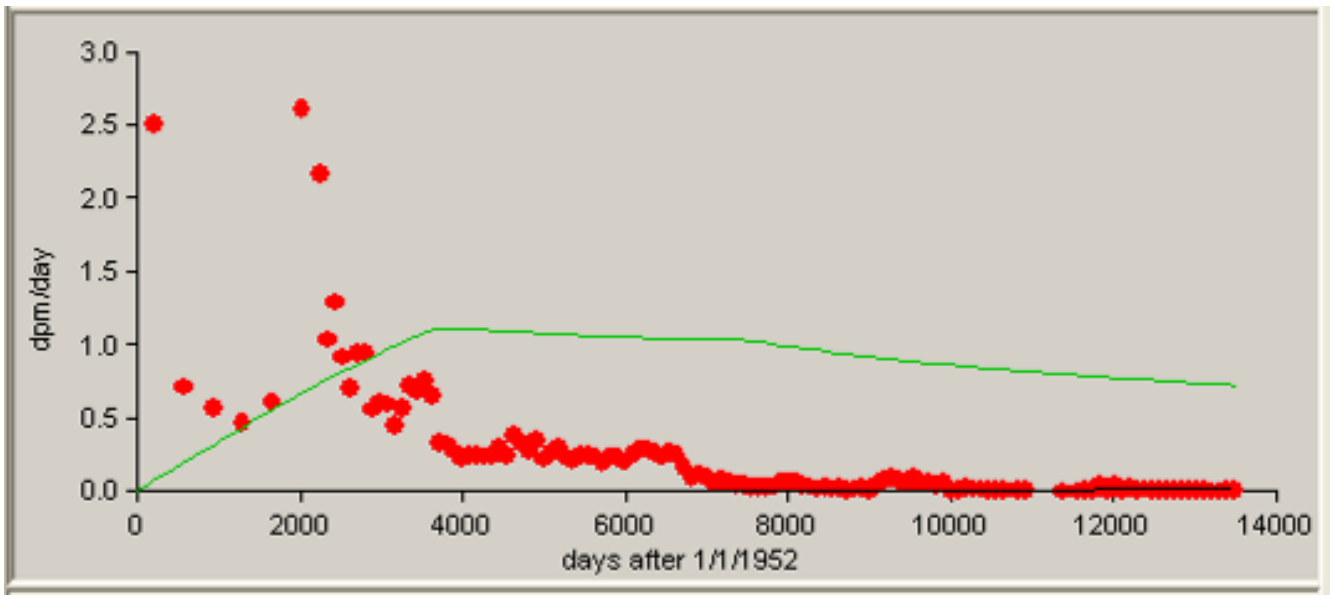

Figure D-41. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/2005, 50th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

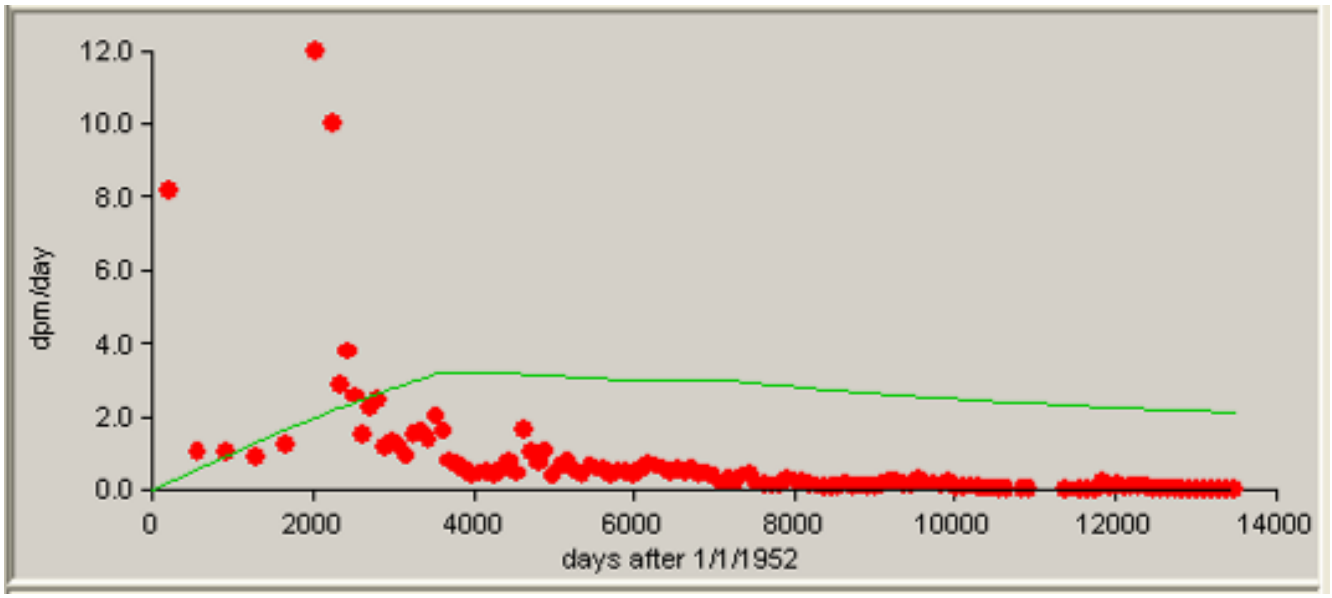

Figure D-42. Predicted plutonium bioassay results (line) calculated using IMBA-derived plutonium intake rates compared with measured plutonium-in-urine results (dots), 1/1/1952 to 12/31/2005, 84th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

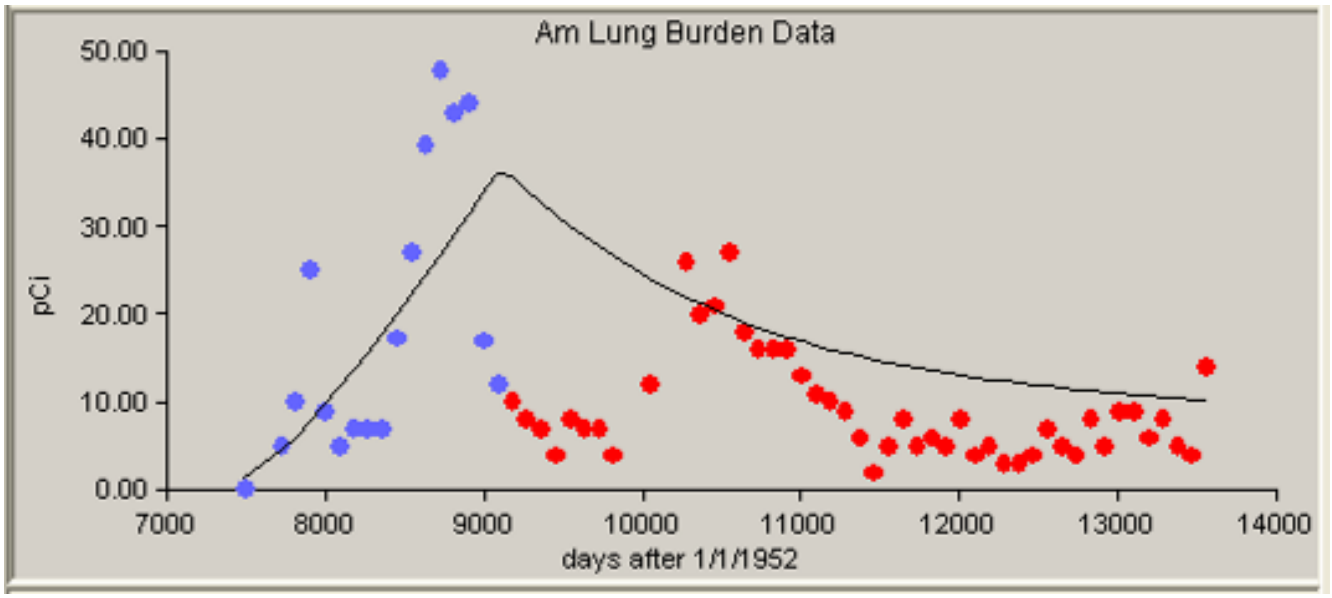

Figure D-43. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1972 to 12/31/1976, 50th-percentile, Type S.

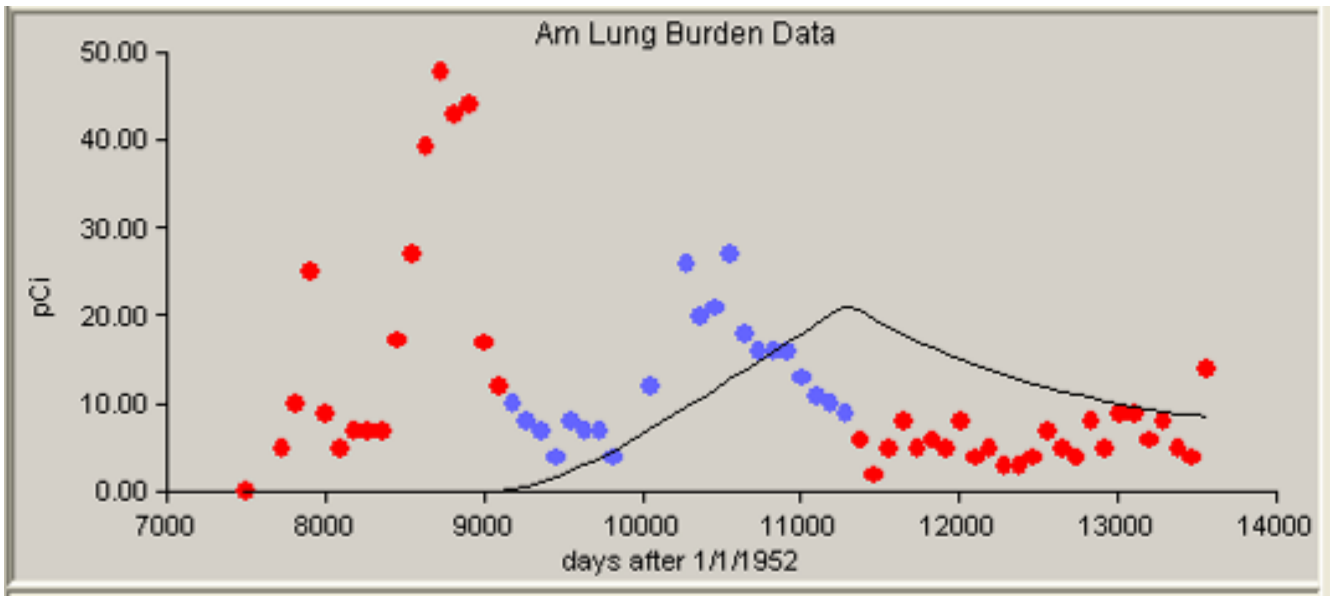

Figure D-44. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1977 to 12/31/1982, 50th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

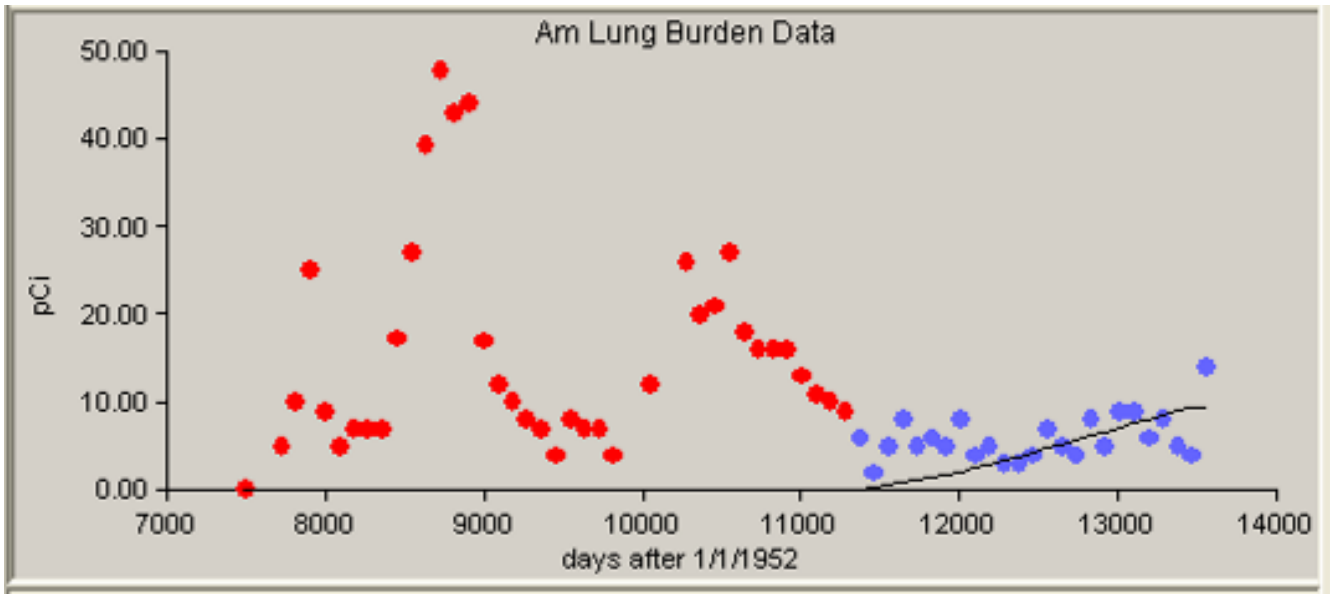

Figure D-45. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1983 to 12/31/1988, 50th-percentile, Type S.

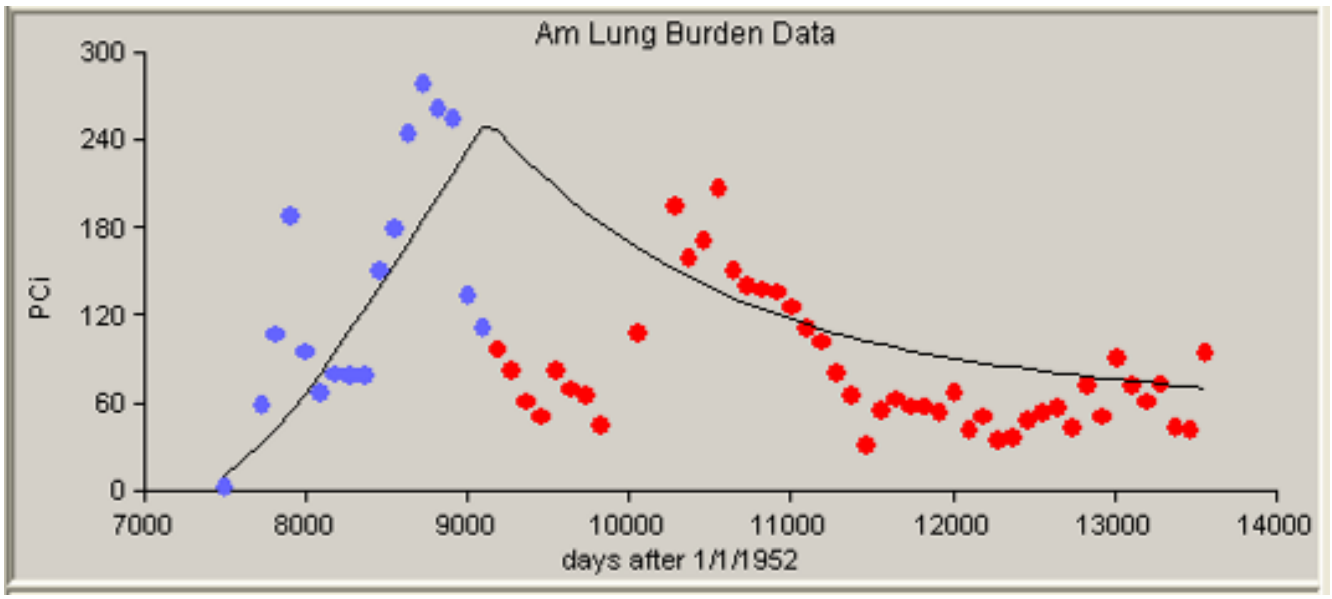

Figure D-46. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1972 to 12/31/1976, 84th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

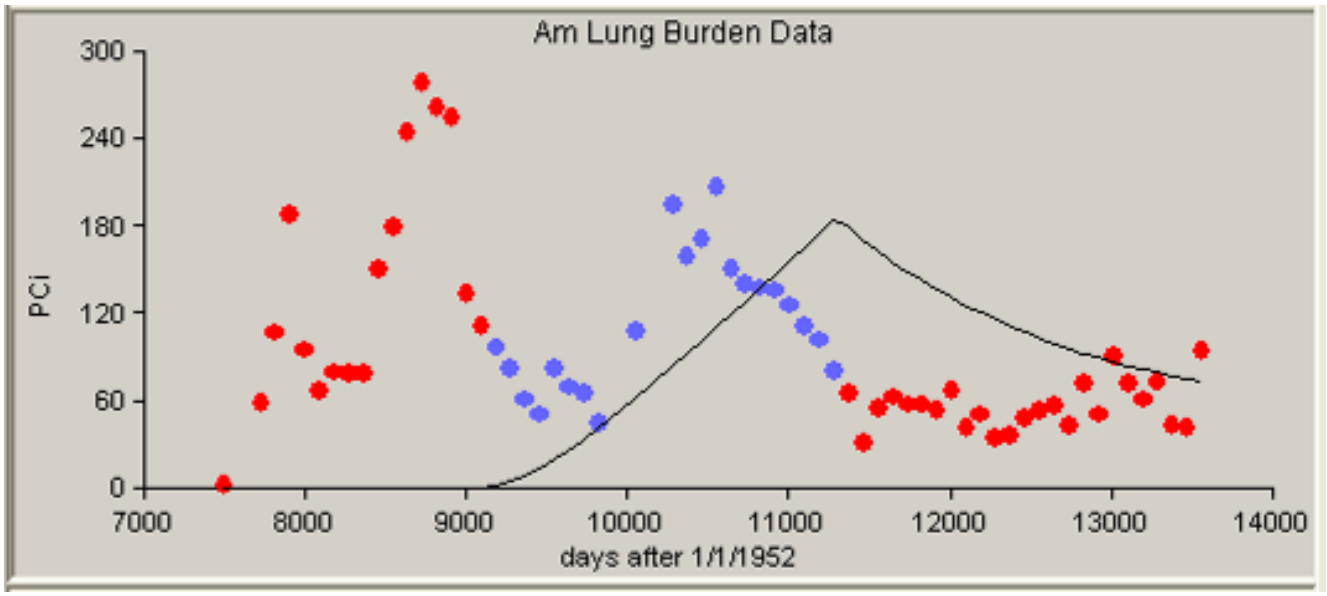

Figure D-47. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1977 to 12/31/1982, 84th-percentile, Type S.

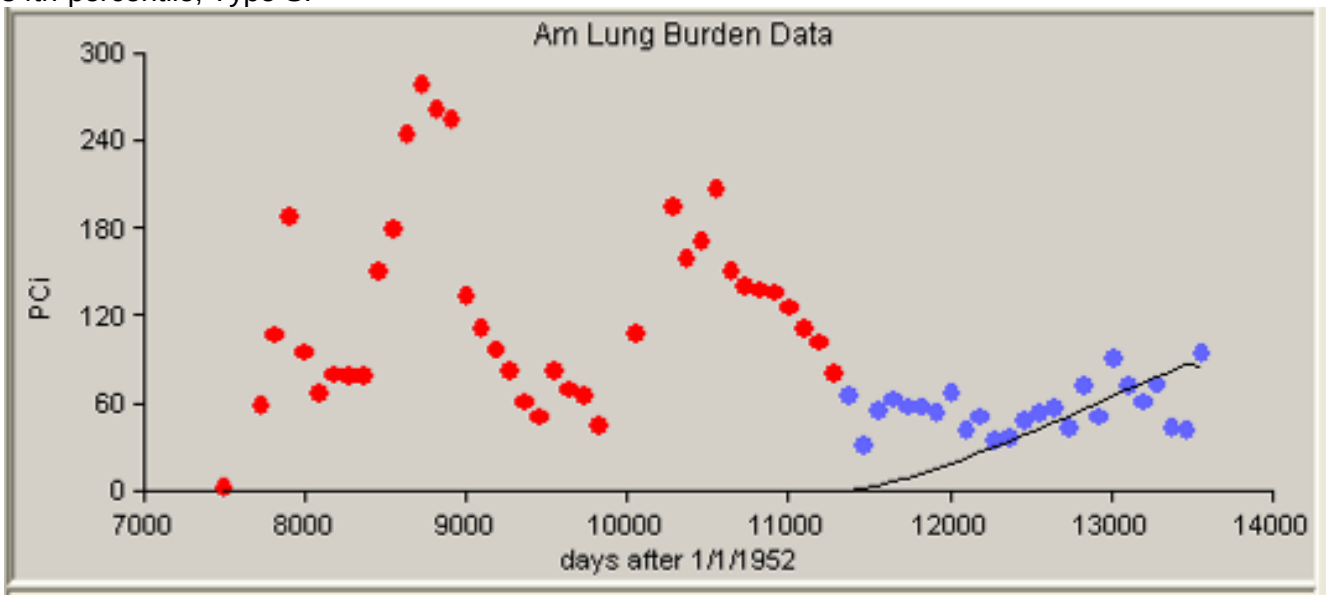

Figure D-48. Predicted americium bioassay results (line) calculated using IMBA-derived americium intake rates compared with measured americium lung burden results (dots), 1/1/1983 to 12/31/1988, 84th-percentile, Type S.

ATTACHMENT D
INTERNAL COWORKER DOSIMETRY DATA FOR ROCKY FLATS PLANT (continued)

Table D-9. IMBA-derived plutonium/americium intake rates, Type S.

Year	Plutonium urinalysis-based results, dpm/day			Americium lung count-based results, 50th percentile, pCi/day		
	Pu 50%	Pu 84%	GSD	Am 50%	Am 84%	GSD
1952–1961	1925	5628	2.92			
1962–1969	781.1	1899	2.43			
1970–1971	112	465.8	4.16			
1972–1976	112	465.8	4.16	0.0862	0.595	6.91
1977–1979	112	465.8	4.16	0.0534	0.465	8.70
1980–1982	13.69	104.1	7.60	0.0534	0.465	8.70
1983–1988	13.69	104.1	7.60	0.024	0.0240	9.12
1989–1993	13.69	104.1	7.60			
1994–2005	0.7993	7.838	9.81			