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Dose Reconstruction
Project for NIOSH

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AMAD	activity median aerodynamic diameter
cm	centimeter
cnts	counts
cnts/dis	counts per disintegration
cnts/min	counts per minute
dia	diameter
dpm	disintegrations per minute
EDP	Electronic Data Processing
EEOICPA	Energy Employee Occupational Illness Compensation Program Act of 2000
ft	feet
g	gram
g/24-hr	grams per 24-hour sample (fecal)
GSD	geometric standard deviation
HPGe	hyper-pure, intrinsic germanium
hr	hour
ID	identification
ICRP	International Commission on Radiological Protection
in.	inch
IREP	Interactive RadioEpidemiological Program
IVGS	*in vivo* gamma-ray spectrometer
keV	kiloelectron-volt, 1,000 electron-volts
L	liter
m	meter
µm	micrometer
MDA	minimum detectable activity
mL	milliliter
mL/24-hr	milliliter per 24-hour sample (urine)
min	minute
MMES	Martin Marrieta Energy Systems
mo	month
MPBB	maximum permissible body burden
MPOB	maximum permissible organ burden
mrem	millirem
N	population size
NaI	sodium iodide
NaI-CsI	sodium iodide-cesium iodide phoswich detector
NaI(Tl)	thallium drifted sodium iodide

NaI/Ge	sodium iodide/germanium
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
ORNL	Oak Ridge National Laboratory
pCi	picocurie
POC	probability of causation
SBC	scanning bed counter
TBD	technical basis document
U.S.C.	United States Code
WBC	Whole-Body Counting (Facility)
wk	week
µCi	microcurie
§	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 historic background information and guidance to assist in the preparation of dose reconstructions for particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

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

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

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

- Radiation from naturally occurring radon present in conventional structures
- Radiation from diagnostic X-rays received in the treatment of work-related injuries

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

5.1.1 Purpose

The purpose of this TBD is to describe internal dosimetry systems and practices at Oak Ridge National Laboratory (ORNL) from late 1943 to the present. It discusses historical and current practices used in the evaluation of internal radiation exposure of monitored and unmonitored workers. This TBD can serve as a supplement to, or substitute for, individual monitoring data.

5.1.2 Scope

ORNL began operations in early 1943. The startup of the Graphite Reactor and plutonium separation activities in late 1943 introduced the potential for personnel exposures from intakes of radioactive material. Laboratory operations involving radioactive materials increased over subsequent years as ORNL expanded its roles in radionuclide production and development of chemical separations processes.

Development of methods and techniques for internal monitoring (bioassay) was one of the many priorities at ORNL in its early years of operation, because such methods simply did not exist. Although ORNL used air sampling and radiological contamination monitoring programs as qualitative indicators of internal exposure, urinalyses for various internal contaminants did not begin at the site until about 1947. (A limited number of *in vivo* measurements appear to have begun at ORNL in 1959.) ORNL maintained early tolerance levels for airborne contamination based on "product" (i.e., ^{239}Pu) concentrations in the air (Cox 1944). [These early tolerance levels for alpha and beta-gamma contaminants were 3E-11 and 1E-07 µCi/cm^3 respectively during the mid 1940s.] In addition, the Laboratory later established tolerance levels for materials such as ^{131}I and noble gases.

As written, this TBD is applicable for the period 1947 to the present.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 5.5.

5.1.3 Radionuclides of Concern

Because of the many diverse processes and experiments that took place there, a complete list of radionuclides encountered at ORNL would be difficult to assemble. Radionuclides likely to produce a measureable internal dose include uranium, activation products, fission products, and transuranics [1]. The earliest urine sample results that were provided by ORNL were for the isotopes ^{239}Pu and ^{90}Sr. The electronic data that were provided by ORNL for use in estimating isotopic MDAs came from a project performed in the early to mid-1990s to convert hardcopy data over to a dBase IV database. Funding ran out on the conversion project in the mid-1990s and the entire set of site data was never completely converted, but a significant number of results were made available for our use. Table 5-1 lists radionuclides included in *in vitro* bioassay results provided by ORNL for the period from 1947 to 1988. The results were provided for estimating minimum detectable activities (MDAs) for various analyses. The source of these data was an electronic database created by ORNL in the early 1990s from hard-copy bioassay records. The sample size values in Table 5-1 are the numbers of analyses for that nuclide included in the data ORNL provided. These values do not reflect the total number of *in vitro* bioassays performed by ORNL in this period, because not all hard-copy records are in the database.

Table 5-1. Radioanalytical results between 1947 and 1988 [2].

Nuclide	Sample size	Nuclide	Sample size	Nuclide	Sample size
Am-241	5,670	I-131	41	S-35	10
Am-243	12	Mn-54	2	Sb-125	1
As-74	7	Mo-99	1	Sm-151	11
BG	2	Na-24	3	Sr-85	1
Bk-249	14	Nb-95	3	Sr-89	37
Br-82	2	Np-237	55	Sr-90	12,893
Br-83	2	P-32	166	Tc-99	20
C-14	11	Pa-231	55	Th-232	1,125
Ca-45	4	Pa-233	16	Tl-201	1
Ce-144	37	Pa-234	1	Tl-204	1
Cf-249	3	Pb-210	2	Tm-170	6
Cf-252	14	Pm-147	80	U-232	1
Cl-36	1	Po-210	66	U-233	829
Cm-242	12	Pu-238	65	U-235	3
Cm-244	299	Pu-239	15,476	U-238	11,434
Co-60	83	Pu-241	112	U-239	11
Cs-134	1	Pu-242	41	Y-88	5
Cs-137	3,561	Ra-226	333	Y-90	31
Fe-59	9	Ra-228	1	Zn-65	7
Gross alpha	4,875	Rare earths	1,098	Zr-95	20
Gross beta	324	Ru-103	1		
H-3	2,070	Ru-106	65		

Isotope-specific analyses for *in vitro* samples did not become routine until 1989. Prior to that time, chemical methods were used to separate radioelements as well as practicable, and the materials were assayed in terms of total activity. The activity measured would later be assigned to a predominant nuclide. Thus, a result from the early years might indicate ^{90}Sr, when in reality it includes ^{89}Sr. The same is true for early plutonium results and results for transuranic materials. Thus, "associated" radionuclides are inherently included in such results. Process knowledge of radionuclides present in various work areas was used to assign nuclides to sample results. The present ORNL internal dosimetry program uses a limited number of radionuclides for screening purposes. Positive results are followed up with additional bioassays.

5.1.4 Solubility Classes

Internal dosimetrists at ORNL provided a list of assumed intake modes and clearance class information for 73 radionuclides. These assumptions, given in Table 5-2, are those used by ORNL when re-evaluating historical bioassay results. They are provided as a reference for dose reconstructors, with the caveat that the assumed solubility classes are not known to be based on any specific studies. In general, they merely reflect conservative choices for dose assessment in terms of committed effective dose equivalent. Dose reconstructors therefore should not assume that these classes, which are in terms of the system promulgated in (ICRP 1966), represent favorable to claimant choices for tissue-specific dose evaluations. Dose reconstructors should instead make a favorable to claimant choice for the radionuclide and tissue(s) of interest using the system of solubility classes described in ICRP Publication 66 (ICRP 1994) [3].

Table 5-2. Solubility classifications used by ORNL for reevaluation of historical bioassay results (ORNL 2002).

Material class	Intake mode	Radionuclides
V (Very soluble)	Ingestion	H-3
L (Labeled organic)	Inhalation	C-14
D (Days)	Inhalation	Na-22, P-32, P-33, S-35, Rb-86, Sr-85, Sr-89, Sr-90, I-125, I-129, I-131, Ba-133, Cs-134, Cs-137, Eu-152, Eu-154, W-188
W (Weeks)	Inhalation	Ca-45, Cr-51, Mn-54, Fe-55, Fe-59, Ni-63, Ge-68, Sc-75, Tc-99, Gd-153, Hg-203, Bi-207, Po-210, Ra-226, Np-237, Am-241, Am-243, Cm-242, Cm-244, Cm-248
Y (Years)	Inhalation	Sc-46, Co-57, Co-58, Co-60, Cu-64, Zn-65, Cu-67, Y-88, Y-90, Pd-103, Ru-106, Cd-109, Ag-110m, Pm-147, Ir-192, Os-191, Th-228, Th-229, Th-230, Th-232, Pa-231, U-232, U-233, U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Cf-249, Cf-252

5.1.5 Route of Intake and Particle Sizes

Unless additional information is provided, it should be assumed in all cases that the route of intake for internally deposited radionuclides was via inhalation of 5 µm Activity Median Aerodynamic Diameter (AMAD)-sized particles.

5.1.6 Bioassay Programs

5.1.6.1 *In Vitro* Monitoring Program

ORNL has collected urine and fecal samples from individuals suspected of potential intakes from 1947 to the present. Urine samples have been and still are the preferred method. While fecal samples can provide good supplementary information to determine when an intake occurred, the chemical solubility of the material, and particle size, there typically is more variation associated with these samples than with urine samples. When fecal samples are obtained with urine samples after a known intake, the results can be used to better understand the intake parameters and provide a more accurate estimate of intake.

Urine samples were collected in the early years of the bioassay program based on the area health physicist's knowledge of field conditions (e.g., known spills/incidents, air and contamination sample results, etc.). This practice of scheduling did not utilize a specified sampling frequency (Auxier 2003; Henley 2003). A 1961 procedure manual (ORNL 1961) references procedures and practices governing the health physics program at that time, including internal and external exposure monitoring. Although referenced, the procedure detailing internal dosimetry was not among the documents available for review.

A similar manual of health physics procedures and practices from 1982 lists a detailed set of requirements for graduated routine sampling, depending on the frequency and extent of a potential exposure (ORNL 1982). Table 5-3 lists ORNL-published routine bioassay monitoring methods and frequencies from 1973 through 1982. NOTE: Historical information of the *in vivo* monitoring program (see Section 5.3) indicates that the whole-body counting frequencies listed in Table 5-3 were not consistently followed. Discussions with previous site personnel indicate that no formal counting frequency was used at ORNL until the later 1980s (Berger 2004).

Section 5.2 discusses the *in vitro* monitoring program and the data it produced.

Table 5-3. Routine bioassay monitoring methods and frequencies from 1973 to 1982[a] (ORNL 1982).

Radioactive material	Routine sampling categories		
	I	II	III
Pu-241 and alpha emitters other than uranium	Urine 3-4 wk Whole-body 3-4 mo	Urine 6-13 wk Whole-body 6-13 mo	Urine 6-12 mo Whole-body 6-12 mo
Sr-90 and uranium	Urine 4-6 wk	Urine 4-13 wk	Urine 6-12 wk
H-3	Urine each wk	Urine each mo	Urine each qtr
I-131	Whole-body each wk	Whole-body each qtr	Urine each qtr
Co-60 and Cs-137	Whole-body 3 mo	Whole-body 6 mo	Whole-body each 6 to 12 mo
All others	Consult w/Internal Dose Group		

a. Frequency of sampling should be in accord with employee's potential for exposure as determined by health physics surveyors. The following is a guide:

Category I Persons actively involved in operations or processes containing quantities of radioactive material, and when there is some evidence of contamination (i.e., positive results from smear and air samples).

Category II Employees working with relatively small quantities of materials that are confined or when there is no evidence of contamination or activity.

Category III Employees working with radioactive material or in the vicinity of material when there is no known exposure but some potential for exposure.

5.1.6.2 *In Vivo* Monitoring Program

ORNL has collected whole-body, lung, and wound counting data for employees since 1959. For the most part, *in vivo* counting was used until the late 1980s to confirm potential intakes from known incidents or identified by the *in vitro* monitoring program. Though Morgan, Snyder, and Struxness (1965) indicated that routine *in vivo* monitoring for all site radiological workers began in 1965, it appears that a formal program did not begin until the mid- to late 1980s. Section 5.3 discusses the *in vivo* monitoring program and the data it produced.

5.1.7 Recordkeeping

ORNL used various formats on paper records to record bioassay results through most of the period of interest. In addition, ORNL has entered much historical *in vitro* monitoring data into a database. These data were used to estimate MDAs as described in this document (see Attachment A). Discussions with personnel responsible for radiological records (Dixon 2004) indicated that hard-copy data are maintained at ORNL and that upon formal request these records are made available to the requester. These records will be used by dose reconstructors in estimating intakes.

It is not practical for this document to provide examples of all record formats used in the past six decades; MMES (1995) contains a guide to historical record formats. Most bioassay records share the following information: name, badge or other ID number, division code, health physics area, date, analysis code, results (in disintegrations per minute per 24-hr sample), and a reason for the analysis. A review of ORNL claim files indicated that many bioassay forms had slight revisions throughout their use, but that the information they maintain is similar from one revision to the next and should be easily interpreted.

5.1.7.1 Division Codes

The division code sometimes provides information on individual locations and job assignments. Table 5-4 lists historical division codes, and Table 5-5 lists more recent codes.

Table 5-4. Historical division codes (MMES 1995).

Codes	DIVNUM	Departments	Division name
AC	01	3390	Analytical Chemistry
BI	02	4455	Biology
CH	04		Chemistry
CT	03	3370	Chemical Technology
DI	20	3200	Directors
EC, GE	38	3060	Gen. Engr. and Construc.
ED	06	3480	Education
EL	07	3320	Electronuclear
FM	22		Finance and Materials
HE	23	3090	Health
HP	08	3810, 4193, 3193, 3490	Health Physics
IC	09	3341, 3075	Instrumentation and Controls
IE	24		Inspection Engineering
IS	25	3369, 3650, 4362, 3360	Isotopes
LP	26	3094	Laboratory Protection
MA	10	3166, 3516, 3152	Mathematics
MC	11	3470	Metals and Ceramics
MET			Metallurgy
NP	12	3410	Neutron Physics
OP	28	3639	Operations
PE	21	3016, 3078, 3003, 3004, 3062	Plant and Equipment
PH	13	3405	Physics
PI	30	3173	Public Information
PR	29	3107, 3141	Personnel
RC	14	3430, 4430	Reactor Chemistry
RE	16	4435, 3435	Reactor
RP	17		Research Participation
SS	18	3475	Solid State
TH	19	4460	Thermonuclear
TI	31	3072, 3148	Technical Information

5.1.7.2 Radioanalytical Abbreviations and Codes

Paper records contain abbreviation codes for recording analytes. These codes were often based on the initials of the analysis or the isotopic abbreviation of the radionuclide. Tables 5-6 and 5-7 list the codes for radionuclides measured in urine and fecal samples, respectively (Mani 1983) (MMES 1995).

5.1.7.3 Whole-Body Counting Results Codes

Table 5-8 lists codes used on some older cards and forms for documenting whole-body counting results. Many hardcopy records have been consolidated into individual personal records folders. However, this compilation is incomplete, with records only for employees with last names beginning with A through G.

Recovered data show that analytical MDAs tend to remain fairly consistent for a number of years. Abrupt changes in MDAs can be identified for groups of radionuclides during specific years. Following these changes, the MDAs remain generally consistent in subsequent years. This “step-wise” pattern allowed MDAs from several years to be grouped to obtain a single, representative MDA. Table A-2 in Attachment A provides annual averages for periods of time where bioassay data were available.

Table 5-5. New division codes (MMES 1995).

Code	DIVNUM	Division
AC	01	Analytical Chemistry
AS	70	Administrative Services
AT	07	Applied Technology
BI	02	Biology
BS	95	Business Systems
CH	04	Chemistry
CM	20	Central Management
CS	63	Computing and Telecom
CT	03	Chemical Technology
EA	93	ESA
EC	35	Environmental Compliance
EH	36	Env. & Health Prot.
EN	15	Energy
EP	12	Eng. Physics & Math
ER	29	Employee Relations
ES	42	Env. Sciences
ET	16	Eng. Technology
EX	90	Executive Offices
FE	19	Fusion Energy
FM	37	Finance and Materials
FR	14	Fuel Recycle
GE	69	ORNL Engineering
GR	71	Graphics
HE	23	Health

Code	DIVNUM	Division
HS	08	Health & Safety RSRH
IC	09	Instrmt & Controls
IF		Isotopes
IR	43	Info. Resource Org.
IS	72	Isotopes Division
LP	26	Lab Prot.
MC	11	Metals and Ceramics
OC	62	Controller Office
OP	27, 32	Operations
OS	22	Operational Safety
PC	64	Procurement
PE	21	Plant & Equipment
PH	13	Physics
PU	73	Publications
QA	24	Quality
RE	87	Env. Restoration
RP	14	Robotics
RR	06	Research Reactors
RU		Rust
SS	18	Solid State
TR	60	Treasurer Office
VI		Visitor
WM	27	Waste Management

Table 5-6. EDP urinalysis abbreviations and codes (Mani 1983; MMES 1995).

Alphanumeric code		Numeric code	
CM0	Cm-244	000	Other
CO0	Co-60	001	S-35
CS0	Cesium B (Cs-137)	002	Co-60
CS7	Cs-137	003	Pb-210
EU	Eu-154	004	Na-24
FP0, FP	Fission products (Cs-137)	005	Zr-95/Nb-95
FU0	Total rare earths ()	006	Tc-99
GA0, GA	Gross alpha (Pu-239)	007	As-74
GB0, GB	Gross beta (Sr-90)	008	Br-82/Br-83
GD0	Gd-153	009	Fe-59
GG0	Gross gamma (Cs-137)	010	Mn-54
GU0	Gross alpha (Pu-239)	011	I-131
HY3	H-3	012	Cs-132
I25	I-125	013	Gross beta
I31	I-131	014	Ba-140
NP0	Np-237	015	Sb-125

Alphanumeric code		Numeric code	
PA0	Pa-230 alpha	016	Tl-204
PA3	Pa-233 beta	017	Np-237
PH2	P-32	018	Ag-110m
PM7	Pm-147		
PO0	Po-210		
PU0, PU	Pu-239 alpha		
PU1	Pu-241		
PU9	Pu-239		
RA0	Ra-226		
RU6	Ru-106		
SR0	Sr-90		
SR5	Sr-85		
SR9	Sr-89		
TA0	Ta-180		
TH	Th-232		
TP0	Trans plutonium alpha (Am-241)		
TRE	Total rare earths ()		
URO	U-234 alpha		

The ORNL database contains values for isotopic activities during times when isotope-specific analyses were not possible or routinely performed. In earlier years, the element of concern was

Table 5-7. Fecal analysis codes (Mani 1983; MMES 1995).

Alphanumeric code	
GF0	Pu-239 gross alpha (includes Th)
PF0	Pu-239
RF0	Rare earths ()
SF0	Sr-90
SF9	Sr-89
TF0	Trans plutonium (Am-241)
UF0	U-234
OF0	Other

Table 5-8. Result codes for whole-body counting [4].

Before 1971		1971–1978	
Code	**Description***	**Code**	**Description***
1	Normal human spectrum	0	= <15% MPOB
2	Less than 10% MPBB	2	= <25% MPOB
3	Less than 25% MPBB	4	= <50% MPOB
4	Less than 50% MPBB	6	= <100% MPOB
5	Greater than 50% MPBB	8	>100% MPOB
		N	Insignificant & indeterminate
		S	Significant & indeterminate

*MPBB = maximum permissible body burden; MPOB = maximum permissible organ burden.

extracted chemically from the biological sample and the total radioactivity of the element in the extract was measured. At some point after the extraction and sample count, the total sample activity was attributed to a specific radionuclide. Many of the isotopic assignments were based on process knowledge.

To reflect the performance of the instrumentation and analytical methods more accurately during the period prior to 1990, Table 5-9 assigns MDAs for some isotopes to their corresponding radioelement rather than the specific radionuclide. For example, routine separation of the alpha emitters ^{238}Pu and ^{239}Pu did not occur until alpha spectrometric analyses became routine in 1989, but the recovered database reports both separately. They have been combined in this document as "Plutonium." NOTE: ^{241}Pu is reported separately because it is a beta emitter that can be assayed separately from the alpha-emitting isotopes of plutonium.

5.2 *IN VITRO* MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PRACTICES

5.2.1 *In Vitro* Minimum Detectable Activities

5.2.1.1 Minimum Detectable Activities between 1947 and 1989

Table 5-9 lists historical MDAs for radionuclides of concern in urine and feces. These MDAs were calculated from analytical records recovered from the *in vitro* sample databases supporting the ORNL historic workforce dose assessment project (MMES 1995). Attachment A contains details of data recovery and additional information on specific radionuclides. Blank MDA entries in the early years in Table 5-9 indicate that no analytical results for that radionuclide were recovered from that year. This could be because the records have not been located, or because the analysis was not performed.

Table 5-9. Recommended *in vitro* MDAs (dpm/24-hr sample) for radionuclides from 1947 to 1989.

	Urine MDAs (dpm/24h sample)																	Fecal MDAs (dpm/24h sample)				
Year	Gross alpha	Gross beta	Am-241	Cm-244	Cs-137	H-3	I-131	Np-237	Pm-147	Plutonium	Polonium	Pu-241	Ra-226	Rare earths	Ru-106	Sr-89 + Sr-90	Uranium	Gross alpha	Am-241	Cm-244	Plutonium	Th-232
1943																						
1944																						
1945																						
1946																						
1947										0.38												
1948										0.38							1.4					
1949										0.38							1.4					
1950					33					0.21						34	1.4					0.53
1951			0.33		33					0.21				129	0.30	34	6.3					0.53
1952	0.26		0.33		33					0.21			31	129	0.30	34	6.3					0.53
1953	0.26		0.33		33					0.21	58		31	129	0.30	34	6.3					0.53
1954	0.26	1,135	0.33		33					0.21	58		1.3	129	0.30	34	6.3					0.53
1955	0.26	1,135	0.33		197					0.21	58		1.3	129	0.30	34	6.3					0.53
1956	0.26	1,135	0.33		197					0.21	58		1.3	129	0.30	34	6.3	0.64				0.53
1957	0.26	1,135	0.33		197			1.1		0.21	58		1.3	129	0.30	34	6.3	0.64			0.34	0.53
1958	0.26	1,135	0.33		197			1.1		0.21	58		1.3	129	0.30	34	6.3	0.64			0.34	0.53
1959	0.26	1,135	0.33		197			1.1	396	0.21	58		1.3	129	0.30	34	6.3	0.64			0.34	0.53
1960	0.26	1,135	0.33		197			1.1	396	0.21	58		1.3	129	53	34	6.3	0.64			0.34	0.53
1961	0.26	288	0.33		197	320,282		1.1	396	0.21	17		1.3	129	53	6.3	6.3	0.64			0.34	0.53
1962	0.26	288	0.33		197	320,282		1.1	396	0.21	17		1.3	129	53	6.3	6.3	0.64			0.34	0.53
1963	0.26	288	0.33		197	320,282		1.1	396	0.21	17		1.3	129	53	6.3	6.3	0.64			0.34	0.53
1964	0.26	288	0.33	0.22	197	320,282		0.09	396	0.21	17		1.3	129	53	6.3	1.1	0.64			0.34	0.53
1965	0.26	288	0.33	0.22	197	320,282	10,358	0.09	396	0.21	17		1.3	129	53	6.3	1.1	0.64	0.46	1.05	0.34	0.53
1966	0.26	288	0.33	0.22	197	320,282	10,358	0.09	396	0.21	17		1.3	129	53	6.3	1.1	0.64	0.46	1.05	0.34	0.53
1967	0.26	288	0.33	0.22	197	320,282	10,358	0.09	396	0.21	17		1.3	129	53	6.3	1.1	0.64	0.46	1.05	0.34	0.53
1968	0.09	39.8	0.08	0.10	51	55,681	10,358	0.09	396	0.06	17	9.2	1.3	129	53	6.3	0.09	0.64	0.46	1.05	0.34	0.53
1969	0.09	39.8	0.08	0.10	51	55,681	10,358	0.09	396	0.06	17	9.2	1.3	129	53	6.3	0.09	0.64	0.46	1.05	0.34	0.53
1970	0.09	39.8	0.08	0.10	51	55,681	10,358	0.09	396	0.06	17	9.2	1.3	129	53	6.3	0.09	0.64	0.46	1.05	0.34	0.53
1971	0.09	39.8	0.08	0.10	51	55,681	10,358	0.09	396	0.06	17	9.2	0.07	5.4	53	4.0	0.09	0.64	0.46	1.05	0.34	0.53
1972	0.09	*39.8*	0.08	0.10	51	55,681	10,358	0.09	396	0.06	17	9.2	0.07	5.4	53	4.0	0.09	0.64	0.46	1.05	0.34	0.53
1973	0.09	*39.8*	0.05	0.10	51	55,681	10,358	0.09	396	0.04	17	9.2	0.07	5.4	53	4.0	0.06	*0.64*	0.46	1.05	0.34	0.53
1974	0.09	*39.8*	0.05	0.10	51	55,681	10,358	0.09	396	0.04	17	9.2	0.07	5.4	53	4.0	0.06	*0.64*	0.46	1.05	0.34	0.53
1975	*0.09*	*39.8*	0.05	0.10	51	55,681	*10,358*	0.09	396	0.04	0.52	9.2	0.07	5.4	*53*	4.0	0.06	*0.64*	0.46	1.05	0.34	0.53
1976	*0.09*	*39.8*	0.05	0.10	51	55,681	*10,358*	0.09	396	0.04	0.52	9.2	0.07	5.4	*53*	4.0	0.06	*0.64*	0.46	1.05	0.34	0.53
1977	*0.09*	*39.8*	0.05	0.10	*51*	55,681	*10,358*	0.09	396	0.04	0.52	9.2	0.07	5.4	*53*	4.0	0.06	*0.64*	0.46	1.05	0.34	*0.53*
1978	*0.09*	*39.8*	0.05	0.10	*51*	55,681	*10,358*	0.09	*396*	0.04	0.52	9.2	0.07	5.4	*53*	4.0	0.06	*0.64*	0.46	1.05	0.34	*0.53*
1979	*0.09*	*39.8*	0.05	0.10	*51*	55,681	*10,358*	0.09	*396*	0.04	0.52	9.2	*0.07*	*5.4*	*53*	4.0	0.06	*0.64*	0.46	1.05	0.34	*0.53*
1980	*0.09*	*39.8*	0.05	0.10	*51*	55,681	*10,358*	0.09	*396*	0.04	0.52	*9.2*	*0.07*	*5.4*	*53*	4.0	0.06	*0.64*	0.46	1.05	0.34	*0.53*
1981	*0.09*	*39.8*	0.05	0.10	*51*	55,681	*10,358*	*0.09*	*396*	0.04	0.52	*9.2*	*0.07*	*5.4*	*53*	4.0	0.06	*0.64*	*0.46*	*1.05*	*0.34*	*0.53*
1982	*0.09*	*39.8*	0.05	*0.10*	*51*	*55,681*	*10,358*	*0.09*	*396*	0.04	*0.52*	*9.2*	*0.07*	*5.4*	*53*	4.0	0.06	*0.64*	*0.46*	*1.05*	*0.34*	*0.53*
1983	*0.09*	*39.8*	0.05	*0.10*	*51*	*55,681*	*10,358*	*0.09*	*396*	0.04	*0.52*	*9.2*	*0.07*	*5.4*	*53*	4.0	0.06	*0.64*	*0.46*	*1.05*	*0.34*	*0.53*
1984	*0.09*	*39.8*	0.05	*0.10*	*51*	*55,681*	*10,358*	*0.09*	*396*	0.04	*0.52*	*9.2*	*0.07*	*5.4*	*53*	4.0	0.06	*0.64*	*0.46*	*1.05*	*0.34*	*0.53*
1985	*0.09*	*39.8*	0.05	*0.10*	*51*	*55,681*	*10,358*	*0.09*	*396*	0.04	*0.52*	*9.2*	*0.07*	*5.4*	*53*	4.0	0.06	*0.64*	*0.46*	*1.05*	*0.34*	*0.53*
1986	*0.09*	*39.8*	0.05	*0.10*	*51*	*55,681*	*10,358*	*0.09*	*396*	0.04	*0.52*	*9.2*	*0.07*	*5.4*	*53*	4.0	0.06	*0.64*	*0.46*	*1.05*	*0.34*	*0.53*
1987	*0.09*	*39.8*	0.05	*0.10*	*51*	*55,681*	*10,358*	*0.09*	*396*	0.04	*0.52*	*9.2*	*0.07*	*5.4*	*53*	4.0	0.06	*0.64*	*0.46*	*1.05*	*0.34*	*0.53*
1988	*0.09*	*39.8*	0.05	*0.10*	*51*	*55,681*	*10,358*	*0.09*	*396*	0.04	*0.52*	*9.2*	*0.07*	*5.4*	*53*	4.0	0.06	*0.64*	*0.46*	*1.05*	*0.34*	*0.53*
1989	*0.09*	*39.8*	*0.05*	*0.10*	*51*	*55,681*	*10,358*	*0.09*	*396*	*0.04*	*0.52*	*9.2*	*0.07*	*5.4*	*53*	*4.0*	*0.06*	*0.64*	*0.46*	*1.05*	*0.34*	*0.53*

5.2.1.2 Minimum Detectable Activities after 1989

MDA values for present (since 1989) samples are available with each sample. Table 5-10 lists typical, current MDAs for radionuclides of concern in urine and feces.

Table 5-10. MDAs for *in vitro* samples after 1989 (McLaughlin 2002).

Isotope	MDA (dpm/24-hr sample)
H-3	9,100
C-14	4,480
P-32	1
Sr-90	3
Tc-99	200
I-131	20
Np-237	0.02
Th-232	0.02
U-232	0.02
U-233	0.02
U-234	0.02
U-235	0.02
U-238	0.02
Pu-238	0.02
Pu-239	0.02
Am-241	0.02
Cm-244	0.02
Cf-252	0.02
Bk-249	26

5.2.2 Counting Methods for *In Vitro* Samples

Several counting methods have historically been available at ORNL for determining radioactivity in *in vitro* samples. The following sections discuss alpha spectrometry, liquid scintillation, zinc sulfide scintillation, gamma spectrometry, and beta counting using a gas flow proportional counter.

5.2.2.1 Alpha Spectrometry

ORNL uses alpha spectrometry in the analysis of nuclides that decay primarily by alpha emission, with only very low-energy photons or none at all. A tracer is added to the bioassay sample before analysis begins to determine the chemical yield of the process. The radioelements are chemically separated from the sample and electrodeposited on stainless-steel disks. Plutonium and the other transuranic elements can be analyzed sequentially; uranium analysis requires a separate sample.

5.2.2.2 Liquid Scintillation

Liquid scintillation is used for the analysis of low-energy, pure beta emitters, specifically ^{3}H and ^{14}C. One milliliter of a urine sample is mixed with a scintillation cocktail for analysis.

5.2.2.3 Zinc-Sulfide Scintillation

ORNL used zinc-sulfide scintillation counting to count alpha emitters such as the trivalent alpha actinides (see Sections 5.2.3.1 and 5.2.3.2).

5.2.2.4 Gamma Spectrometry

ORNL uses gamma spectrometry to identify and quantify radionuclides that emit photons with energies greater than 60 keV. A high-resolution, hyper-pure, intrinsic germanium (HPGe) detector with a beryllium end-window is used. Urine samples are placed in a 1-L Marinelli beaker, which is placed over the detector for counting. If the total sample volume is less than 1 L, distilled or deionized water is added to bring the volume up to 1 L. Fecal samples were counted directly in the sample counter for screening purposes, and they might have been ashed and placed into a 2-in. Petri dish for quantitative results.

5.2.2.5 Gas Flow Proportional Counter (Beta Counting)

ORNL uses the gas flow proportional counter system for the analysis of strontium. Strontium is chemically separated from the sample and filtered onto a glass fiber filter. The filter is placed on a planchet for insertion into the counter. The counting system does not distinguish between beta energies, so the reported result is total strontium (^{89}Sr plus ^{90}Sr).

5.2.3 <u>Notes on Measurements of Alpha Emitters</u>

5.2.3.1 Trivalent Alpha Actinides

Before 1989, ORNL did not perform radionuclide-specific analyses for americium, curium, and other high atomic number elements beyond plutonium. [These radionuclides were typically recorded on the HP Body Fluids Analysis Request cards as transplutonium (TPO or TPL).] Rather, the Laboratory separated trivalent alpha actinides as a group and analyzed by zinc-sulfide scintillation counting. Therefore, monitoring of transplutonium elements was unable to differentiate between such nuclides as ^{241}Am and ^{244}Cm. The default radionuclide to use with measurements involving trivalent alpha actinides would be ^{241}Am. The detection sensitivity of the transplutonium analysis technique is not well-documented for samples processed before 1985. However, laboratory records suggest that the transplutonium detection level was about 0.2 dpm through 1985.

For analyses performed after 1985, an estimate of the sample-specific MDA is generally reported. Alpha spectroscopy analysis for transuranic elements in bioassay samples began in the early- to mid-1980s. Differentiation, by alpha energy separation, for isotopes such as ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{241}Am, and ^{244}Cm became possible.

5.2.3.2 Plutonium

Before 1989, ORNL did not routinely perform isotope-specific analyses for plutonium. Rather, the Laboratory separated plutonium as an element and analyzed it by zinc-sulfide scintillation counting. Therefore, the historic plutonium analysis technique was unable to differentiate among the alpha emitting isotopes ^{238}Pu, ^{239}Pu, and ^{240}Pu. Site analytical personnel assert that in the early 1980s, positive total Pu measurements were recounted on the limited number of alpha spectrometers that were available at the site. Though many HP Body Fluids Analysis Request cards in the claims files gave results for the isotopes ^{238}Pu and ^{239}Pu, this was not consistently observed. The default isotope for positive, total Pu measurements should be ^{239}Pu [5].

5.2.3.3 Environmental Uranium

The following paragraphs were taken entirely from the Oak Ridge National Laboratory Internal Dosimetry Program Technical Basis Document (McLaughlin 2002). They are included as an aid to dose reconstructors in the interpretation of uranium bioassay results for ORNL workers.

> *Environmental levels of naturally-occurring uranium are found throughout eastern Tennessee. The environmental activity levels in the immediate area surrounding Oak Ridge are sufficiently high such that dietary intake of uranium is detectable with 24-hour urine samples. A urinary uranium background study was conducted in the mid 1990's using non-occupationally exposed employees to quantify the range of typical background uranium excretion. Based upon the results of that study, a discrimination level (set at the 99th percentile level) of 0.14 dpm/day was established for both U-234 and U-238 to differentiate between environmental and occupational exposure to uranium. A value of 0.25 dpm/day is applied to total uranium results. Plots of the observed uranium excretion distributions for U-234, U-238, and total uranium are provided below.*
>
> *An activity ratio of U-234 to U-238 of approximately 2 to 1 has been observed within the analyzed local potable water samples. Activity ratios in this range have been reported for various aquifers (Osmond). Though U-234 and U-238 should be in secular equilibrium within nature, the observed enrichment of the isotope U-234 is believed to be caused by several factors which include the direct transfer of U-238 decay products across a solid/liquid phase boundary by alpha recoil and differences in solubility between uranium decay chain members. Recognizing this trend, uranium bioassay results that are less than 0.2 dpm/day that do not exhibit a U-234 to U-238 activity ratio of 2:1 should be considered suspect and investigated.*

5.3 *IN VIVO* MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PRACTICES

5.3.1 Shielded Counting Room

The ORNL Whole-Body Counting (WBC) facility, sometimes referred to as the *In Vivo* Gamma-Ray Spectrometer or IVGS), is in Building 2008 in the northwest corner of the Main Plant area. Several Health Physics Division annual progress reports (Morgan, Snyder, and Struxness 1961, 1963, and 1966) indicate that the WBC facility began operation in June 1960 and another report (Brown, Patterson, and Abee 1971) indicates a May 1960 date; a more recent document (Watts et al. 1995) indicates several counts occurred earlier. Thus, it is likely that the WBC facility began limited operations in 1959.

The main counting room has inner dimensions of 10 ft by 10 ft by 10 ft. Its walls consist of four layers of pre-World War II steel with a total thickness of 14 in. Some documents called it the *iron* or *big* room. The first recorded count in the WBC facility occurred on July 13, 1959, and, “was a background count [conducted] in the corner of the steel room with the door not in place and the roof incomplete” (Watts et al. 1995). The room was completed on July 24, 1959 (with the exception of a 0.125-in. layer of “special low-radioactivity lead” added to all inner surfaces in 1960 to reduce background radiation levels) and was used for a time to conduct background studies of paint and interior samples and to count biological samples such as milk, grass, and cow thyroids. The first recorded lung count of an employee occurred on May 19, 1960. Almost immediately after (May 19 to 20, 1960), it was used to conduct lung counts of three employees involved in an onsite contamination incident. The original

counting facility utilized a 4- by 4-in. NaI (Tl) crystal, but it is not clear what the counting geometry was. There is a photograph of an individual laying on a nylon-strapped, aluminum beach chair, but it was not clear whether that was the geometry initially employed in the facility. The *Argonne Chair* counting geometry was set up for whole-body counting on February 27, 1961 (Morgan, Snyder, and Struxness 1961). The tubular steel chair was tilted so the individual's body was in a V position with the detector placed approximately 50 cm directly over the hips (Morgan, Snyder, and Struxness 1961, p. 223).

Before construction of the WBC facility, health physicists realized that there could be problems with siting a low-level radiation counting facility in the plant environment, with radioactive effluents from operations being generated immediately adjacent to the facility. Thus, ORNL designed and installed a recirculating air treatment system that pumped air from inside the counting room through charcoal traps, cooling coils, and heaters to remove radon, odors, and excess moisture (Brown, Patterson, and Abee 1971; Morgan, Snyder, and Struxness 1961). A slight positive pressure was maintained in the counting room by using cylinders of "aged" breathing air to make up for leakage from the system.

To improve the detection sensitivity, additional layers of material were applied to the floor in the WBC facility counting room over the years. Morgan, Snyder, and Struxness (1963) indicates that 0.04-in.-thick layers of tin and cadmium were laid over the interior surface of the lead, only on the floor, with a 0.01-in.-thick layer of copper over the tin and cadmium. The original vinyl tile was placed over the copper. These materials were installed to reduce background radiation emitted from the lead shield as a graduating shield to minimize the contribution of low-energy X-rays. In the mid-1990s, the vinyl tile surface was removed and a 0.03-in.-thick layer of stainless steel was placed directly over the other metal layers. Site personnel indicated that the stainless-steel layer was primarily for aesthetic purposes to cover the oxidized copper layer. With the addition of frictional surfaces for slip reduction, this was the latest form of the floor. Table 5-11 lists the construction history of the counting room.

Table 5-11. Construction history of the counting room.

Installation date	Material	Thickness (in.)
1959 (entire facility)	Pre-WW II steel	14
1960 (entire facility)	Special low-radioactivity lead	0.125
1963 (floor only)	Tin	0.04
1963 (floor only)	Cadmium	0.04
1963 (floor only)	Copper	0.01
Mid-1990s	Stainless steel	0.03

Subsequent improvements to the shielded counting room, detectors, and counting geometries are described below. The shielded room was used to conduct all *in vivo* measurements at ORNL until 1992, when a Canberra scanning bed counter (SBC) was installed in Room 16 of Building 2008.

A discussion with Berger (Berger 2003), who was responsible for the facility from the mid-1970s until the early 1980s, indicated that although the facility was in operation, a formal *in vivo* monitoring program was not in place until the late 1980s. Prior to that time, the WBC facility was used almost exclusively to either confirm known or suspected intakes of radioisotopes or for research purposes. Selection of individuals for counting was performed by field health physics personnel based on expected contaminants of concern and the probability of exposure until the early 1990s, when the selection of individuals for *in vivo* monitoring became the responsibility of the Internal Dosimetry staff.

5.3.2 Detectors, Geometries, and Techniques

As described above, the initial *in vivo* counter at ORNL utilized a 4- by 4-in. NaI (Tl) crystal and a tilted chair counting arrangement (Morgan, Snyder, and Struxness 1961). In July 1961, counting activities in the WBC facility were suspended to modify the detector arrangement with the installation of an 8- by 4-in. NaI (Tl) crystal to replace the smaller detector to increase efficiency and reduce counting times. In 1962, several thin (5-in.-diameter by 0.0625-in.) NaI (Tl) crystals were installed in the facility to quantify low-energy photons (e.g., mainly isotopes of plutonium). Also in that year, calibration studies were conducted using an arc-shaped geometry. This geometry was expected to result in less variation in counting efficiency than the chair geometry. (The arc geometry placed an individual laying in an arced position with the anterior portion of the body facing the detector at a distance of about 1 m, so each portion of the body was approximately the same distance from the detector (Mani 1983).

Morgan, Snyder, and Struxness (1963) indicated that the thin crystal detectors could see approximately 40 nCi of ^{238}Pu, if there was a preexposure chest count. If there was no preexposure count, the detection capability was approximately 80 nCi. In 1963, a SBC replaced the chair geometry using the 8- by 4-in. NaI (Tl) crystal for whole-body counting. The bed and individual were moved under the stationary detector. This geometry was used to determine roughly the part of the body in which the gamma-emitting radioisotopes were located. The detection efficiency was approximately equivalent to that of the chair geometry (Morgan, Snyder, and Struxness 1963). Morgan, Snyder, and Struxness (1963) noted that the computer output provided a "gross" spectrum as well as having the ability to "strip the ^{40}K and ^{137}Cs background counts." Several spectra observed during that period indicated a large portion of the "net" spectra with negative values. Morgan, Snyder, and Struxness (1964) noted that in March 1963 "written weekly reports of *in vivo* counting activities and results was begun." (The research to generate this document did not find any of these reports).

Morgan, Snyder, and Struxness (1965) reported that "Baseline counts on essentially every person with a potential for future exposure was completed in May 1965." By the third week of May 1965, a prioritizing system for selecting individuals for whole-body counting was initiated. Table 5-12 describes this system. Although several located documents stated that baseline and specified monitoring frequencies were utilized to make *in vivo* measurements, Berger (2003) and McLaughlin (2004) indicated that a full *in vivo* monitoring program did not exist at ORNL until approximately 1994, when site internal dosimetrists became responsible for identifying personnel for counting. Prior to that time, the area health physicists selected individuals for *in vivo* monitoring. The area health physicists were responsible for determining radioisotopes to which the worker could have been exposed and counting frequencies. This led to inconsistent approaches to the selection of individuals for monitoring.

Table 5-12. 1965 selection criteria for whole-body counting.

Priority	Selection criteria
1	Persons suspected of having sustained exposure
2	Persons being recounted as follow-up to initial elevated *in vivo* results
3	Persons who work directly with radioactive material (once every 3 mo)
4	Persons who work in areas where radioactive materials are handled, but do not work directly with the material (once every 6 mo)
5	New hires or other persons requiring a baseline count and a limited number of persons prior to termination

Morgan, Snyder, and Struxness (1966) indicated that the "job of determining which employees should be counted, the frequency with which they should be recounted, and preparing the necessary schedule cards and lists is requiring more time than had been anticipated." For this reason, the WBC

facility was still being used to verify if intakes had taken place rather than for routine monitoring of site personnel. A new Health Physics Division Report (Morgan, Snyder, and Davis 1967) begun in 1966 included data on numbers of *in vivo* counts and basic statistics for *in vivo* monitoring. They included numbers of individuals whose results exceeded the U.S. Atomic Energy Commission (AEC) reporting level (50% of the permissible body burden averaged over the year) or some other specified amount. Table 5-13 summarizes this information. NOTE: The use of annual averages could be misleading in cases involving short-lived or rapidly cleared materials. Mani (1983, Table 15) provided similar data, but he reported all *in vivo* counts that exceeded specified levels rather than annual averages. Table 5-14 lists this information.

Table 5-13. Qualitative information concerning radioactive material detected utilizing the WBC facility from 1966 to 1983.

Year	Number of persons exceeding the permissible body burden (based upon *in vivo* measurements)
1966[a]	0 persons exceeded 50% of permissible body burden
1967[a]	0 persons exceeded 50% of permissible body burden
1968[a]	0 persons exceeded 50% of permissible body burden
1969[a]	0 persons exceeded 50% of permissible body burden
1970[b]	0 persons exceeded 25% of permissible body burden
1971[b]	0 persons exceeded 25% of permissible body burden
1972[b]	0 persons exceeded 25% of permissible body burden
1973[b]	0 persons exceeded 25% of permissible body burden
1974[b]	2 (employees inhaled Cm-244 believed to range from 20 (15 pCi) to 40% (30 pCi) of organ (lung) burden)
1975[b]	4 (1 - Cm-244 of ~50%, 1 – U-238 of <15%, 1 – Zn-65 and Co-57 of <15%, and 1 – Co-60 of <15% of lung burden)
1976[b]	6 (all six appear to have had detectable amounts of Cm-244 of <15% of the lung burden)
1977[b]	20 (8 – ^{239}Pu and ^{241}Am of <25%, 8 – ^{137}Cs, ^{60}Co, ^{153}Gd, and ^{109}Cd of <15%, and 4 – ^{125}I and ^{123}Te <25% of the lung burden)
1978[b]	6 (1 - Pu-241 and Am-241 of ~70% (lung), 1 – ^{131}I-131 minor amount that could not be quantified (thyroid), and 4 – Co-60 of <25% (lung)
1979[b]	0 persons exceeded 25% of permissible lung burden
1980[b]	0 persons exceeded 10% of permissible lung burden
1981[b]	0 persons exceeded 10% of permissible lung burden
1982[b]	0 persons exceeded 10% of permissible lung burden
1983[b]	0 persons exceeded 10% of permissible lung burden

a. The AEC permissible level was 50% of the permissible body burden from 1966 through 1969.

b. From 1970, ORNL indicated in reports that in vivo measurements did not exceed specified amounts.

Table 5-14. Qualitative information concerning radioactive material detected utilizing the WBC facility through 1978 (Mani 1983).

Count year	Insignificant and indeterminate	Significant and indeterminate	<15% of MPOB	Normal spectrum	<10/25% of MPOB[a]	<25% of MPOB	<50% of MPOB	>50% of MPOB	<100% of MPOB	>100% of MPOB	Total counts
1962				59	9	3					71
1963				906	15						921
1964				1,485	6	20	13	31			1,555
1965				1,111	16	8	23	11			1,169
1966				632	11	12	28	13			696
1967				918	13	6	25	4			966
1968				815	135	12	6				968
1969				905	126	8	18				1,057
1970				499	14	2	1	2			518
1971	310	4	9					1			324
1972	255		4								259
1973	321	1	3								325
1974	246	2	8		3		6		7	6	278
1975	297	23	5		4		1				330
1976	215	6	29				3			1	254
1977	256	14	15		5		1				291
1978	257	12	30		3		3		1		306
Total	2,157	62	103	7,330	360	71	128	62	8	7	10,288

a. 10% before 1971 and 25% from 1971 to 1978.

Table 5-15 summarizes the maximum activity measured through *in vivo* monitoring for various nuclides for the period from 1961 to 1966.

Table 5-15. Maximum measured *in vivo* activity (nCi) from 1961 to 1966.

Isotope	1961	1962	1963	1964	1965	1966
Na-24			140			
Sc-46						4
Cr-51		320		42	Trace[a]	
Co-56					15	
Co-57						65
Co-58		20		38	Trace	
Fe-59		40		15		
Co-60	80	13	2	220	5	12
Co-64				<5		
Zn-65		40	3	16	Trace	
Se-75				250		
Sr-90			40	4,500	800	626
Zr-95/Nb-95	30	11	14	47	18	39
Ru-106/Rh-106		131		47	200	30
Sb-125		162	6	18		
I-131	12	200	28[b]	73	54	6
Cs-137	440	360	570	310	104	92
Ce-144/Pr-144		30		<7	75	50
Eu-155				Trace		
Au-198						
Hg-203	800	0.0005			169	22
Ra-226				<5		<0.5[c]
Pa-233	2,800					
U (enriched)				Trace		Trace

a. *Trace* is not defined within any of the annual reports, but is provided here to give a qualitative "feel" for the amount present.
b. Thyroid.
c. micrograms.

The results for a given *in vivo* analysis contained the following information: name, badge number (or equivalent), division code, HP area, date, analysis code, and results code. Table 5-8 lists result codes for the period prior to 1978. The reason for the analysis was given. The ORNL Division codes are the same as those listed in Tables 5-4 and 5-5. [NOTE: The ICRP Publication 2 methodology of reporting percentages of body or lung burdens (ICRP 1959) was used at ORNL through 1983. AEC Manual Chapter 0502 required "an evaluation of the radiation exposure status of an employee when monitoring techniques indicated that a body burden equaled or exceeded 50% of a maximum permissible limit." A phone discussion with the current internal dosimetry staff (McLaughlin 2004) indicated that the staff that was in place in the 1960s through 1980s probably conducted dose assessments when necessary and that documentation would exist for these assessments. Since 1989, each *in vivo* measurement showing a positive result has been assessed, including making estimates of intake and dose. The ORNL Internal Dosimetry Program TBD (McLaughlin 2002) states that dose estimates of less than 1 mrem are reported as zero. The overwhelming number of *in vivo* measurements obtained at the facility indicate no elevated activity (McLaughlin 2004).

Mani (1983) indicated that a NaI-CsI phoswich detector was installed and operational in 1967, though Brown, Patterson, and Abee (1971) state, "The several detectors used for measuring gamma radiation from the human body are all NaI (Tl) crystals." The 1974 annual report of the Health Physics

Division (Auxier et al. 1975) stated that a phoswich detector was in place and apparently was installed between 1971 and 1974, but no other information was located stating when the detector was placed in the WBC facility. A 9- by 9-in. NaI (Tl) crystal was installed in the WBC facility in 1974 (Auxier, Davis, and Turner 1975) and placed under the chest area of the counting bed. This detector was used for stationary chest counts. (A statement in Brown, Patterson, and Abee (1971) indicated that a distortion of the spectrum occurred if the large detector was used with the bed moving. Thus, it was not used for whole-body scans.) The whole-body scans performed during this period used the 8- by 4-in. detector at a distance of 12 in. from the bed surface and an elapsed counting time of 20 minutes. (If transuranic isotopes were suspected, the thin crystal assembly was used.) Stationary chest counts used either the 8- by 4-in. or thin crystal assembly in contact with the chest depending on what contaminants were expected. When the 9- by 9-in. NaI (Tl) crystal was used the individual was positioned either face-up or –down on the bed and the bed was lowered just shy of contact with the face of the large detector. The time to conduct stationary chest counts typically ranged from 10 to 40 minutes and sometimes longer to ensure "better counting statistics" (Brown, Patterson, and Abee 1971). The arc counting geometry was described in (Brown, Patterson, and Abee 1971) with the 8- by 4-in. detector positioned 6 ft above the center point of the counting bed. This geometry differs from that described in Mani's assessment of ORNL internal dosimetry data (Mani 1983). The counting bed was placed on the floor with the ends supported to give the proper curvature. This geometry was intended for use only when the count rate was too high for the scanning or stationary chest counting methods. (No evidence of the arc counting geometry actually being used was noted in the references reviewed in the generation of this document.)

In addition to the NaI (Tl) detectors described above, Brown, Patterson, and Abee (1971) indicated that three other detectors were in use within the counting room in 1971. A 3- by 3-in. NaI crystal was used to measure radiation emanating from specific organs (e.g., mainly for the thyroid, kidneys, liver, and spleen). Because of its smaller size, it could be placed directly over the organ of interest. Two other NaI crystals were employed in the WBC facility as wound probes: 2- by 2-in. and 1- by 0.03125 in., with the latter used to measure low-energy photons less than about 200 keV.

The next bulk sources of information describing instruments and procedures in the WBC facility were provided by Berger and Goans (1981), Berger and Lane (1981, 1982, 1984), and Berger (2004). Berger (2004) indicated that use of the phoswich detector began in late 1976 to early 1977. Until that time, the three thin NaI(Tl) crystal detector assembly was in use for detection of low-energy photons. The phoswich detector had the following dimensions: 5.25- by 0.0625-in. NaI - 2-in. CsI. Mani (1983) indicates that in 1978, an 80-cm^2 HPGe counting array was in the WBC facility, but it was not fully operational until May 1980 (Berger and Lane 1981). The array consisted of six separate detectors in an aluminum block in a close-packed rectangular array (e.g., a six-die pattern) designed to cover one lung (Berger and Lane 1981). The phoswich and HPGe detectors were used together from the early 1980s to obtain lung count data; they were positioned over the right and left lungs, respectively. Berger and Lane (1984) indicated that there were problems with getting all the HPGe detectors positioned close to the chest surface due to the construction of the aluminum block. Detectors on the chest surface were almost a factor of 3 more sensitive than those farther away from the chest. This lung count geometry was used until approximately 1987, when problems developed with the HPGe/liquid nitrogen feed system (Hembree 2004). At that time, the HPGe detector array was taken out of service, and the phoswich detector was the only device used for routine chest counts until 1994.

In 1992, a Nuclear Data SBC was put in service, primarily to measure mixed fission/activation products ranging in energy between 100 and 2000 keV, and is in use today. This counter is in Room 16 of Building 2008, and is the only *in vivo* monitoring device outside the shielded counting room (Watts et al. 1995). It has three germanium detectors facing downward and two NaI detectors on the

sides of the bed. As described below, it can also be used as a backup chest counter, if necessary, due to its 3-to-50% efficient germanium detectors (Watts et al. 1995). The detectors are in a solid center shield. The shield is designed so there is a minimum of 4 in. of lead around each detector.

During this time, the 5.25- by 0.0625-in. NaI - 2-in. CsI phoswich detector was used to conduct chest counts. Some time between 1977 and 1994, another phoswich detector with similar dimensions was procured and used until three 70% efficient germanium detectors were installed in the counting room in 1994 (Hembree 2004; McLaughlin 2004). A chair counting geometry was reinitiated at that time, with the individual positioned with his/her back reclined 45 degrees from the vertical. The three germanium detectors are positioned with two detectors over the right lung (one over the upper portion and the other over the lower portion) and one over the left lung. The third detector is over the upper portion of the left lung.

In addition to the above, there are three other detector arrangements in the WBC facility. A germanium thyroid counter can be set up using one of the detectors used for lung counting. The individual is in the same chair used for lung counting, but the chair is inclined at a 25-degree angle. Other organ-specific geometries can be used as necessary (skull, liver, hand, etc.). The 5.25- by 0.0625-in. NaI - 2-in. CsI phoswich and the SBC (in a fixed-bed position) are backup systems that could be used for lung counts if needed.

Until 1994, field health physicists selected individuals for *in vivo* monitoring. At that time, the internal dosimetry group became responsible for the selection of personnel for *in vivo* monitoring with guidance from field health physicists. Berger (2003) indicated that the *in vivo* monitoring program may not have been effective for monitoring ORNL personnel until the late 1980s because of reliance on field health physicists to determine the individuals to be monitored.

5.3.3 Cesium Counting Artifact

Many ORNL workers had measured body burdens from intakes of ^{137}Cs from nonoccupational sources (e.g., fallout and consumption of local venison). At present, ORNL uses a value of 20 nCi ^{137}Cs in a whole-body count for consumers of venison as the decision level to follow up or conduct a dose assessment (ORNL 2002). However, the Laboratory has never used it to negate an individual's occupational dose that was received on the site (McLaughlin 2004).

Fallout affects everyone in North America, and body burdens of ^{137}Cs measurable in whole-body counters were common in the 1960s and 1970s. NCRP Report 94 (NCRP 1988) provides mean body burdens of ^{137}Cs for the United States for the years most likely to produce interference with occupational whole-body count results. Table 5-16 lists these values.

5.3.4 Minimum Detectable Activity

As stated above, the thin (5-in.-diameter by 0.0625 in.) NaI (Tl) crystal array used in 1963 to detect low-energy photons emitted primarily from transuranics could detect approximately 40 nCi of ^{238}Pu, if there had been a preexposure chest count. If there was no preexposure count, the detection capability was approximately 80 nCi (Morgan, Snyder, and Struxness 1963).

Table 5-17 lists MDAs for the HPGe array of six individual detectors in the aluminum block (used around early- to mid-1980s).

Table 5-16. Mean body burdens of ^{137}Cs from fallout in the United States.

Year	Body burden (nCi)
1953	0.27
1954	1.1
1955	2.2
1956	4.3
1957	5.1
1958	6.5
1959	8.1
1960	6.8
1961	4.6
1962	6
1963	11
1964	19
1965	16

Year	Body burden (nCi)
1966	9.7
1967	5.6
1968	3.5
1969	2.7
1970	2.7
1971	2.7
1972	2.7
1973	2.7
1974	1.6
1975	1.1
1976	1.6
1977	1.1

Table 5-17. MDA values for six-detector HPGe array (Berger and Lane 1981).

Nuclide	Organ	MDA (nCi)
Pu-239	Lung	21.5
Am-241	Lung	1.1
I-125	Thyroid	0.04
Cs-137	Lung	16.4
Gd-153	Lung	0.87
Eu-152	Lung	0.71
U-233	Lung	1.42
Co-57	Lung	0.94

Tables 5-18, 5-19, and 5-20 list MDA values for the ORNL SBC, germanium thyroid counter, and fixed bed counter, respectively (Watts et al. 1995). MDA values are calculated for each individual measurement made using the germanium lung counter, SBC, and fixed bed counter. Figure 5-1 is a plot of MDA values for different nuclides for the ORNL germanium lung counter as a function of chest wall thickness. Figure 5-2 shows MDA values for ^{238}Pu and ^{241}Am for the ORNL phoswich lung counter as a function of chest wall thickness. Table 5-21 lists the frequencies of *in vivo* counting since 1960 and, for the various detectors, a summary of the best estimates of the applicable periods of use.

Table 5-18. MDA values for ORNL scanning bed counter (NaI/Ge) (Watts 1995).

Isotope	MDA (nCi)
Cr-51	24/30
Fe-59	3.9/4.5
Rb-88	12/15
Zr-95	3.7/4.1
Ru-106	21/30
Sb-124	2.2/3.1
I-133	2.7/2.9
Ce-139	4.3/4.6
Ce-141	7.7/8.4
Gd-153	21/21
Bi-214	5.8/9.8

Isotope	MDA (nCi)
Mn-54	2.1/2.4
Co-60	2.2/1.9
Y-88	1.8/1.9
Tc-99m	4.4/5.3
Cd-109	NG/224*
I-131	2.7/3.1
Cs-134	2.2/3.0
Ba-140	9.2/9.1
Ce-144	36/43
Eu-154	6.1/7.2
U-235	5.9/6.1

Isotope	MDA (nCi)
Co-57	4.9/5.9
Zn-65	4.3/6.0
Nb-95	2.1/2.7
Ru-103	2.6/2.9
Sn-113	3.4/4.0
Ba-133	3.7/4.5
Cs-137	2.4/2.8
La-140	2.3/2.4
Eu-152	8.5/9.7
Pb-212	6.3/7.6

*NG = not given

Table 5-19. Current MDA values for ORNL germanium thyroid counter.

Isotope	MDA (nCi)
I-125	0.06
I-131	0.12

Table 5-20. Current MDA values for ORNL fixed bed counter.

Isotope	MDA (nCi)
Co-60	0.31
Ce-144	6.28

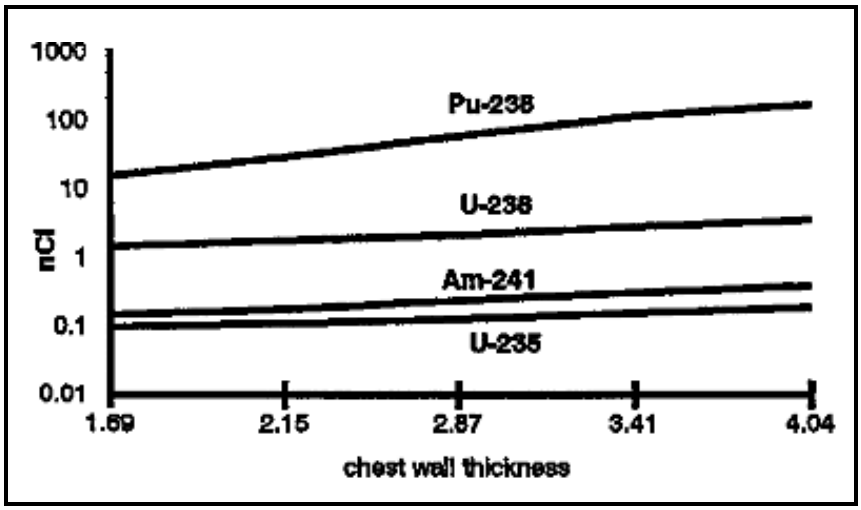

Figure 5-1. Current MDA values (nCi) for ORNL germanium lung counter versus chest-wall thickness (cm) (Watts et al. 1995).

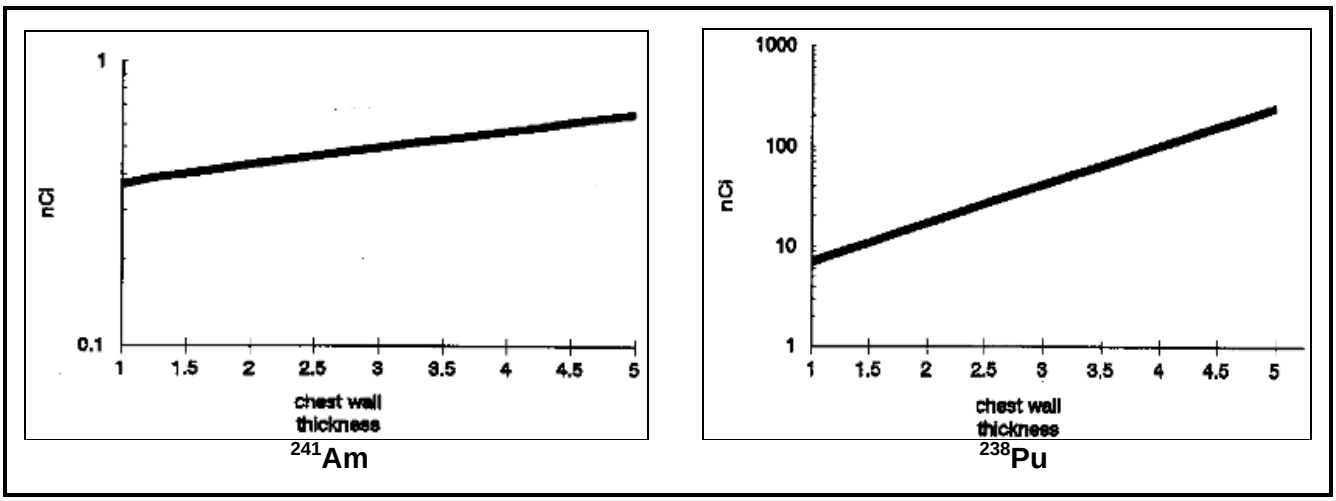

Figure 5-2. Current MDA values (nCi) for ORNL phoswich lung counter versus chest-wall thickness (cm) for ^{241}Am and ^{238}Pu (Watts et al. 1995).

Table 5-21. Frequency of *in vivo* monitoring.

Monitoring type	Applicable period	Frequency
Lung counting	1960–1961 4- × 4-in. NaI (Tl)	No set frequency. (Period of facility development –limited routine human counting, typically of incidents)
	1961–1962 8- × 4-in. NaI (Tl)	No set frequency. (Development activities to change over to larger detector – typically incidents)
	1962–1967-71 8- × 4-in. NaI (Tl)	No set frequency. (Period spent obtaining baseline counts onsite employees and incident counts)
	1965–?? 8- × 4-in. NaI (Tl)	Frequency stated in Annual Report (see Table 5-12), but discussions with site personnel indicate that it was not consistently implemented.
	ca 1967-71 to 1976 8- × 4-in. NaI (Tl) and 9- × 9-in. NaI (Tl)	Large detector was added below counting table and used when conducting stationary lung counts only.
	1962–1976 5-in. dia. × 1/16-in. NaI (Tl)	No set frequency. (Thin NaI crystals used to quantify low energy photons.)

Monitoring type	Applicable period	Frequency
Lung counting (cont'd.)	1976–1994 5.25- × 1/16-in. NaI - 2- CsI phoswich	Monitoring frequency was not effectively implemented. This phoswich detector was used to monitor for low energy photons.
	1980–1987 Germanium array and phoswich	Monitoring frequency was not effectively implemented. Detectors used together.
	1992–present combination germanium and NaI	Fixed Bed Counter is SBC in fixed position. Used only to measure fission and activation products (e.g., ^{60}Co and ^{144}Ce) immediately after assimilation.
	1994–present Germanium system	Varies from annual to semiannual depending on exposure potential set by Internal Dosimetrist.
	1994–present 5.25- × 1/16-in. NaI - 2- CsI phoswich	Phoswich detector can be used, but germanium detection system noted above is routinely used.
Whole-body counting	1960–1961 4- × 4-in. NaI (Tl)	No set frequency. (Period of facility development –limited routine human counting, typically incidents)
	1961–1962 8- × 4-in. NaI (Tl)	No set frequency (Development activities to change to larger detector – typically incidents)
	1963–1976 8- × 4-in. NaI (Tl)	Table 5-12 Selection criteria used in 1965 for WBC
	1992–present combination germanium and NaI	SBC. Monitoring frequency implemented in 1994 to include annual and semiannual counts depending on exposure potential set by Internal Dosimetrist.
Thyroid counter (and other organs)	1963–?? 3- × 3-in. NaI (Tl)	Incident counting and research. May have been used for frequent monitoring of workers potentially exposed to iodine, but this information was not located.
	1994–present germanium (uses one of three germanium detectors used for lung counting)	Typically, site research personnel would be placed on this program and monitored on semiannual frequency, as well as persons involved in known or potential iodine intake.

5.4 USING AIR MONITORING DATA TO ASSIGN BEST ESTIMATES OF DOSE

Information regarding internal dose at Oak Ridge National Laboratory (ORNL) from 1943 through 1950 is somewhat limited, with no bioassay results appearing in the DOE claimant files. However, there are fairly extensive air-monitoring records in the 1944 through 1947 time frame that are suitable for assigning best estimates of dose. The general protocol presented in OTIB-0018, "Internal Dose Overestimates for Facilities with Air Sampling Programs" is used with the following modifications:

1) The approach applies only to best estimates of doses received in the 1/1/1944 to 12/31/1950 time period at ORNL.

2) For the alpha-emitter category, the radionuclides are limited to plutonium and uranium.

3) A constant distribution is used in IREP for the doses rather than a lognormal distribution.

The technical justification for this approach for 1944 through 1947 is given in Attachment B.

Air monitoring data are not available for 1948 through 1950. A fit to the coworker data for 1951 and 1952 indicate an intake rate of 200 pCi/day of Type S ^{239}Pu for 1948 through 1950 [6] whereas application of the OTIB-0018 protocol to the same period gives an intake rate of about 198 pCi/day[2]

[2] See Table 4-1 in OTIB-0018 (ORAUT 2005a).

for gross alpha (the OTIB-0018 intake rate is the same for both Type M and Type S). This supports extending the use of OTIB-0018 from 1947 through 1950, assuming that the exposure potential did not increase between 1947 and 1952. A review of ORAUT-TKBS-0012-2, "Oak Ridge National Laboratory – Site Description" indicates that, while the site was by no means static during this period, the assumption of relative constant exposure potential is reasonable. Thus, the OTIB-0018 protocol should be applied to calculate intake rates for 1/1/1948 through 12/31/1950 with the same modifications used for 1/1/1944 through 12/31/1947 [7].

5.5 ATTRIBUTIONS AND ANNOTATIONS

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

Kenny Fleming served as one of the initial Subject Experts for this document. Mr. Fleming was previously employed at ORNL and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision and earlier revisions have been overseen by a Document Owner who is fully responsible for the content, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by Mr. Fleming, those materials are fully attributed to the source.

Michael Bollenbacher served as one of the initial Subject Experts for this document. Mr. Bollenbacher was previously employed at ORNL and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision and earlier revisions have been overseen by a Document Owner who is fully responsible for the content, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by Mr. Bollenbacher, those materials are fully attributed to the source.

Elizabeth Brackett serves as a Site Expert for this document. As such, she is responsible for advising on site-specific issues and incidents as necessary and ensures the completeness and accuracy of the document. Because of her prior work experience at the site, she possesses, or is aware of information that is relevant for reconstructing radiation doses experienced by claimants who worked at the site. In all cases where such information or prior studies or writings are included or relied upon by the document owner, those materials are fully attributed. Ms. Brackett's Disclosure Statement is available at www.oraucoc.org.

[1] Burns, Robert E. Shonka Research Associates. Senior Health Physicist. April 2007. Section 2 of this Site Profile (ORAUT 2006) provides a more detailed discussion of the key radionuclides workers encountered.

[2] Burns, Robert E. Shonka Research Associates. Senior Health Physicist. April 2007. This list was compiled from queries of the database of *in vitro* bioassay results provided by ORNL.

[3] Burns, Robert E. Shonka Research Associates. Senior Health Physicist. April 2007. The system of absorption types promulgated in ICRP Publication 66 (ICRP 1994) is used for

all inhalation dose reconstructions under EEOICPA, as this represents use of the best available science. Use of older solubility class information or assumptions to maximize committed dose could result in organ dose assessments that would not be favorable to the claimant.

[4] Burns, Robert E. Shonka Research Associates. Senior Health Physicist. April 2007.
These codes are from worker dosimetry records.

[5] Burns, Robert E. Shonka Research Associates. Senior Health Physicist. April 2007.
The selection of ^{239}Pu as the default radionuclide, when the true isotopic mix is not known from measurement, is made to ensure assessments favorable to claimants.

[6] LaBone, Thomas R. ORAU Team. Deputy Principal Internal Dosimetrist. September 2007.
The urinary excretion in 1951 (0.13 dpm/day on 7/1/1951) and 1952 (0.11 dpm/day on 7/1/1952) from OTIB-0034 (ORAUT 2005b) were modeled with a chronic intake of Type S ^{239}Pu from 1/1/1948 to 12/31/1952. The calculated intake rate was 200 dpm/day.

[7] Brackett, Elizabeth M. ORAU Team. Principal Internal Dosimetrist, and LaBone, Thomas R. ORAU Team. Deputy Principal Internal Dosimetrist. October 2006.
The changes to Section 5.4 were initiated by Brackett and LaBone and the calculations were performed by LaBone.

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A.1 Recovery of Historical Data

A large amount of raw bioassay data exists at the ORNL complex. A portion of these data was transcribed into an electronic database hosted on ORNL equipment and maintained by ORNL personnel.

As part of this dose reconstruction effort, this database was queried for historical urinalysis records on October 30, 2003. A supplemental query was performed on December 12, 2003. These queries requested information on the samples and the instrumentation used in the analysis. No personal information or individual dose results were requested or received.

These database queries generated 66,204 records from the period between 1945 and 1988. These records were compiled into a single database. A data validation and comment field was added to each record. Then the records were examined and classified according to usability.

A.1.1 DATA DESCRIPTION

The database queries retrieved 17 parameters from the ORNL internal dosimetry database:

- NUCLIDE – A unique abbreviated identifier for specific nuclides
- SAMPLEID – A unique identifier for individual samples
- SAMPLETYPE – A one-character identifier designating the type of sample (urine, fecal, etc.)

- ALIQUOTVOL – The volume or mass of the sample analyzed
- SAMPLEVOL – The sample volume or mass collected
- COUNTDATE – The date the aliquot was counted
- STARTEDATE – The date sample processing began
- STOPDATE – The date sample processing ended
- RECOVERY – The chemical yield, in percent
- EFFICIENCY – The detector's counting efficiency, in percent
- EFF_FACTOR – The reciprocal of the detector's counting efficiency
- COUNTMIN – The sample count time, in minutes
- T_BKG_RATE – The total background count rate, in counts per minute
- BKG_MIN – The background count time, in minutes
- BKG_RGNT_R – The count rate of any reagent used
- BKG_COUNTS – The number of background counts
- MDA – The minimum detectable activity

A.1.2 DATA CLASSIFICATION

The 66,204 individual records in the database were classified into one of four categories:

- **Duplicate Records** - 272 duplicate records were found. These were marked with a "D" in the record's validation field.

- **Usable without Modification** - 58,972 of the records collected (89%) were usable without qualification.

- **Usable with Modification** - Some of the records collected contained input errors that were easily identified. Examples include a nuclide entry of C-137, instrument efficiencies listed in fractions instead of percent, and dates from the turn of the century. Other information in the database enabled correction of some of these records. Such modifications were designated by placing an "M" in the records validation field and entering the reason for the change in the comment field. These records comprise 3% (2,193) of the database.

- **Unusable Records** - 4,767 of the records recovered (7%) were determined to be unusable in their present form. Most of these records are incomplete or contain gross errors. These were marked with an R in the validation field. An explanation for a record's rejection was entered in the comment field.

The resulting database provided all the raw data used to determine the MDAs of radionuclides in urinalysis calculated in this attachment.

A.1.3 SYNOPSIS OF RECOVERED DATA

The database queries recovered 61,165 usable records. Of these, 59,288 are clearly records of urinalyses and 1,859 are records of fecal analyses. Special analyses made up the remaining 18 records. Seventy-five different analyte labels are listed, although some are clearly different names for

the same analyte (e.g. GA, GAO, and G_ALPHA are all gross alpha measurements). Table A-1 lists the recorded analyses performed by radionuclide.

A.2 Calculation Method for Sample MDAs

Equations A-1 and A-2 show the derivation of the estimated MDA, which was calculated from 61,165 analytical records. MDAs were calculated using the method in Brodsky (1992):

$$\mathrm{MDA} = \frac{3 + 4.65 \times \mathrm{SD_b}(\mathrm{cnts})}{\mathrm{T_b}\ (\mathrm{min}) \times \mathrm{k}\left(\dfrac{\mathrm{cnts} \times 24\ \mathrm{h\ sample}}{\mathrm{dis}}\right)} \qquad \text{(A-1)}$$

where

MDA	=	Minimum Detectable Activity (dpm/24 h sample)
SD_b	=	Standard deviation of the total background counts (cnts/min)
T_b	=	Duration of the background count (min)
k	=	Calibration factor (cnts•24 h sample/dis)

$$\mathrm{k}\left(\frac{\mathrm{cnts} \times 24\ \mathrm{h\ sample}}{\mathrm{dis}}\right) = \mathrm{InstrEff}\left(\frac{\mathrm{cnts}}{\mathrm{dis}}\right) \text{ x Yield x } \frac{\mathrm{V_a}\ (\mathrm{mL})}{\mathrm{V_r}\left(\dfrac{\mathrm{mL}}{24\ \mathrm{h\ sample}}\right)} \qquad \text{(A-2)}$$

where:

InstrEff	=	Instrument efficiency (cnts/dis)
Yieldi	=	Fractional chemical recovery
Va	=	Volume of the aliquot analyzed (mL for urine, g for feces)
Vr	=	Volume of the sample submitted, or the volume of a standard 24-hr sample from reference man (1,400 mL/24-hr sample for urine, 135 g/24-hr sample for feces) (ICRP Publication 23 [ICRP 1975] default values for 24-hr voids), whichever is larger

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Table A-1. Data used to determine historical MDAs for radionuclides in urine and feces.

Nuclide	Fecal	Urinalysis	Row totals
Totals	**1,859**	**59,288**	**61,147**
Alpha	1	3	4
Am-241	130	5,540	5,670
Am-243	6	6	12
As-74		7	7
BG		2	2
Bk-249	2	12	14
Br-82		2	2
Br-83		2	2
C-14		11	11
Ca-45		4	4
Ce-144	4	33	37
Cf-249	2	1	3
Cf-252	3	11	14
Cl-36		1	1
Cm-242	1	11	12
Cm-244	26	273	299
Co-60	2	81	83
Cs-134		1	1
Cs-137	4	3,557	3,561
Fe-59	1	8	9
GA	69	1,875	1,944
G-alpha		6	6
GAO	154	2,771	2,925
GB		278	278
GBO		46	46
H-3		2,070	2,070
I-131		41	41
Mn-54		2	2
Mo-99		1	1
Na-24		3	3
Nb-95		3	3
Np-237	1	54	55
P-32		166	166
Pa-231	1	54	55
Pa-233	3	13	16
Pa-234		1	1

Nuclide	Fecal	Urinalysis	Row totals
Pb-210		2	2
Pm-147	16	64	80
Po-210	9	57	66
Pu-238	4	61	65
Pu-239	124	15,352	15,476
Pu-241	1	111	112
Pu-242		41	41
Ra-226	1	332	333
Ra-228		1	1
RE	39	698	737
Ru-103		1	1
Ru-106		65	65
S-35		10	10
Sb-125		1	1
Sm-151		11	11
Sr-85		1	1
Sr-89		37	37
Sr-90	142	12,751	12,893
T	1		1
Tc-99		19	19
Th-232	1,079	46	1,125
Tl-201	1		1
Tl-204		1	1
Tm-170	1	5	6
TPU		1	1
TRE	2	197	199
TRE B		1	1
TRE-B		4	4
U-232		1	1
U-233	3	826	829
U-235		3	3
U-238	25	11,409	11,434
U-239		11	11
Y+RE		157	157
Y-88		5	5
Y-90		31	31
Zn-65		7	7
Zr-95	1	19	20

To use this equation one must know, or be able to calculate, the standard deviation of background counts, the duration of the count, the yield of any chemical extractions used, the efficiency of the detector used, and the amount of the sample analyzed. The method of recovering this information from the historical database is discussed in the following sections.

A.2.1 STANDARD DEVIATION OF BACKGROUND COUNTS

The data field BKG_COUNTS contained recorded background counts for a sample and was the preferred source of background count information. When a record's BKG_COUNTS field was blank, the background count information was calculated as the product of the T_BKG_RATE and BKG_MIN fields. If the BKG_MIN field was blank, but the T_BKG_RATE field contained a value, the background count information was calculated as the product of the T_BKG_RATE and COUNTMIN fields.

The variability in repeat counts, such as background counts, follows the Poisson distribution (Evans 1955, Chapter 26). It is a property of the Poisson distribution that its standard deviation is the square root of its mean value. Therefore, in this analysis, the standard deviation of the total background counts was calculated as the square root of the total background counts.

A.2.2 BACKGROUND COUNT TIME

The data field BKG_MIN contained the recorded background count time for a sample and was the preferred source of background count time information. If the BKG_MIN field was blank, the time in the COUNTMIN field was used. This last method of estimating the background count time is likely to be favorable to claimants because in almost all cases where there is paired data, the COUNTMIN value is less than the BKG_MIN value. This produces a higher MDA when substituted into Equation 1.

A.2.3 INSTRUMENT EFFICIENCY

Information on instrument efficiencies was contained in two fields. The EFFICIENCY field contained the percent of disintegrations detected by the instrument. The EFF_FACTOR field contained the reciprocal of the EFFICIENCY field.

A.2.4 YIELD

The chemical yield for a given radiochemical analysis was contained in the RECOVERY data field. This value was given in percent recovery.

Many of the records had blank recovery fields. However, due to the repetitive nature of the chemical extractions used, application of an average, radionuclide-specific, chemical recovery to records with no recovery value was judged to be acceptable. This approach is supported further by the lack of variability observed in daily runs on known standards recorded in laboratory logbooks recovered from the late-1950s through the mid-1970s.

A.2.5 VOLUME OF ALIQUOT ANALYZED

The ALIQUOTVOL field usually contained a value for the amount of material subjected to radioanalysis. When this field was blank, it was assumed that the entire sample provided was used in the analysis. The amount of the sample was recorded in the SAMPLEVOL field.

A.2.6 VOLUME OF 24-HOUR SAMPLE

Excretion rates and concentrations vary greatly within a 24-hr period and between individuals. The models developed for calculating intakes from *in vitro* analyses are based on 24-hr samples. The daily urine excretion rate is 1,400 mL/24-hr sample; for feces, this value is 135 g/24-hr sample.

Often, the volume analyzed (V_a) was less than the reference excretion rate from standard man. When this was observed, the calculated activity in the sample was normalized to the reference man volume to compensate. This adjustment increased the MDA and, therefore, would be favorable to claimants.

A.3 Calculation Method for MDAs for Various Periods Between 1945 and 1988

Once the MDAs were calculated for individual samples, they were segregated by radionuclide and imported into Excel spreadsheets. Using Excel, the values were plotted for visual inspection. Figures A-1 through A-18 at the end of this attachment show plots of data the was obtained and calculated for the following radioisotopes: (urine) ^{241}Am, ^{244}Cu, ^{137}Cs, gross alpha, gross beta, ^{3}H, ^{147}Pm, ^{238}Pu, ^{239}Pu, ^{241}Pu, rare earths, ^{90}Sr, ^{233}U, ^{238}U, and (feces) ^{241}Am, gross alpha, ^{239}Pu, ^{232}Th. In the first plot, the MDA values were plotted against the date of their analysis. This provided a visual representation of individual MDAs over time. The density of the available data and a gross approximation of typical sample MDAs for a given period were quickly observed using this format.

Two bar charts were created for each radionuclide. One shows the annual average MDAs between 1945 and 1988. The other shows the number of analytical results recovered for each year in the same period. These two charts provide additional information on gross trends exhibited by sample MDAs, and on the temporal distribution of the data.

These visual aids were used to establish general periods during which sample MDAs were similar. The sample MDAs in these periods were grouped as similar populations. Using Excel's statistical functions, the following statistical parameters for each group of samples were determined:

- The population size (N)
- The 50th percentile value of the population,
- The population's arithmetic mean
- The arithmetic standard deviation of the population
- The 95% confidence level on the mean

The 95% confidence levels on the mean for each radionuclide are the recommended MDAs for that radionuclide during the period evaluated.

A.4 Additional Considerations

The recovered data indicate the analytical MDAs tended to remain fairly consistent for years. Abrupt changes in the MDAs were identified for groups of radionuclides during specific years. Following the change, the MDAs remained generally consistent in the several succeeding years. This "step-wise" pattern allowed MDAs from several years to be grouped as one MDA for a specified period. Table 5-9 highlights these groupings by enclosing similar MDAs in a box.

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Some MDAs changed frequently, creating a pattern of similar MDAs in short adjacent periods. When this occurred, the largest estimated MDA from that period was selected to be the representative MDA for the entire period. The tritium (^{3}H) MDA listed between 1968 and 1981 in Table 5-9 is an example of this adjustment.

Most recovered data contained isotope-specific entries in the NUCLIDE field. However, the ability to differentiate routinely between isotopes of the same radioelement did not always exist before 1989. Many of the isotopic assignments were based on process knowledge. Therefore, during the period prior to 1989, the MDA for some isotopes are reported as the MDA for the radioelement instead of the radionuclide. For example, the alpha emitters ^{238}Pu and ^{239}Pu are reported separately in the database. They have been combined in this TBD as plutonium. NOTE: ^{241}Pu is reported separately because it is a beta emitter that can be measured separately from the alpha emitting isotopes of plutonium. If the recovered data did not extend to 1989, the MDA calculated from the last known data was extrapolated forward (see Table 5-9).

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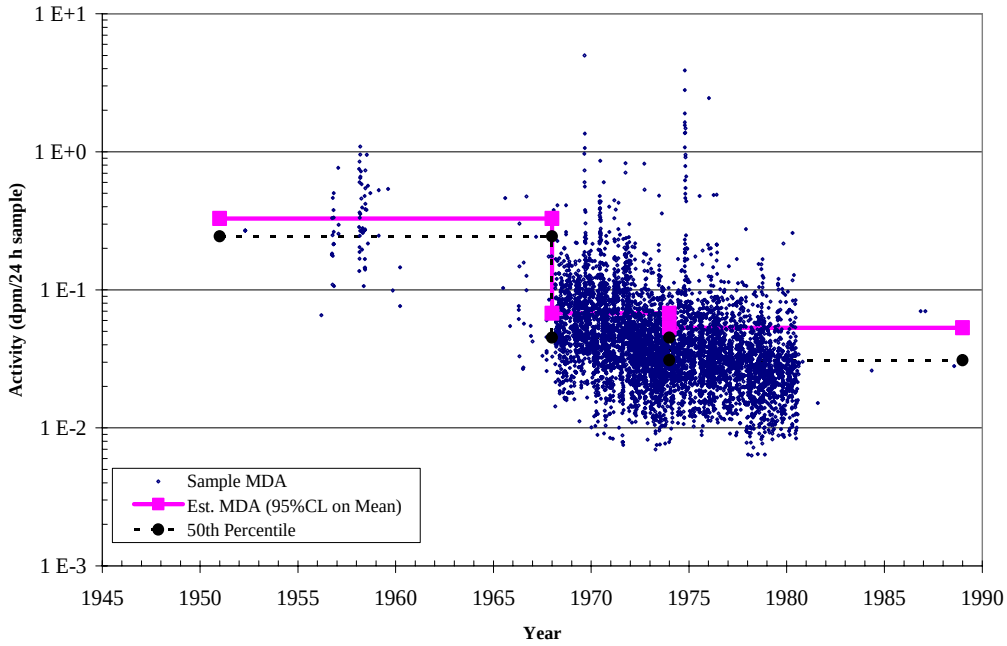
Table A-2. MDAs (dpm/24-hr sample) calculated from recovered data for radionuclides.

	Urine MDAs (dpm/24h sample)																	Fecal MDAs (dpm/24h sample)				
Year	Gross alpha	Gross beta	Am-241	Cm-244	Cs-137	H-3	I-131	Np-237	Pm-147	Plutonium	Polonium	Pu-241	Ra-226	Rare earths	Ru-106	Sr-89 + Sr-90	Uranium	Gross alpha	Am-241	Cm-244	Plutonium	Th-232
1943																						
1944																						
1945																						
1946																						
1947										0.33												
1948										0.40							0.965					
1949										0.34							0.998					
1950					24.34					0.19						53.95	1.401					0.176
1951					29.54					0.18				33.34	0.036	22.34	3.600					
1952		74.44	0.27		34.93					0.23			25.44	31.41		27.50	5.988					
1953	0.17				29.49					0.19	2.98			30.45		27.01	6.196					
1954	0.25	351.14			32.90					0.17	9.42		0.32	41.97		29.00	6.296					
1955	0.22	960.93			26.52					0.19	33.32		0.87		0.142	26.76	6.810					
1956	0.23	889.95	0.25		32.05					0.22	57.11		0.16		0.238	32.75	7.651					
1957	0.69	998.73	0.44		30.61			1.07		0.39			0.80		0.272	28.08	5.257	0.055			0.072	0.226
1958	0.35	1385.09	0.42		38.61			0.26		0.19			0.34		0.255	47.99	7.423	0.728				0.537
1959	0.31	1321.55	0.35		312.8			0.10	587	0.23			0.29	331.74	0.359	36.83	5.636	0.647				0.303
1960	0.19	938.01	0.11		147.3			0.27	196	0.18	10.67		0.17	167.87	61.85	19.60	5.740	0.568				0.573
1961	0.21	290.98			42.62	71382				0.18	12.74		0.29	50.04		5.86	5.686	0.512				0.424
1962	0.32	252.50				641672				0.14				54.69		8.88	8.687					0.422
1963	0.45	146.59			72.18	393539		0.49	25.81	0.10				87.30		14.79	7.747	1.075			0.789	0.633
1964	0.24	198.28		0.57	62.09	214873				0.19					78.91	12.26	2.561				0.118	1.052
1965	0.46	140.95	0.21	0.17	55.66	124229	353.2	0.08		0.14				63.41		13.78	0.992	0.848	0.208	0.172	0.163	0.112
1966			0.12	0.16	52.71	153077				0.08						6.58	0.128	0.447		0.268		
1967			0.082	0.099	75.55	150699	35718			0.08						7.30	0.405				0.769	
1968	0.056	36.17	0.073	0.080	55.59	54872	1131	0.07		0.07	2.55	25.76	15.70		49.40	6.68	0.112		0.079	1.059	0.349	
1969	0.066	38.82	0.099	0.082	50.80	53036	1281	0.09	23.10	0.07	31.75	7.66		56.96	46.95	4.73	0.094		0.768	0.306	0.536	
1970	0.038	32.31	0.074	0.096	44.32	48763	1140	0.05		0.05		5.67				4.05	0.071		0.097	0.271	0.097	
1971		37.06	0.072			49037		0.06	677	0.06		6.00			2.82	4.33	0.076		0.230		0.351	
1972	0.18		0.046		75.05	45882		0.04	24.06	0.05		9.18		5.00		3.95	0.061	0.093	0.035		0.033	
1973	0.10		0.039			50535			6.44	0.04		6.94		5.44		4.14	0.071		0.105		0.060	
1974			0.093			46154			5.72	0.04		4.53		4.86		4.80	0.054		0.543		1.073	1.040
1975			0.040	0.032		44192				0.04	0.46			1.49		3.26	0.054		0.047	0.129	0.073	0.046
1976			0.050	0.080	46.12	45554				0.04		1.84	0.014	2.24		3.58	0.057		0.173		0.053	
1977			0.037			36984			2.82	0.04		9.65	0.049			3.62	0.053		0.121		0.100	
1978			0.032	0.016		31838		0.14		0.03		8.16		3.39		3.30						
1979			0.030	0.046		35406		0.03		0.03	0.23	12.40				3.06			0.029		0.032	
1980			0.032	0.036		48449										3.05						
1981			0.015																			
1982																						
1983																						
1984			0.026																			
1985																						
1986			0.07																			
1987			0.07																			
1988			0.028																			



Visual Comparison of Data to Estimated Am-241 MDAs in Urine

Figure A-1. MDA data for americium-241 in urine (part 1 of 3).

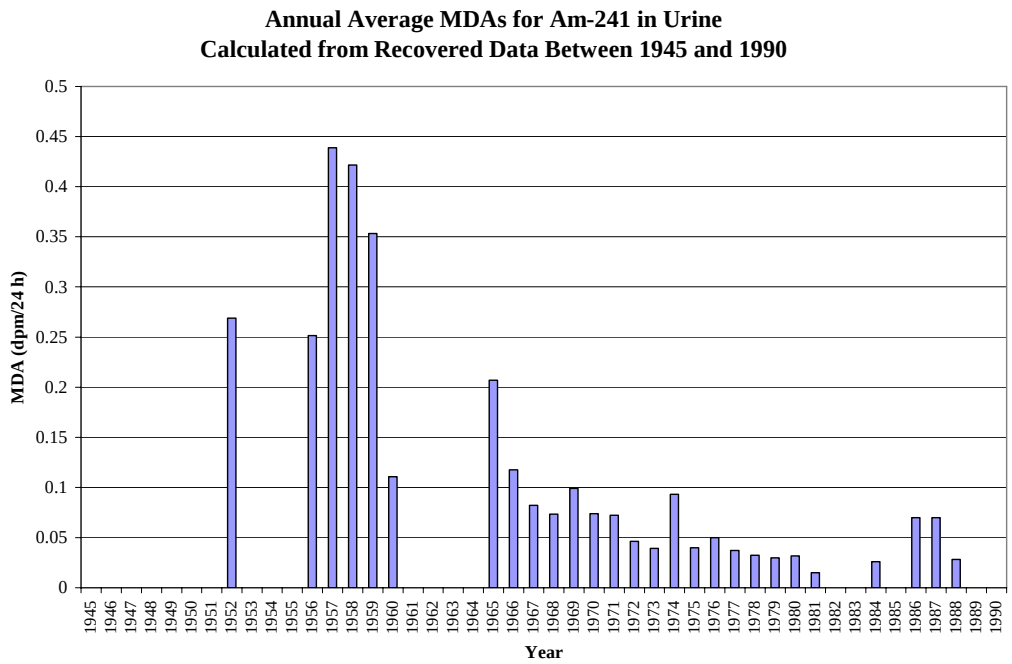


Figure A-1. MDA data for americium-241 in urine (part 2 of 3).

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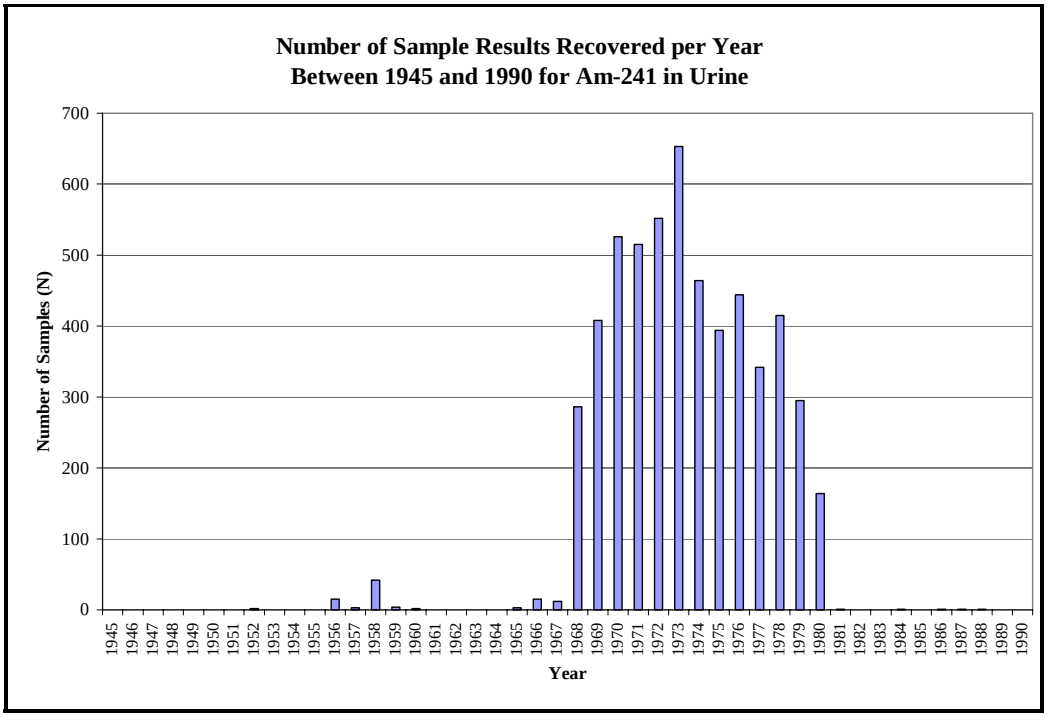

Figure A-1. MDA data for americium-241 in urine (part 3 of 3).

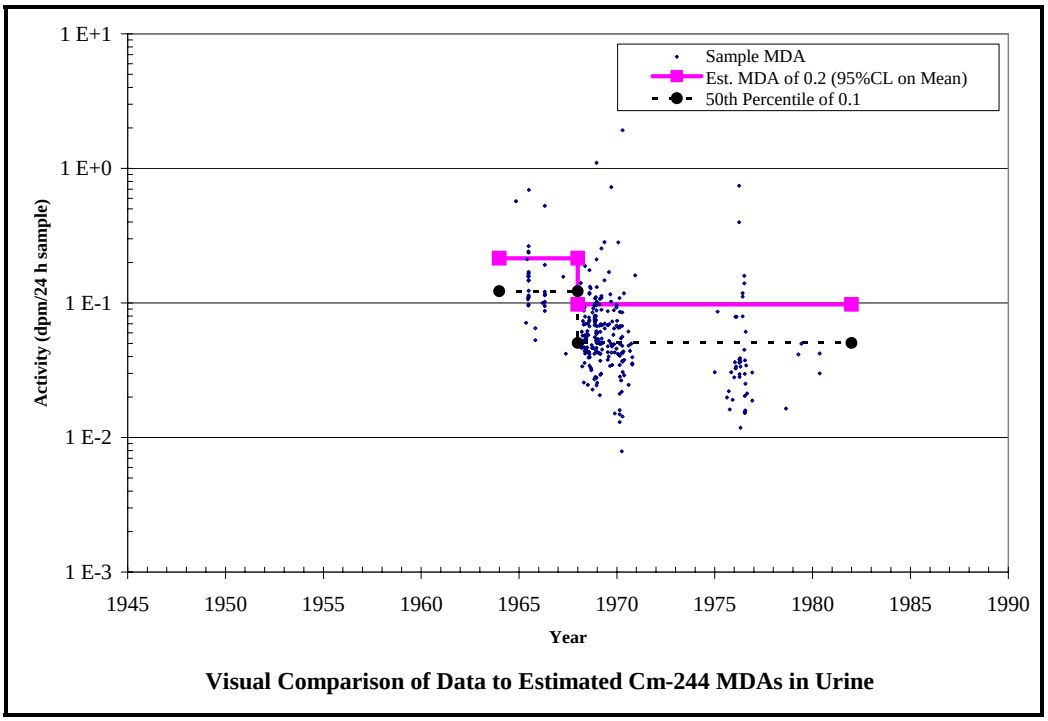

Figure A-2. MDA data for curium-244 in urine (part 1 of 3) (some later tables are “in feces”).

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CALCULATION OF MDAS FROM HISTORICAL DATA
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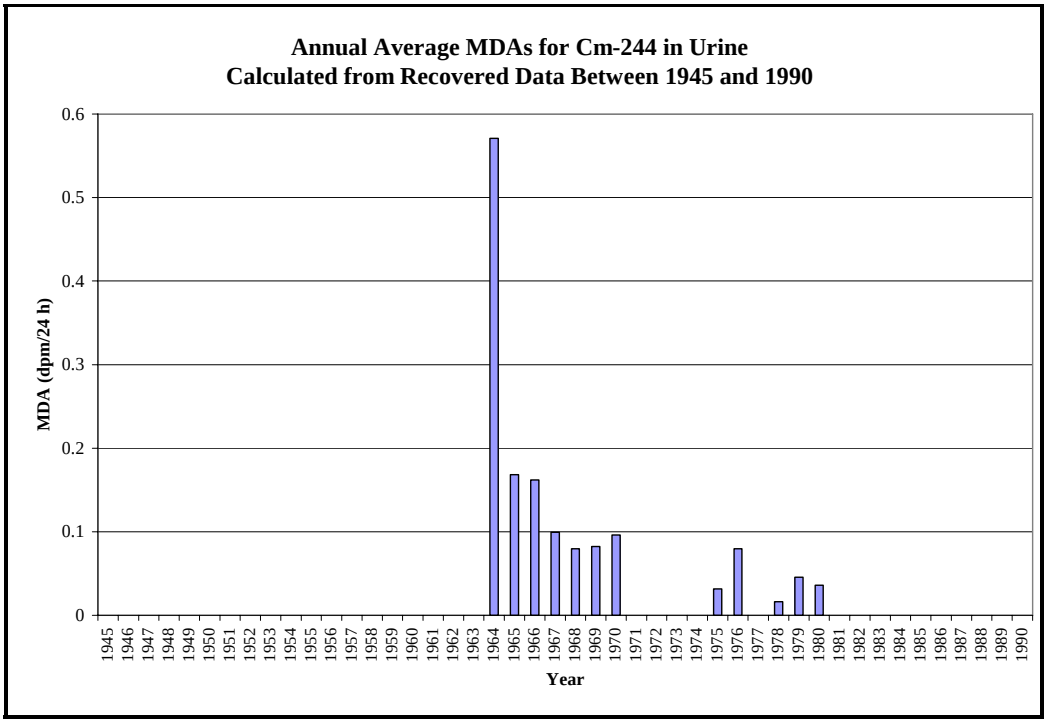

Figure A-2. MDA data for curium-244 in urine (part 2 of 3).

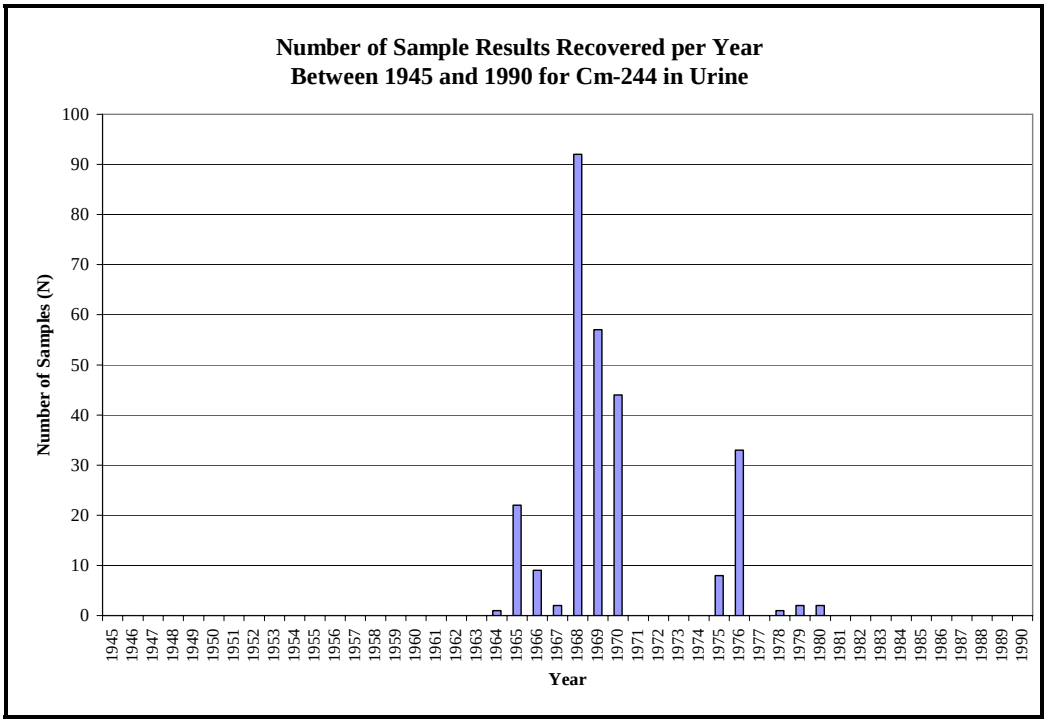

Figure A-2. MDA data for curium-244 in urine (part 3 of 3).

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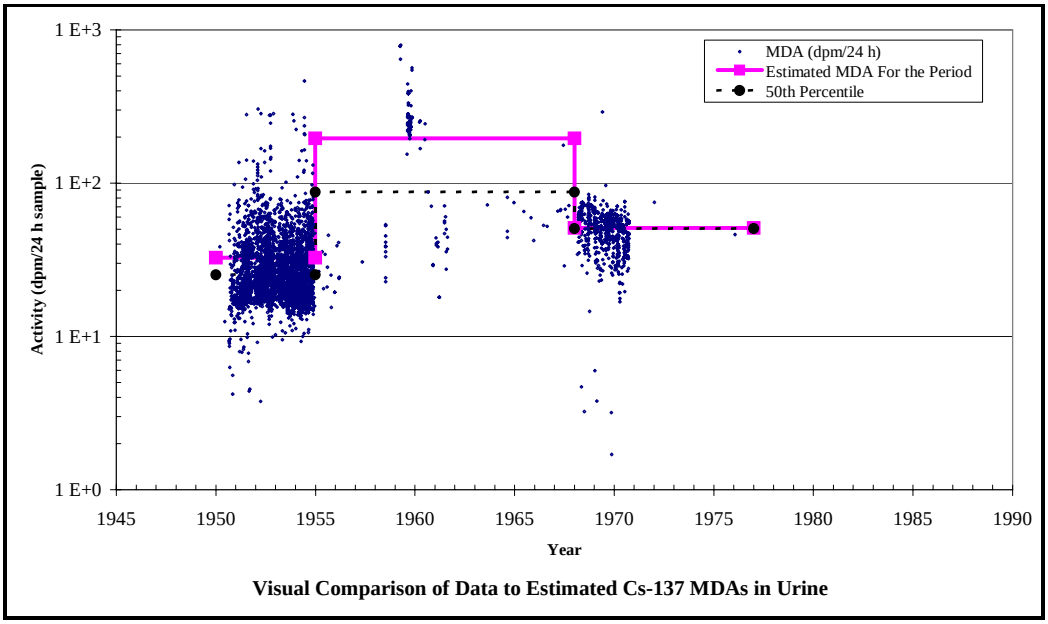

Figure A-3. MDA data for Cs-137 in urine (part 1 of 3).

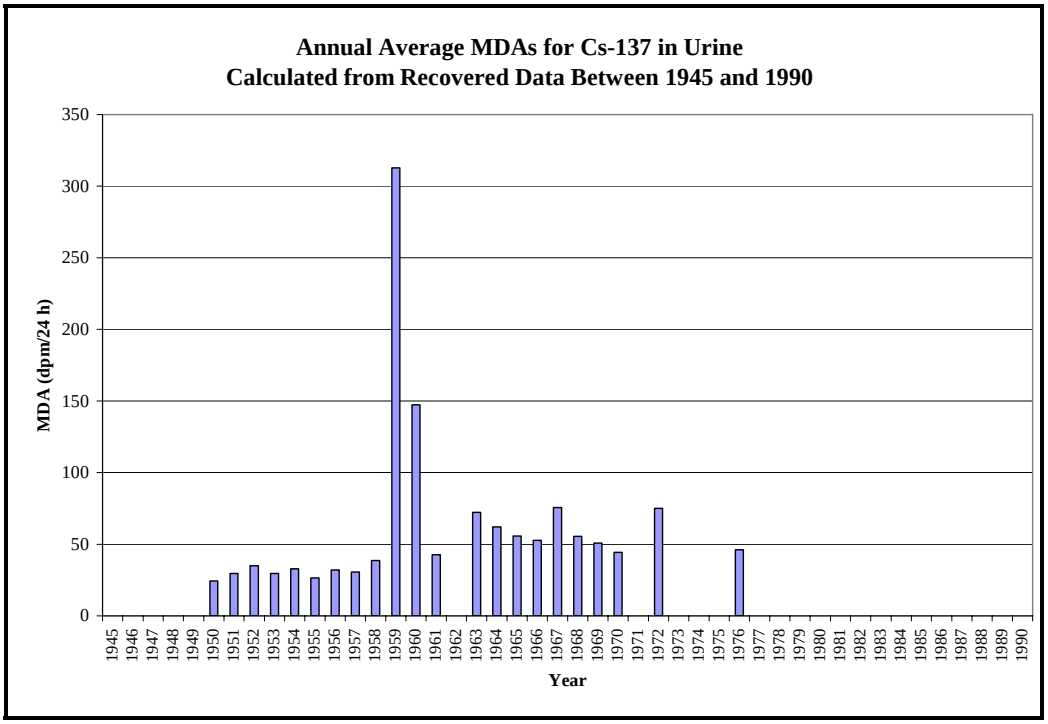

Figure A-3. MDA data for Cs-137 in urine (part 2 of 3).

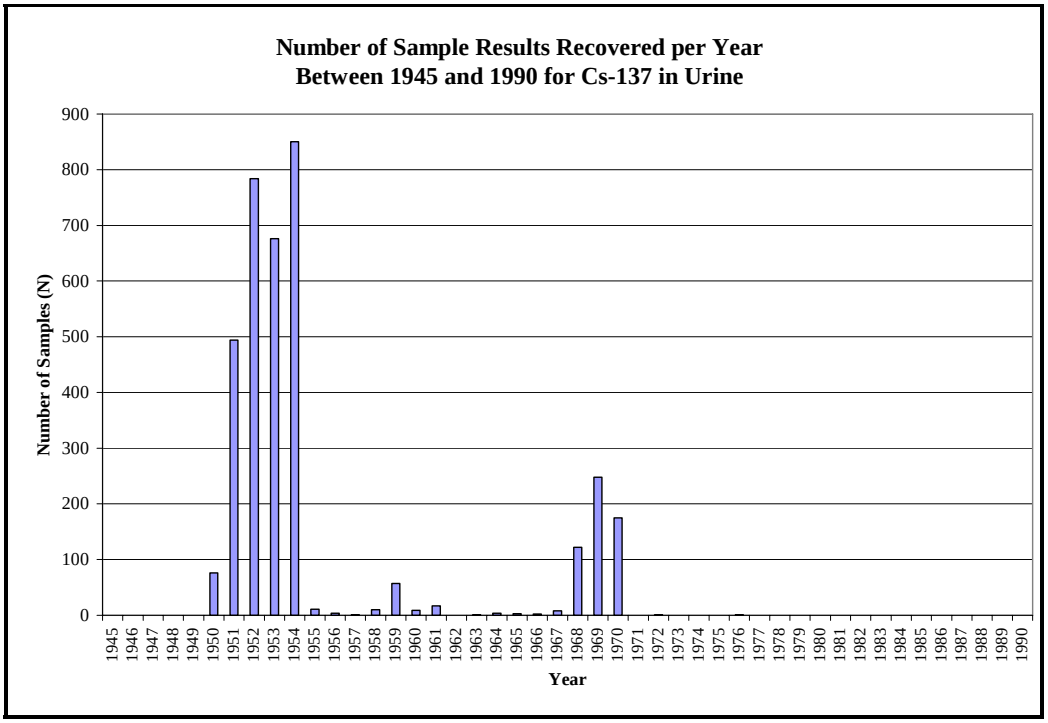

Figure A-3. MDA data for Cs-137 in urine (part 3 of 3).

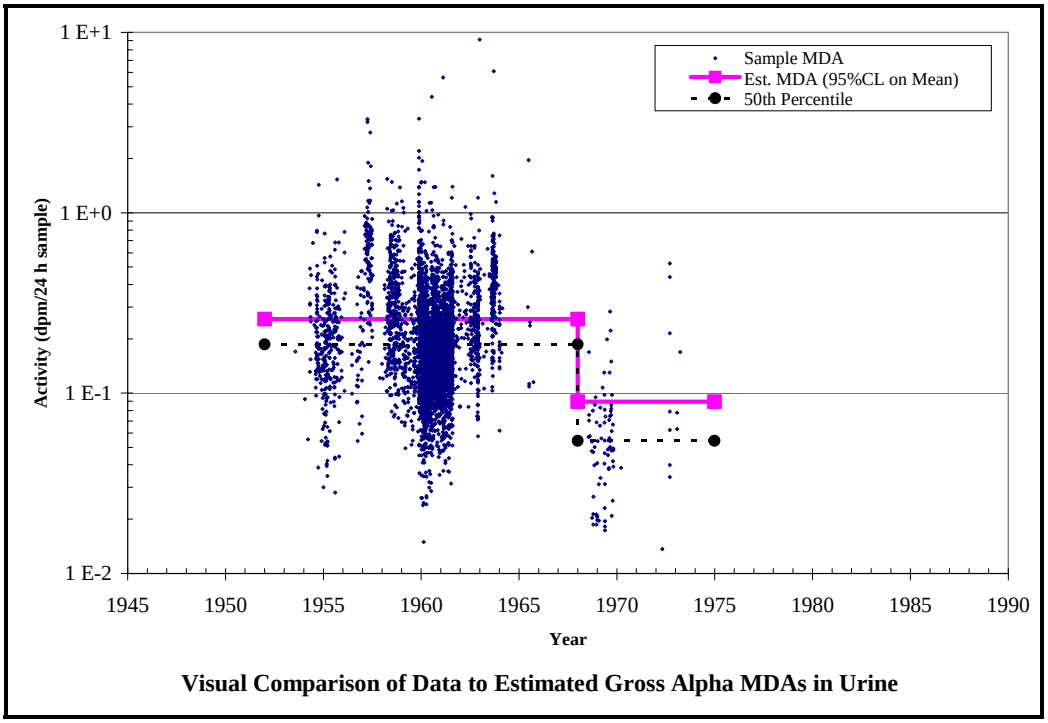

Figure A-4. MDA data for gross alpha in urine (part 1 of 3).

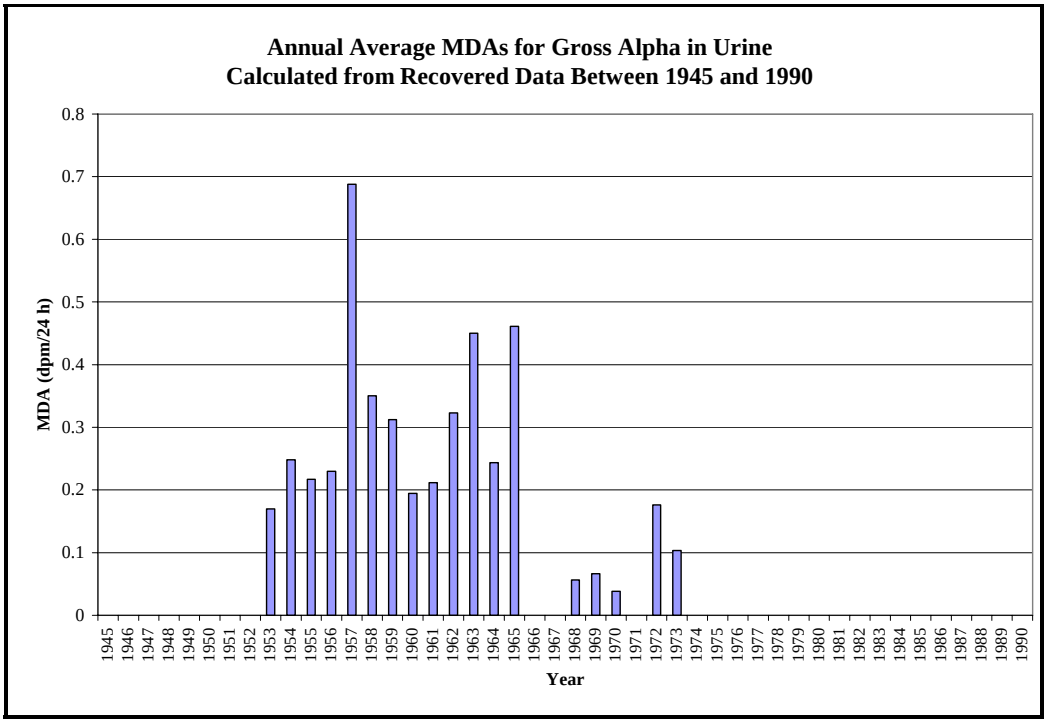

Figure A-4. MDA data for gross alpha in urine (part 2 of 3).

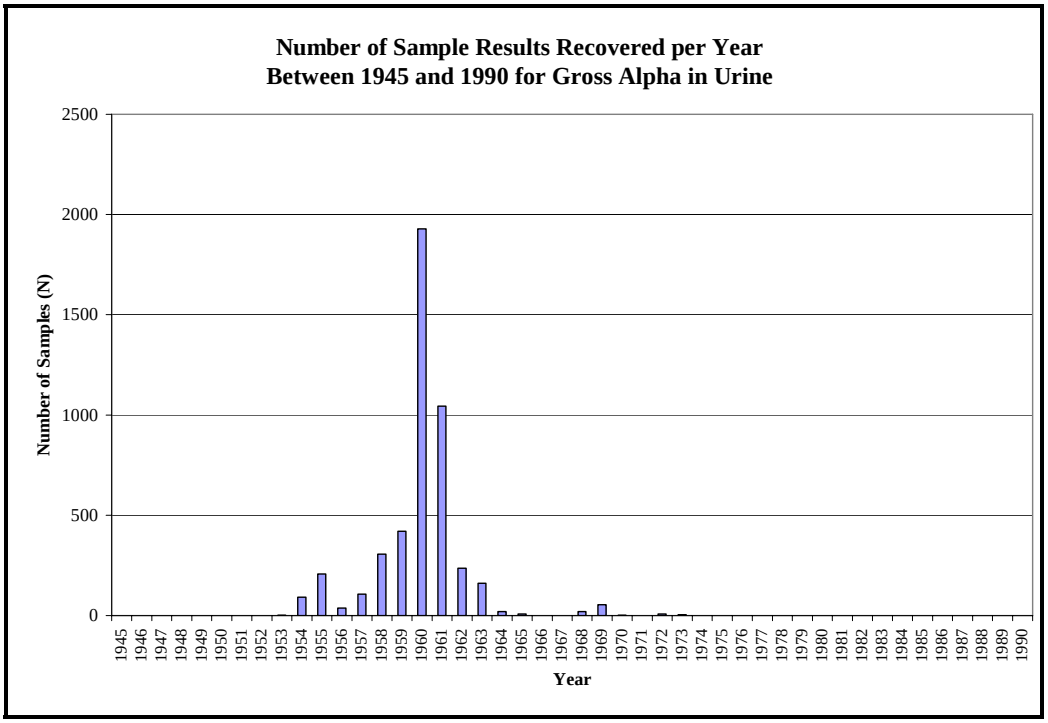

Figure A-4. MDA data for gross alpha in urine (part 3 of 3).

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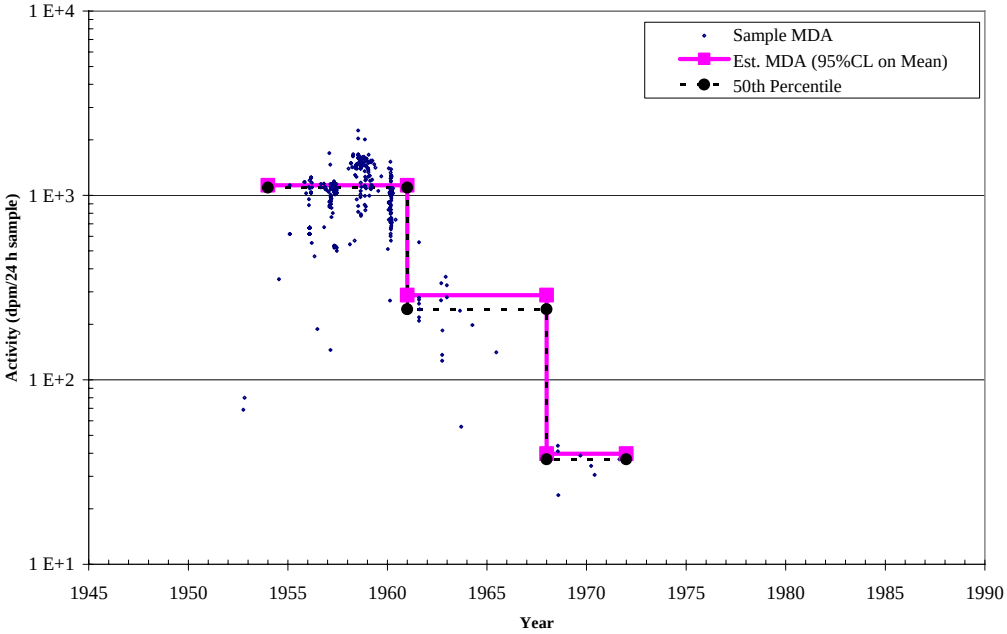


Visual Comparison of Data to Estimated Gross Beta MDAs in Urine

Figure A-5. MDA data for gross beta in urine (part 1 of 3).

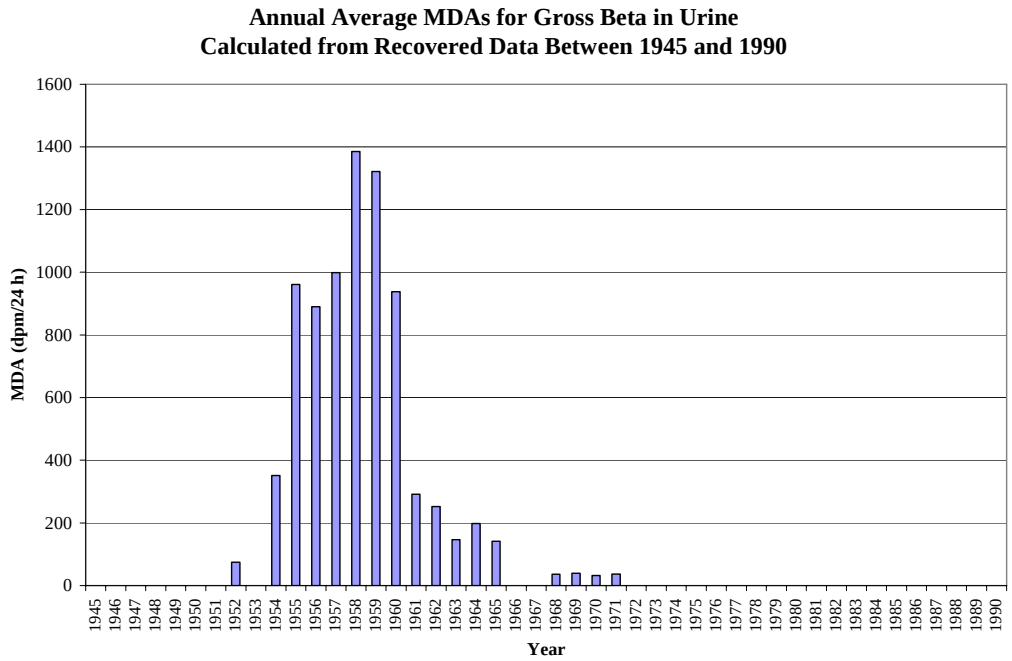


Figure A-5. MDA data for gross beta in urine (part 2 of 3).

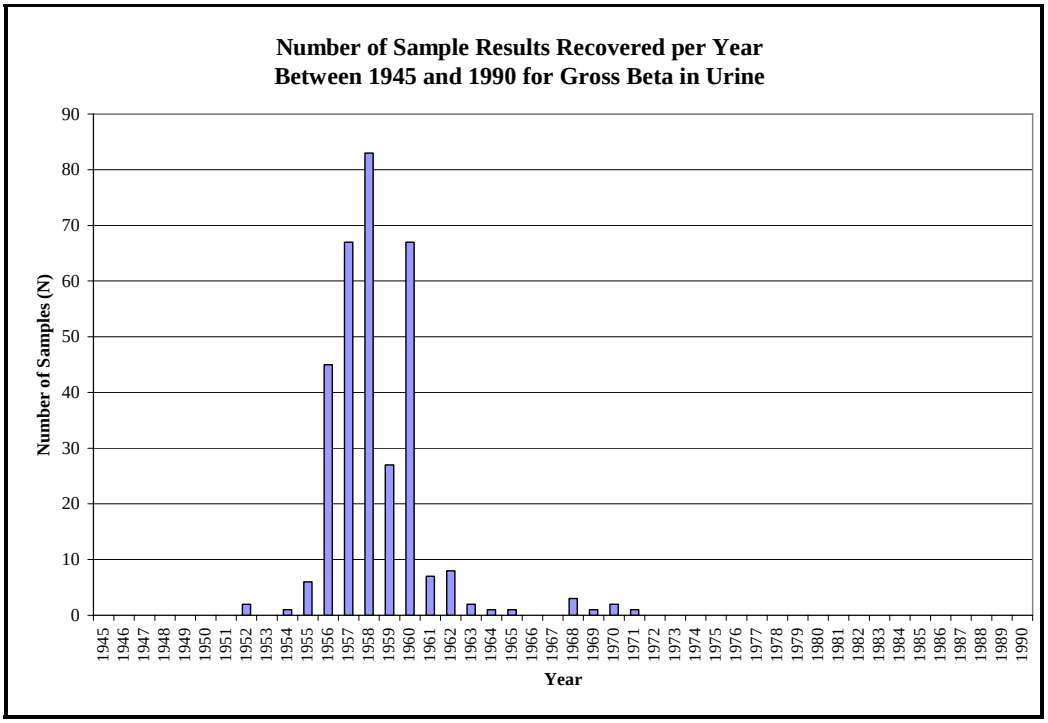


Figure A-5. MDA data for gross beta in urine (part 3 of 3).

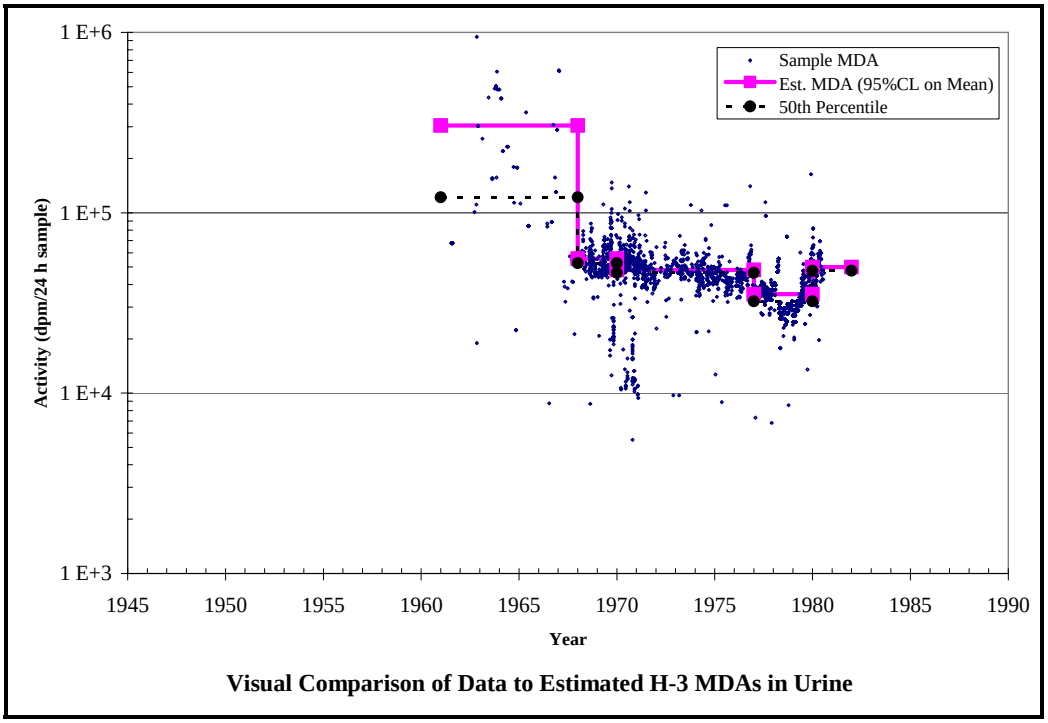


Figure A-6. MDA data for H-3 in urine (part 1 of 3).

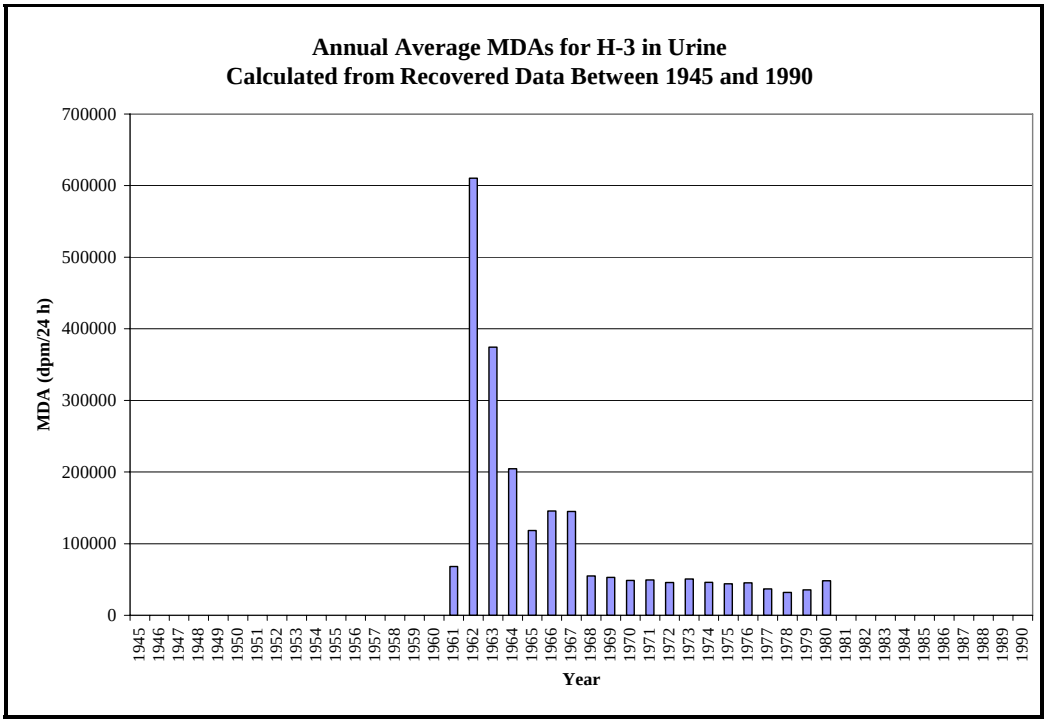


Figure A-6. MDA data for H-3 in urine (part 2 of 3).

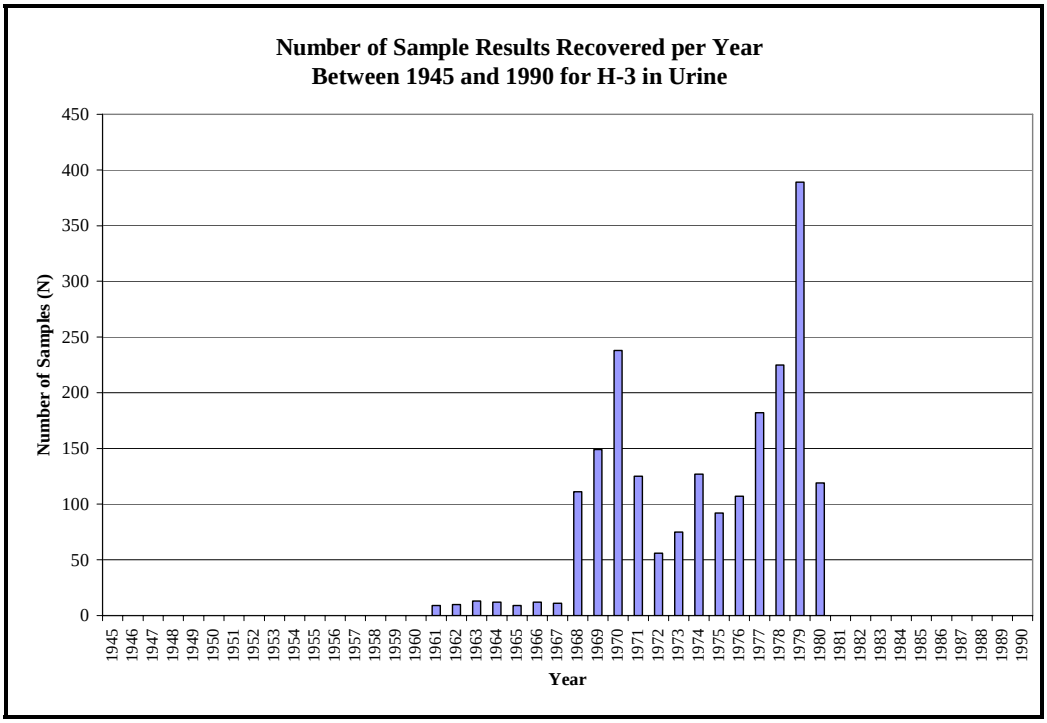
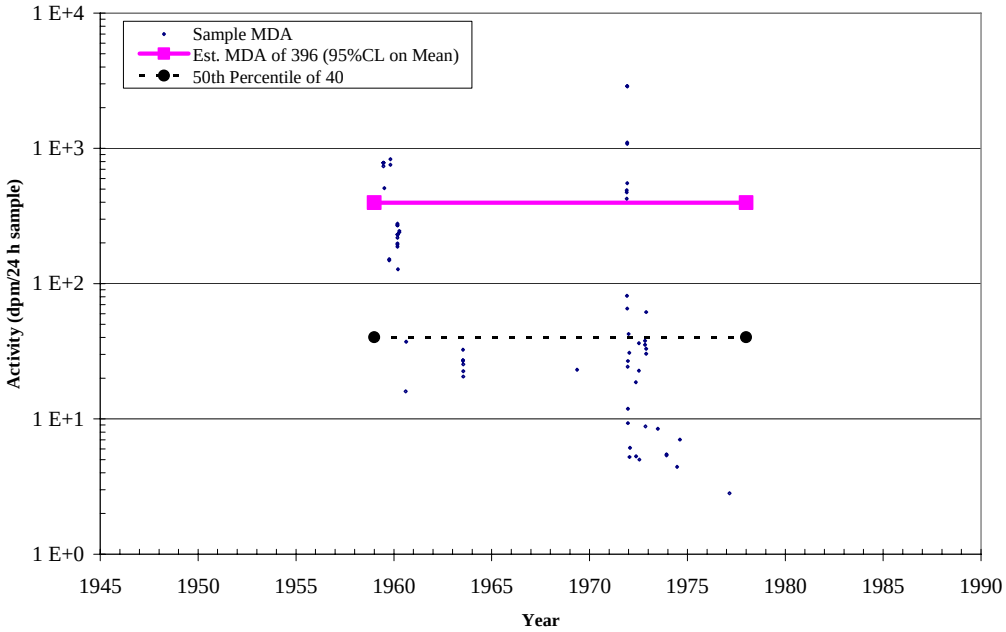


Figure A-6. MDA data for H-3 in urine (part 3 of 3).



Visual Comparison of Data to Estimated Pm-147 MDAs in Urine

Figure A-7. MDA data for Pm-147 in urine (part 1 of 3).

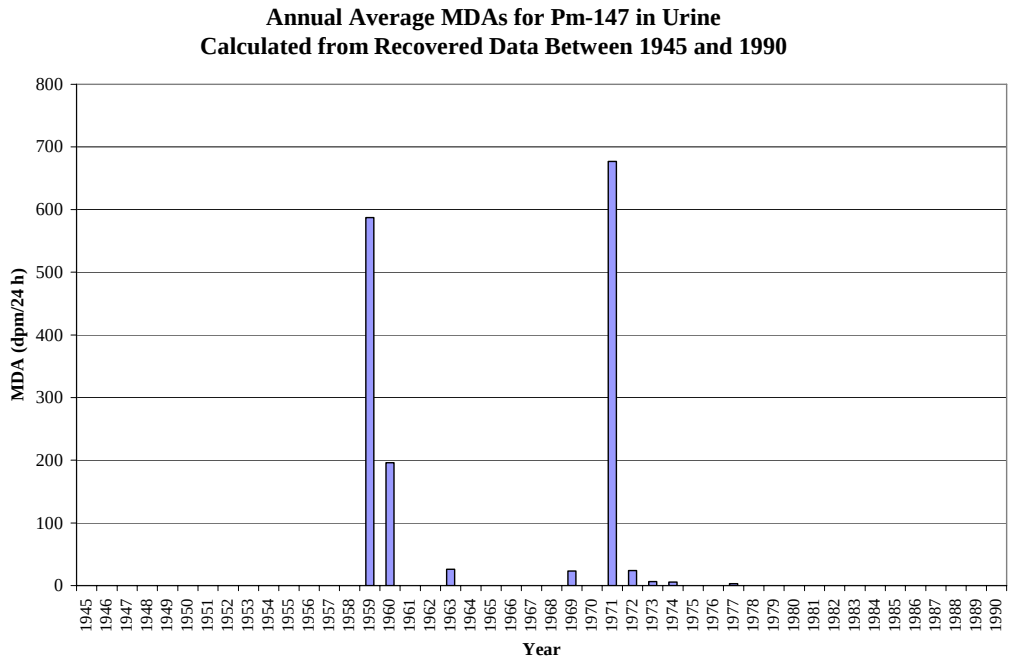


Figure A-7. MDA data for Pm-147 in urine (part 2 of 3).

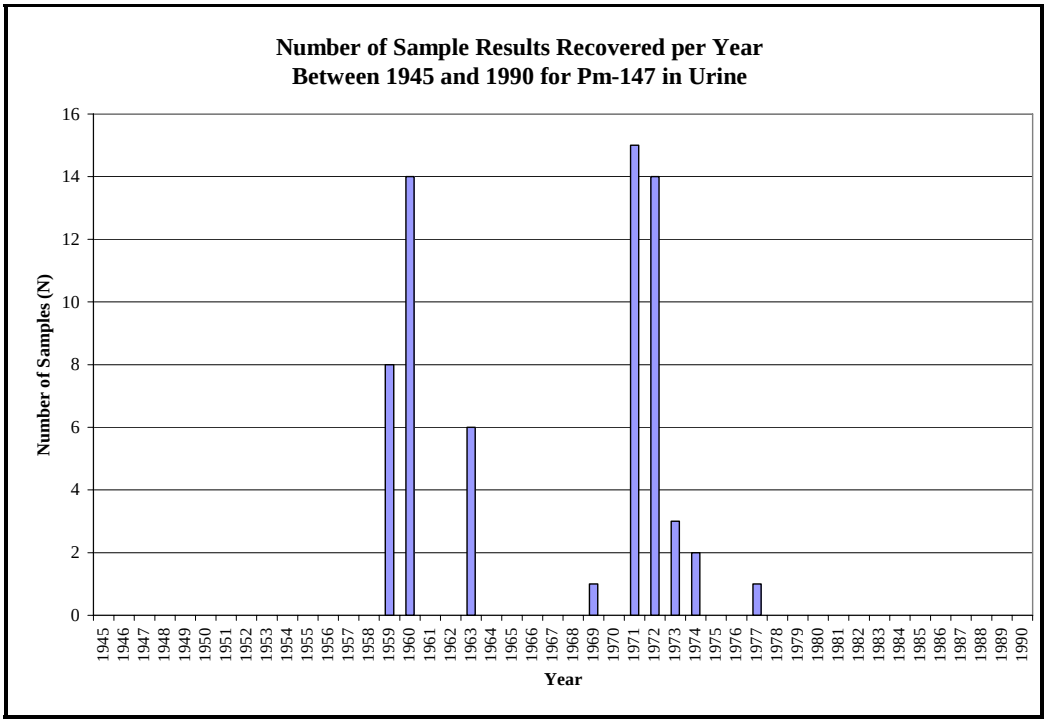


Figure A-7. MDA data for Pm-147 in urine (part 3 of 3).

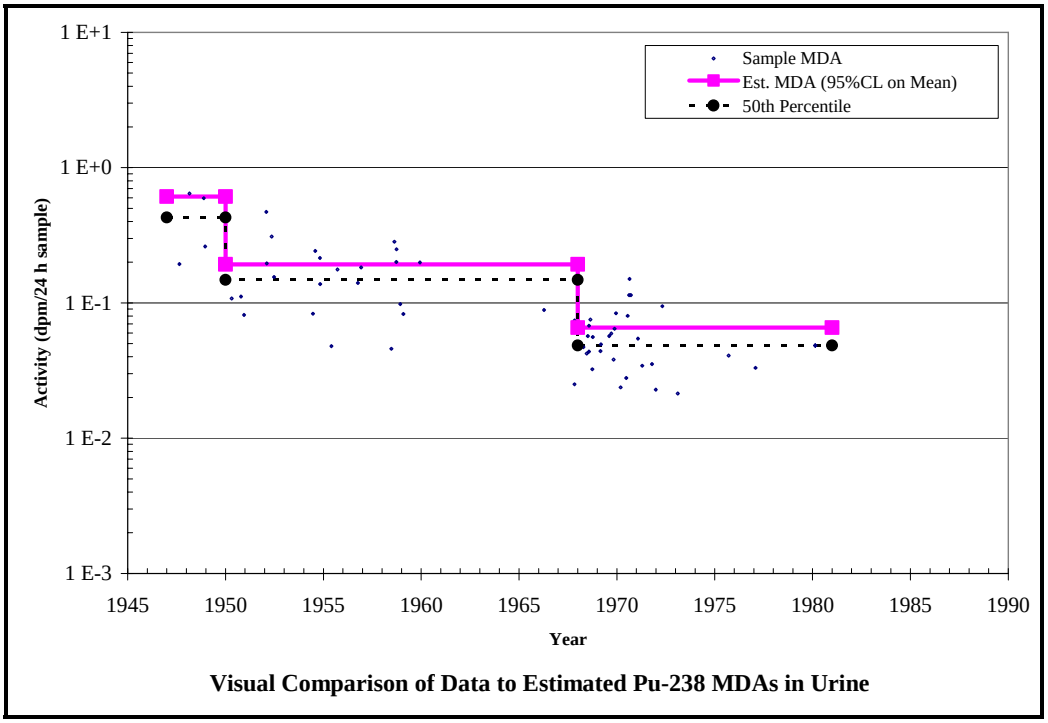


Figure A-8. MDA data for Pu-238 in urine (part 1 of 3).

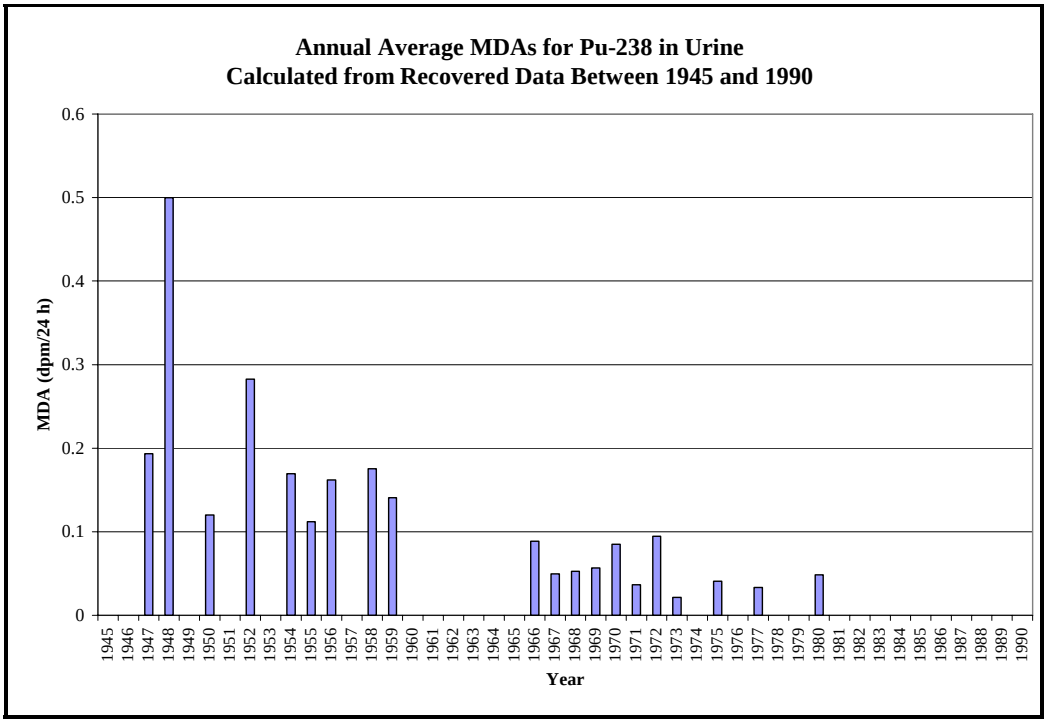


Figure A-8. MDA data for Pu-238 in urine (part 2 of 3).

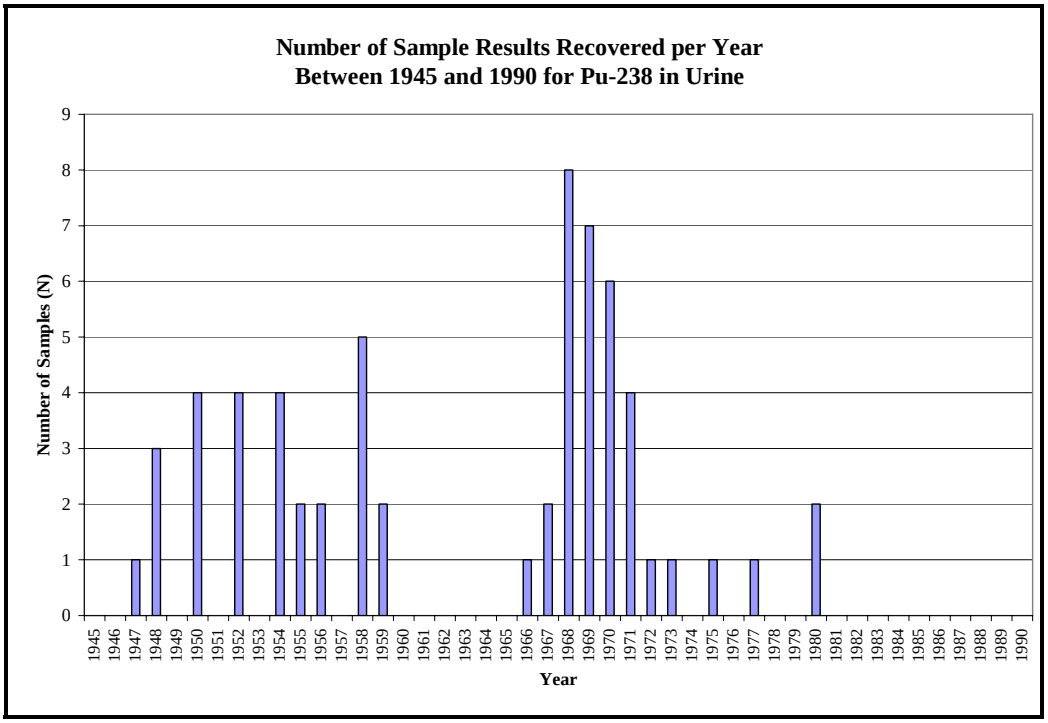


Figure A-8. MDA data for Pu-238 in urine (part 3 of 3).

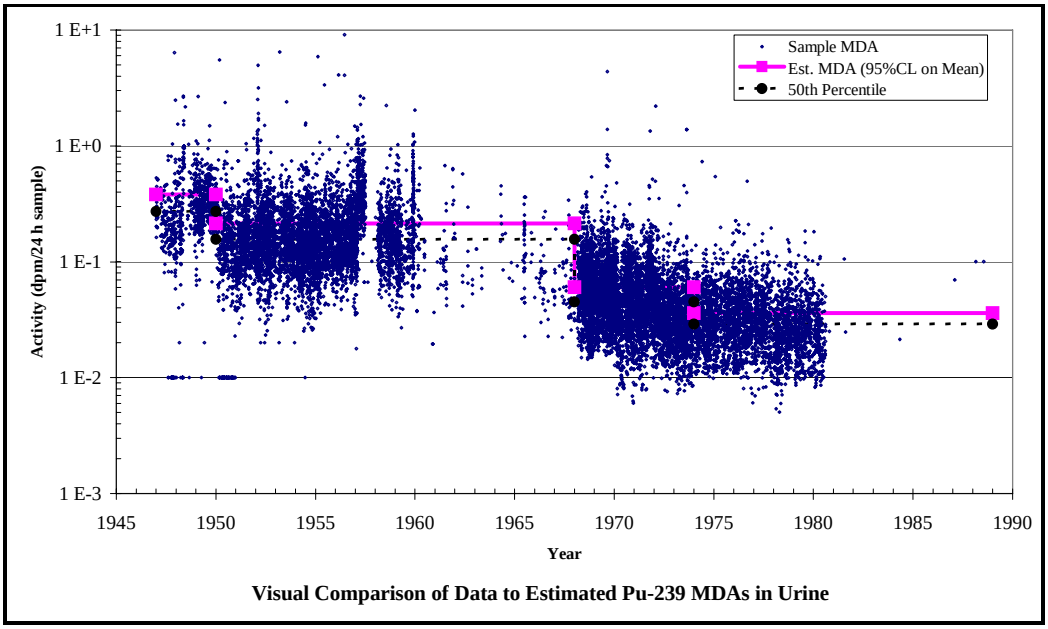


Figure A-9. MDA data for Pu-239 in urine (part 1 of 3).

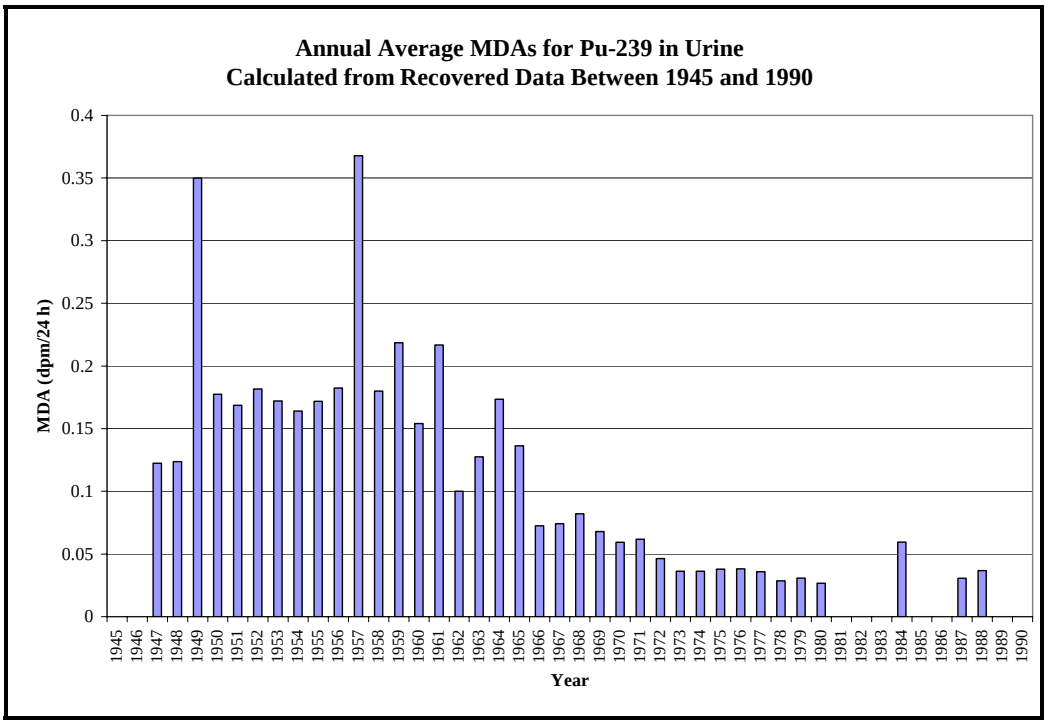


Figure A-9. MDA data for Pu-239 in urine (part 2 of 3).

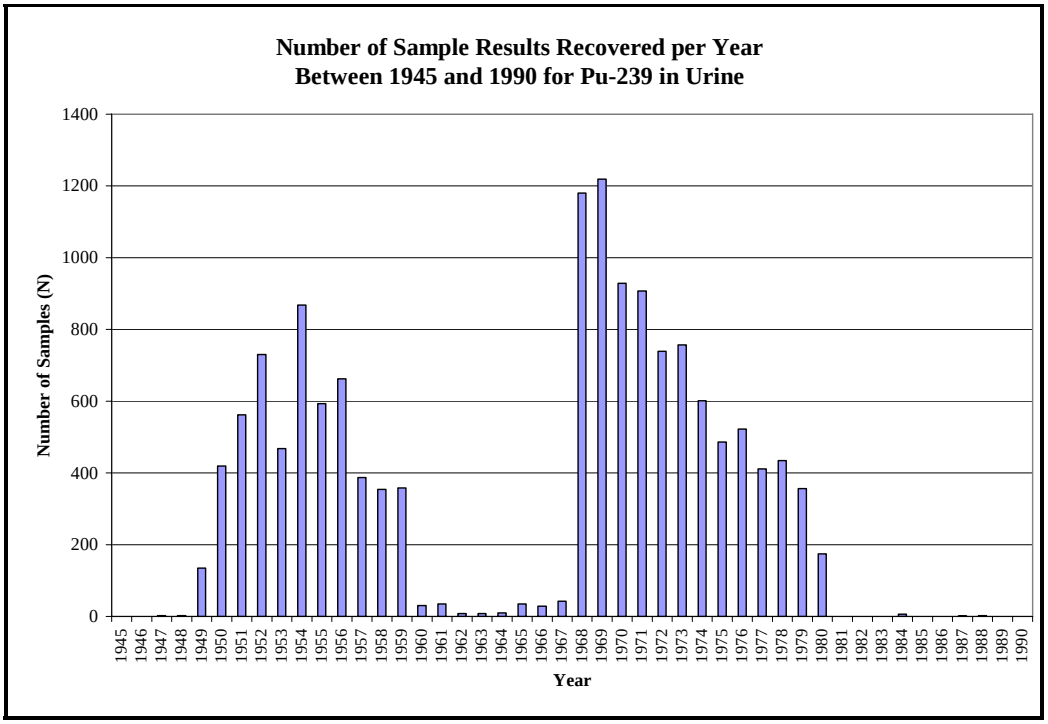


Figure A-9. MDA data for Pu-239 in urine (part 3 of 3).

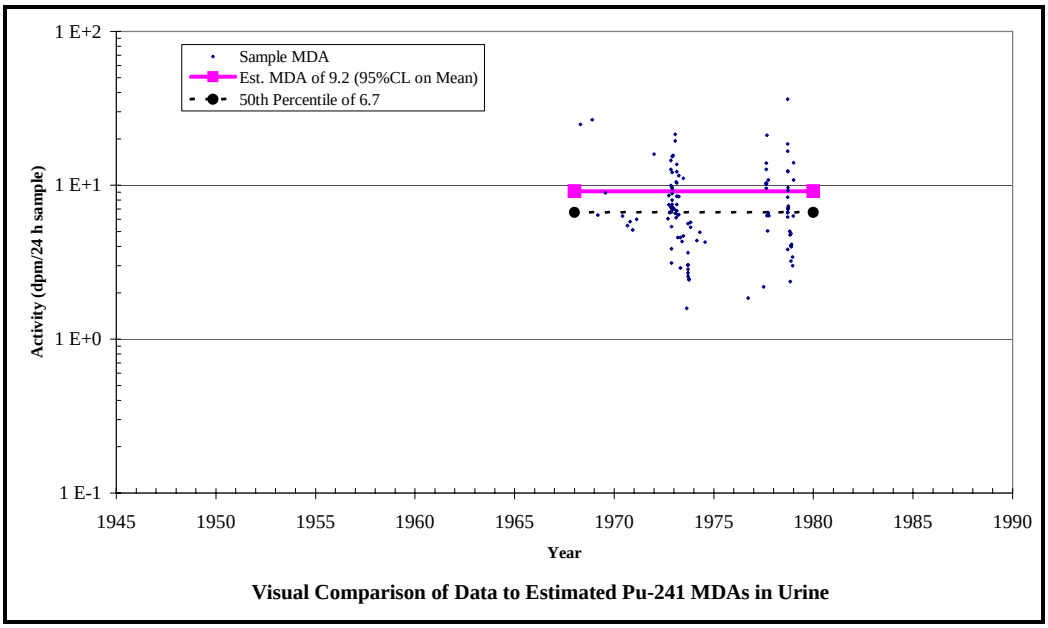


Figure A-10. MDA data for Pu-241 in urine (part 1 of 3).

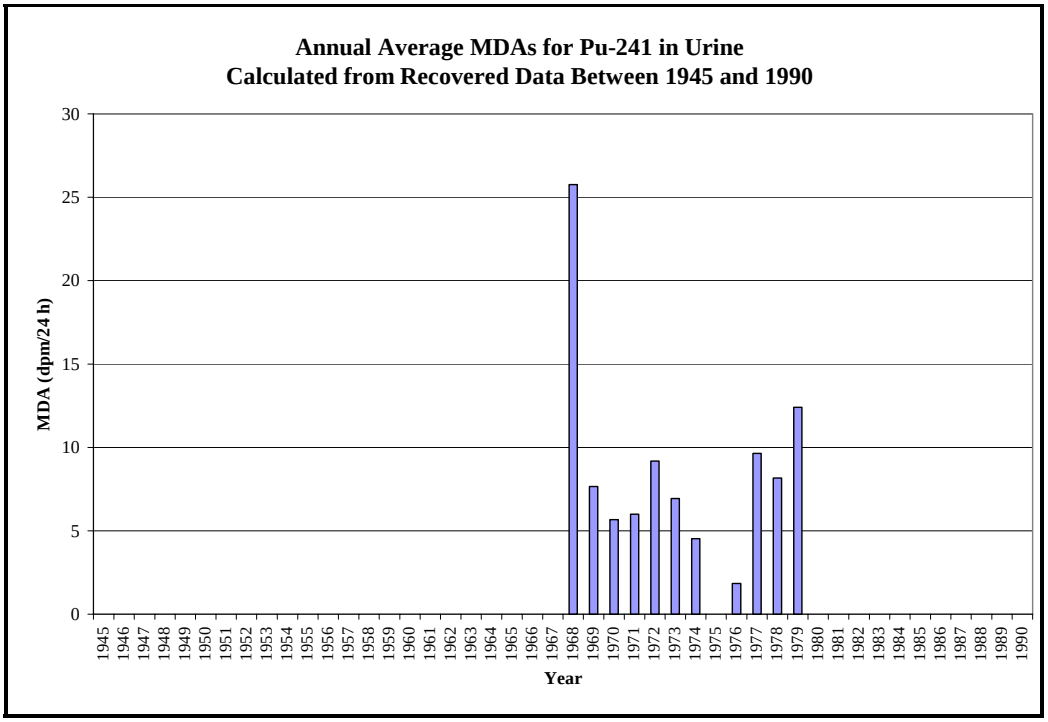


Figure A-10. MDA data for Pu-241 in urine (part 2 of 3).

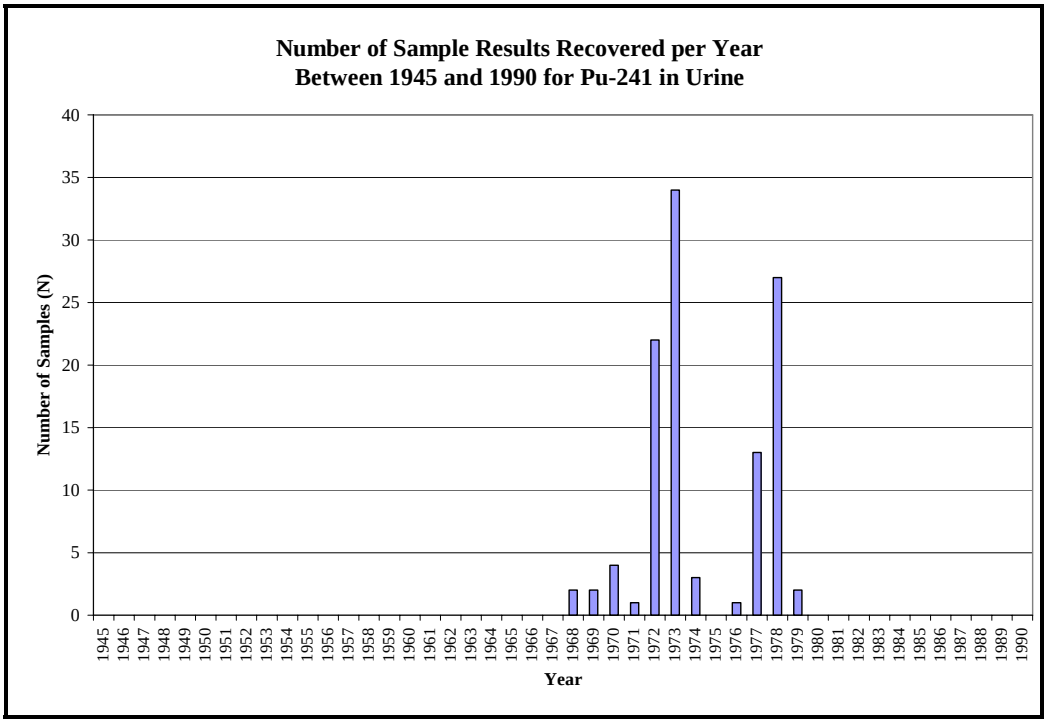


Figure A-10. MDA data for Pu-241 in urine (part 3 of 3).

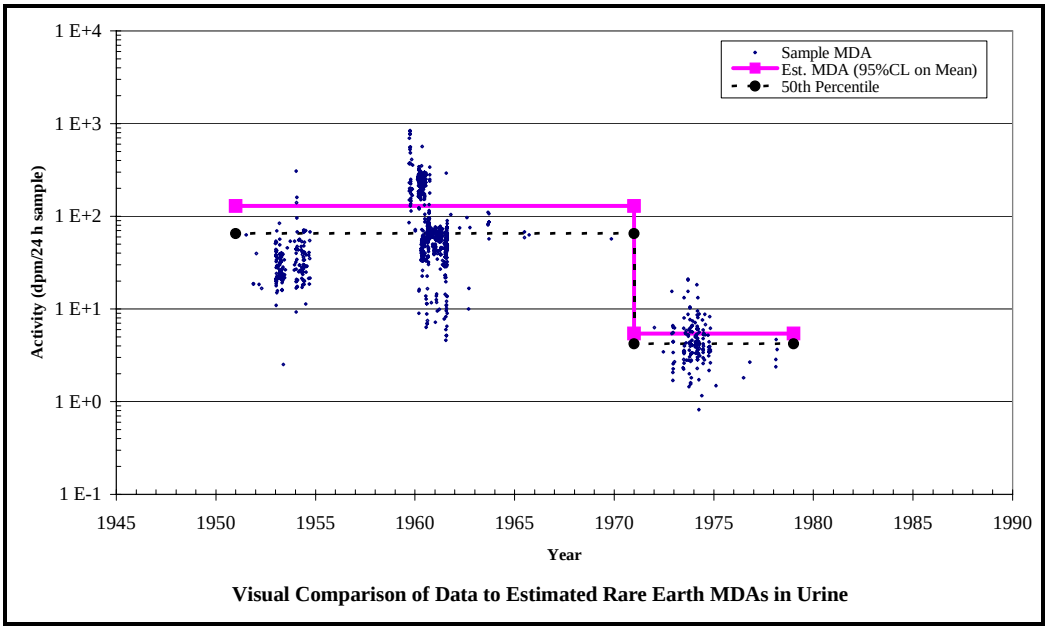


Figure A-11. MDA data for rare earths in urine (part 1 of 3).

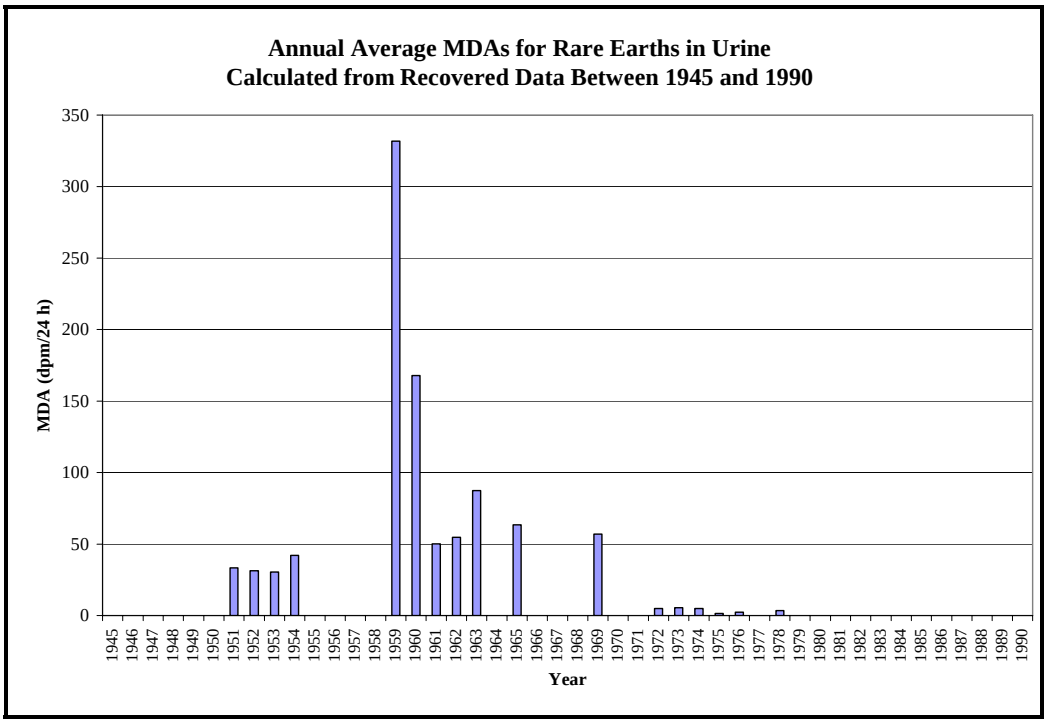


Figure A-11. MDA data for rare earths in urine (part 2 of 3).

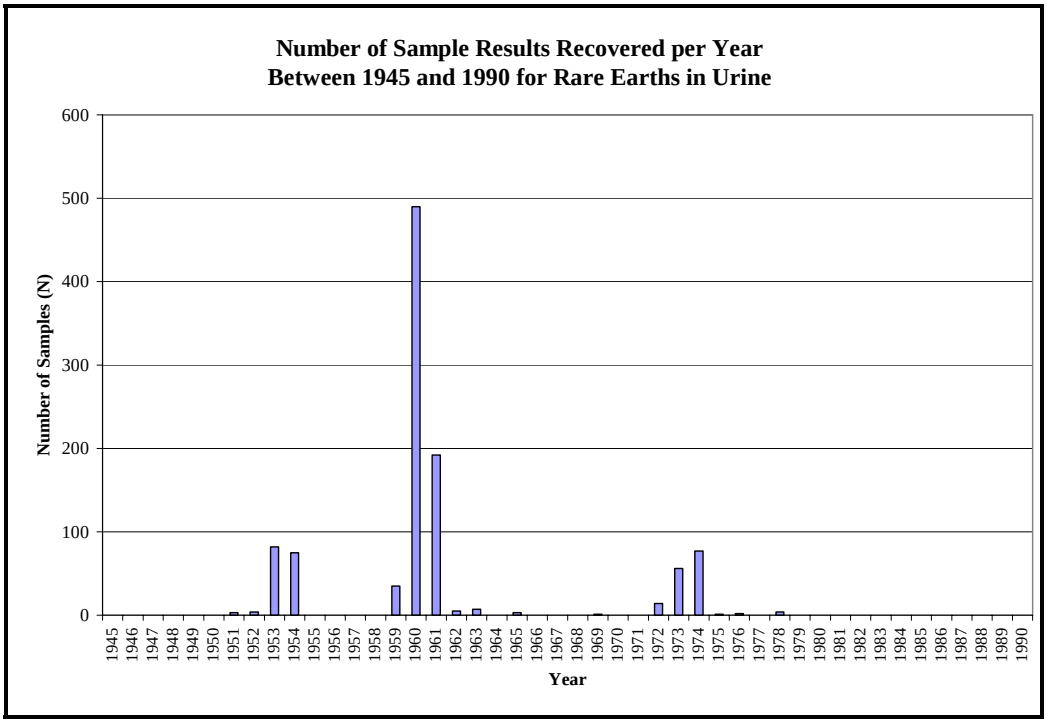


Figure A-11. MDA data for rare earths in urine (part 3 of 3).

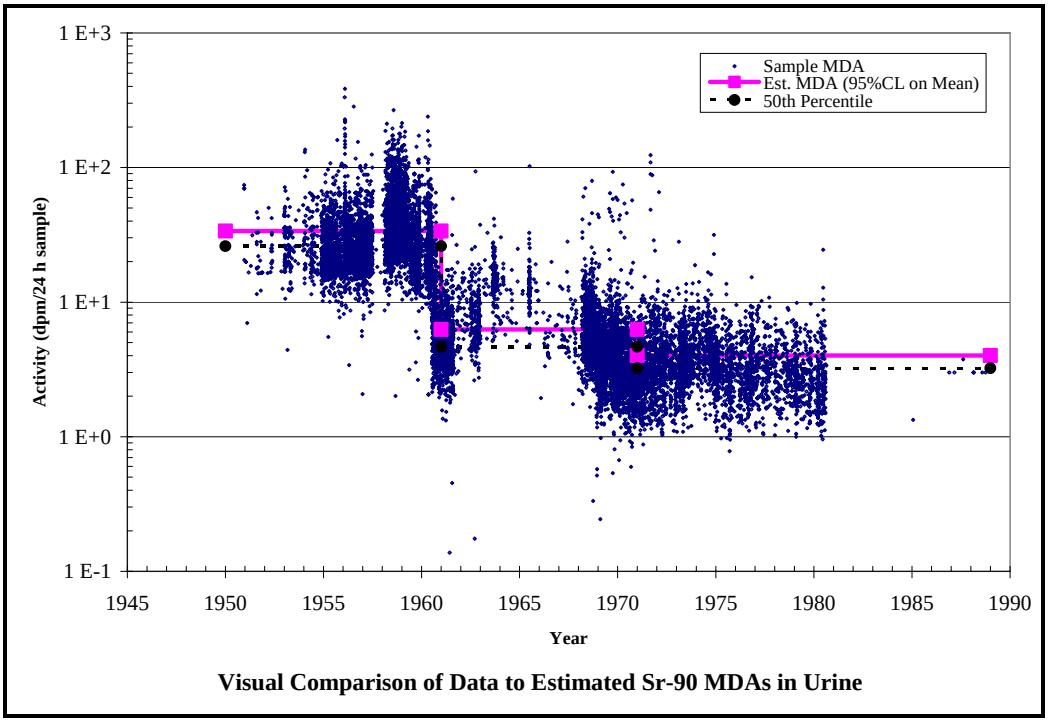


Figure A-12. MDA data for Sr-90 in urine (part 1 of 3).

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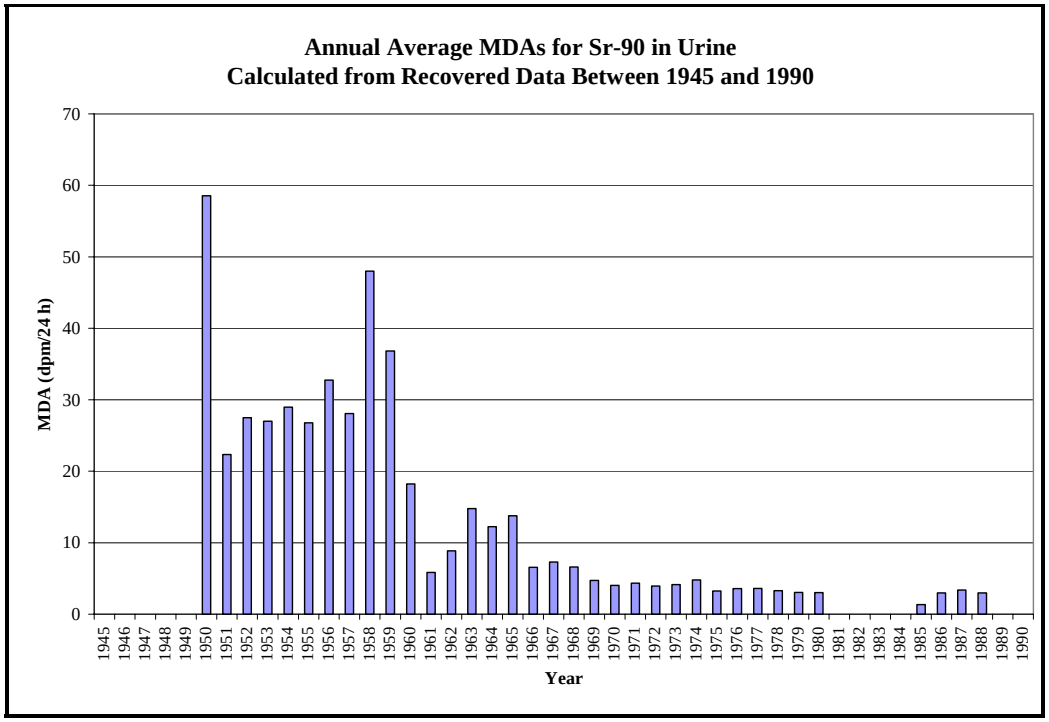


Figure A-12. MDA data for Sr-90 in urine (part 2 of 3).

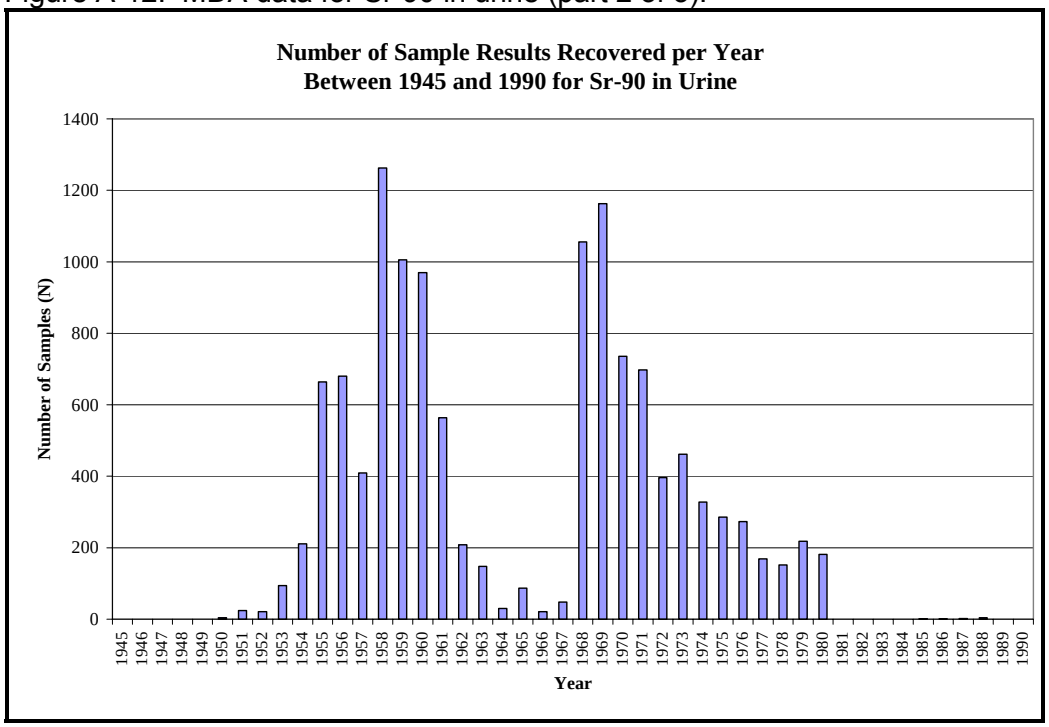
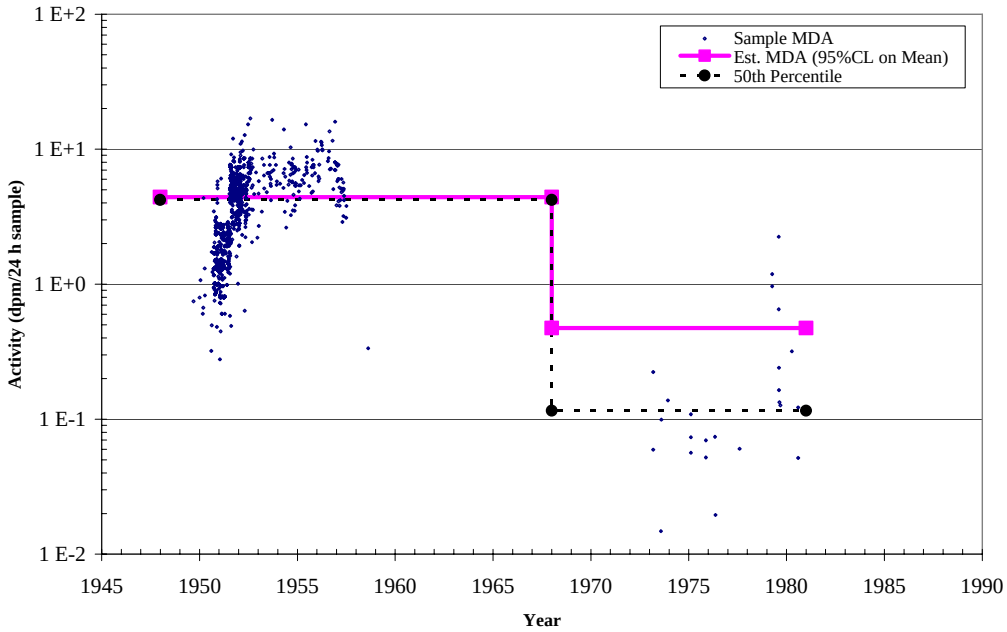


Figure A-12. MDA data for Sr-90 in urine (part 3 of 3).



Visual Comparison of Data to Estimated U-233 MDAs in Urine

Figure A-13. MDA data for U-233 in urine (part 1 of 3).

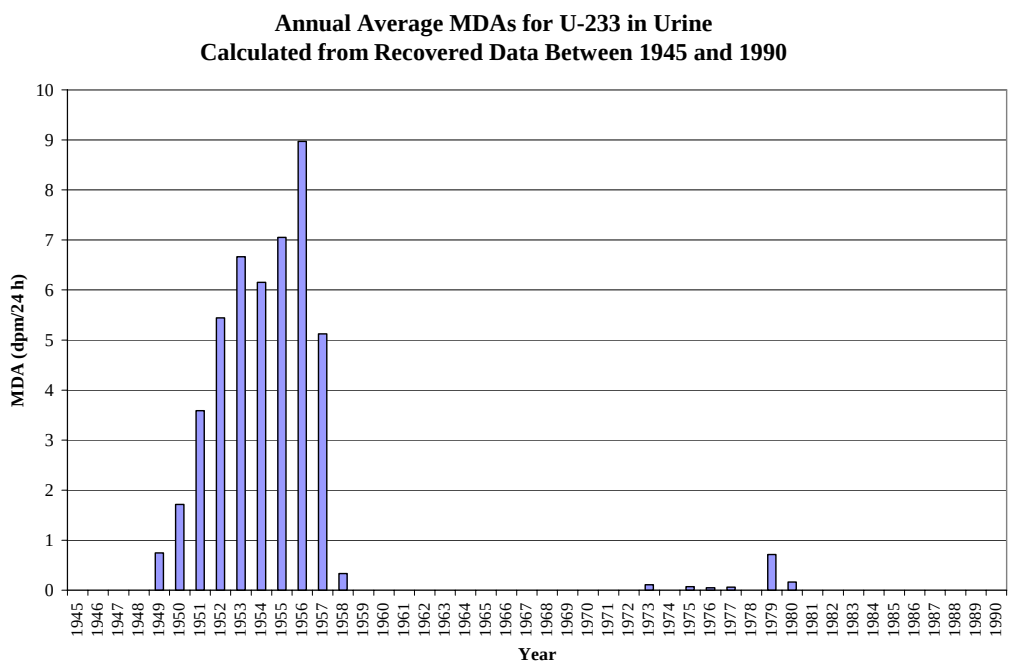


Figure A-13. MDA data for U-233 in urine (part 2 of 3).

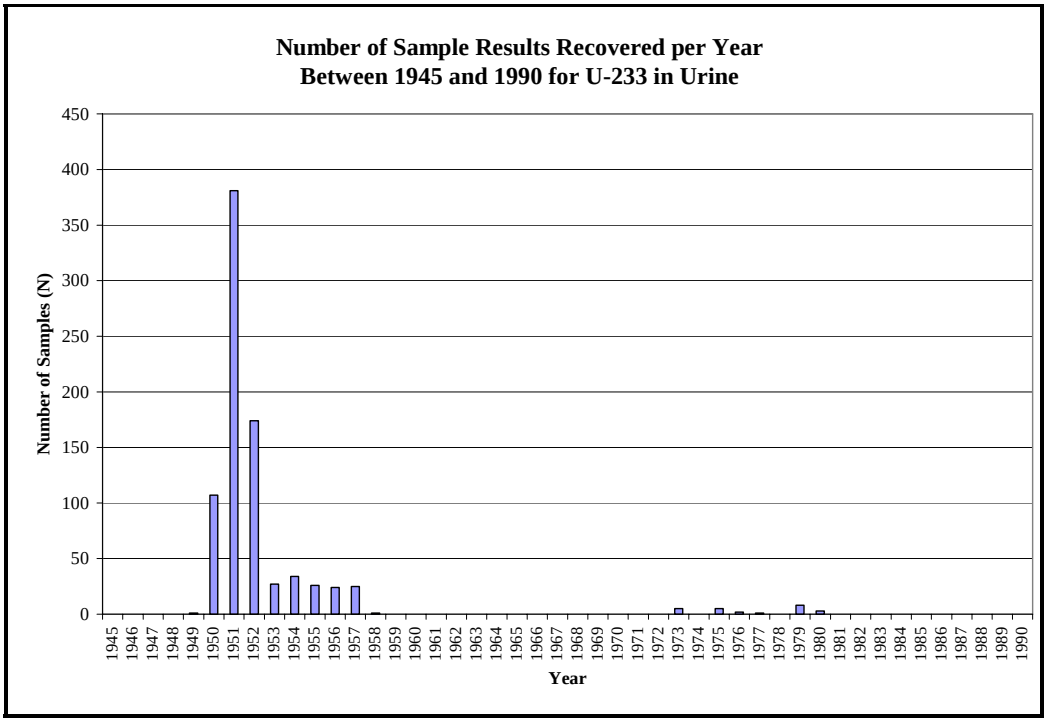


Figure A-13. MDA data for U-233 in urine (part 3 of 3).

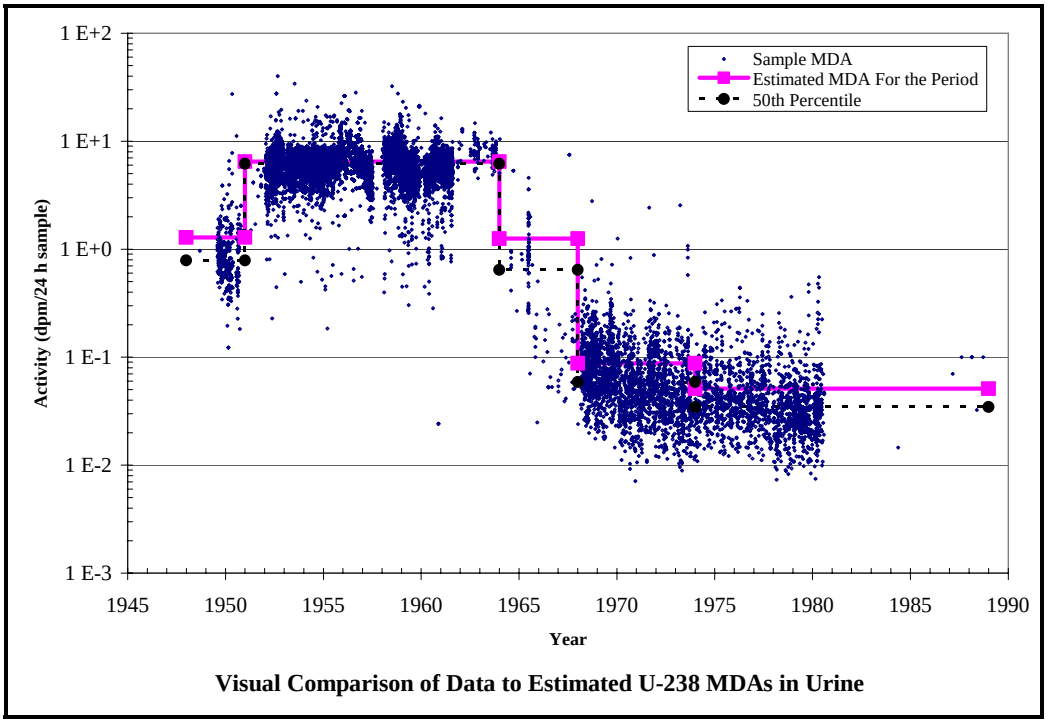


Figure A-14. MDA data for U-238 in urine (part 1 of 3).

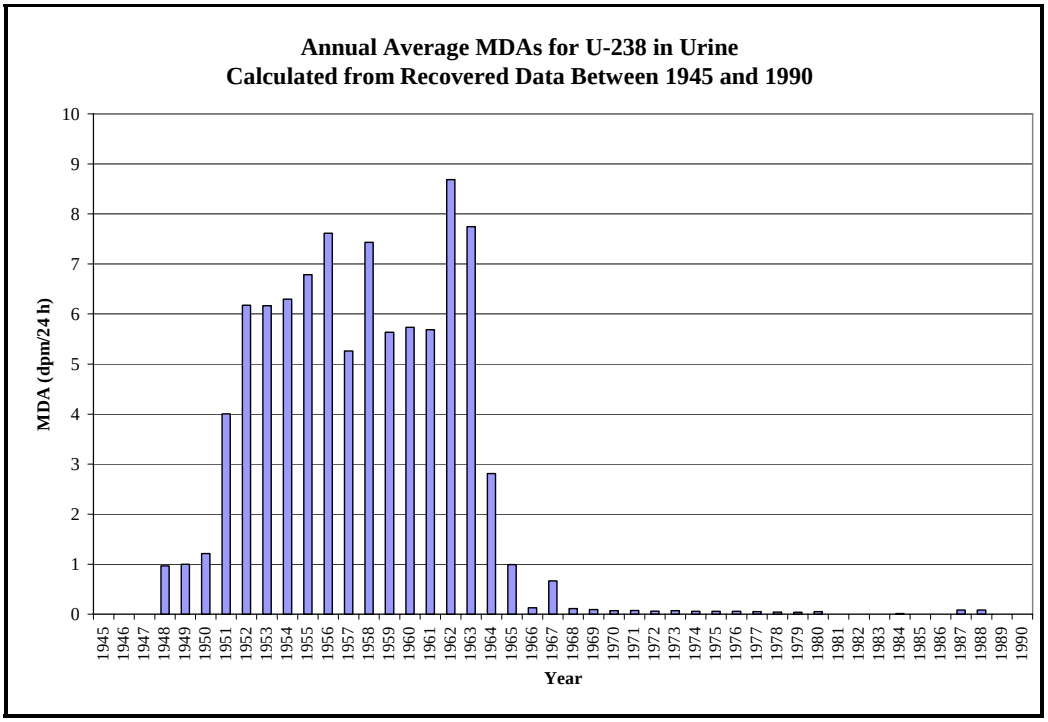


Figure A-14. MDA data for U-238 in urine (part 2 of 3).

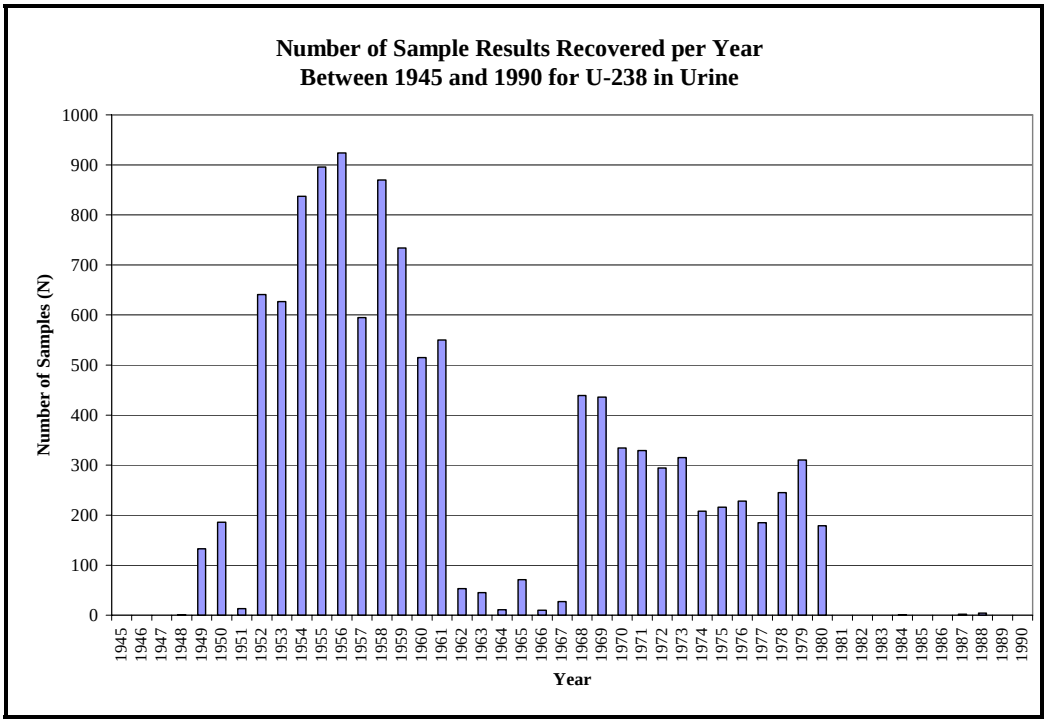


Figure A-14. MDA data for U-238 in urine (part 3 of 3).

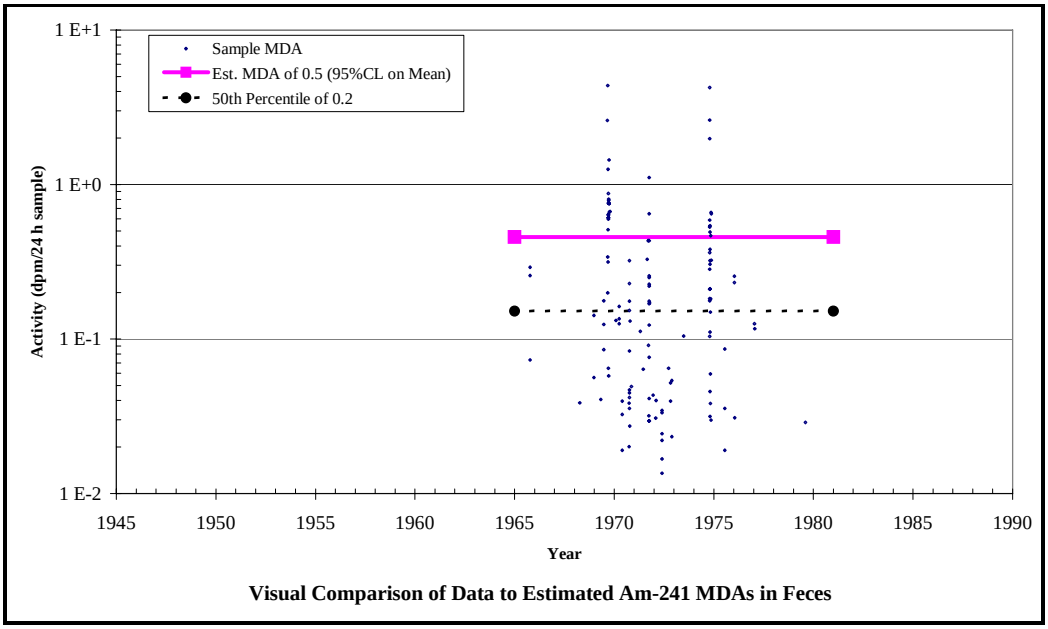


Figure A-15. MDA data for Am-241 in feces (part 1 of 3).

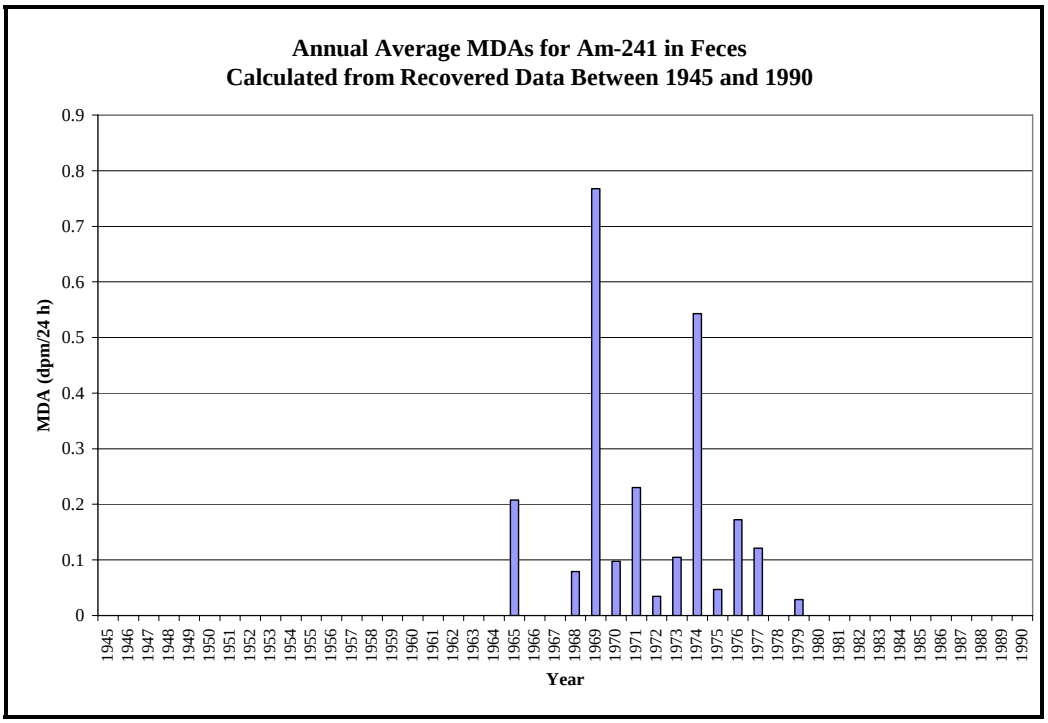


Figure A-15. MDA data for Am-241 in feces (part 2 of 3).

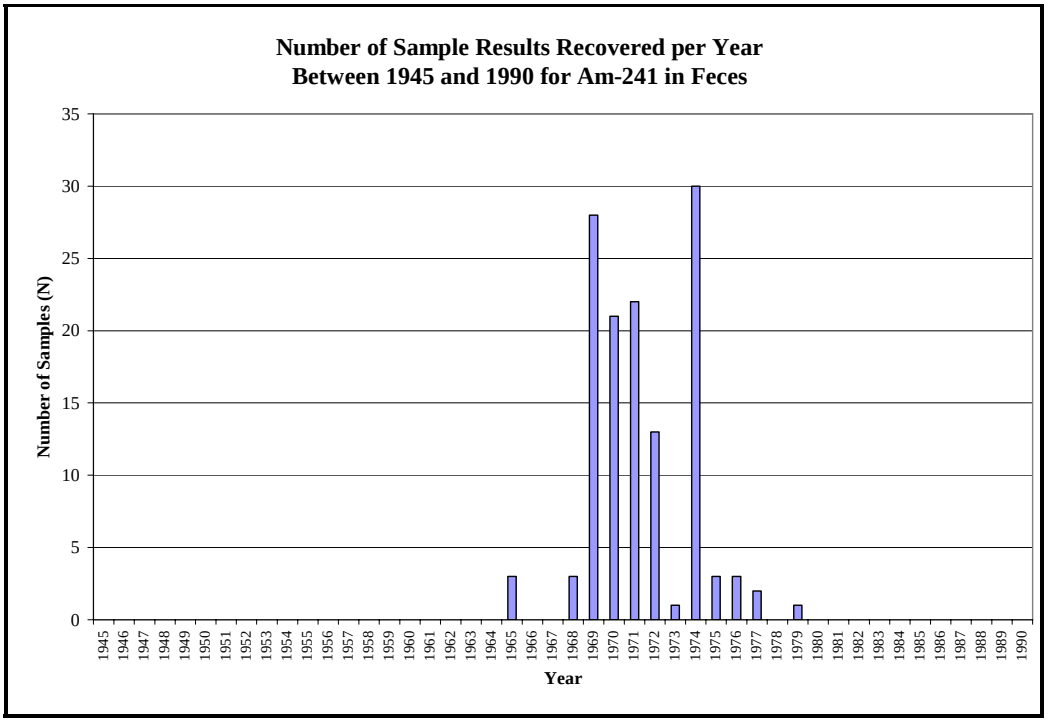


Figure A-15. MDA data for Am-241 in feces (part 3 of 3).

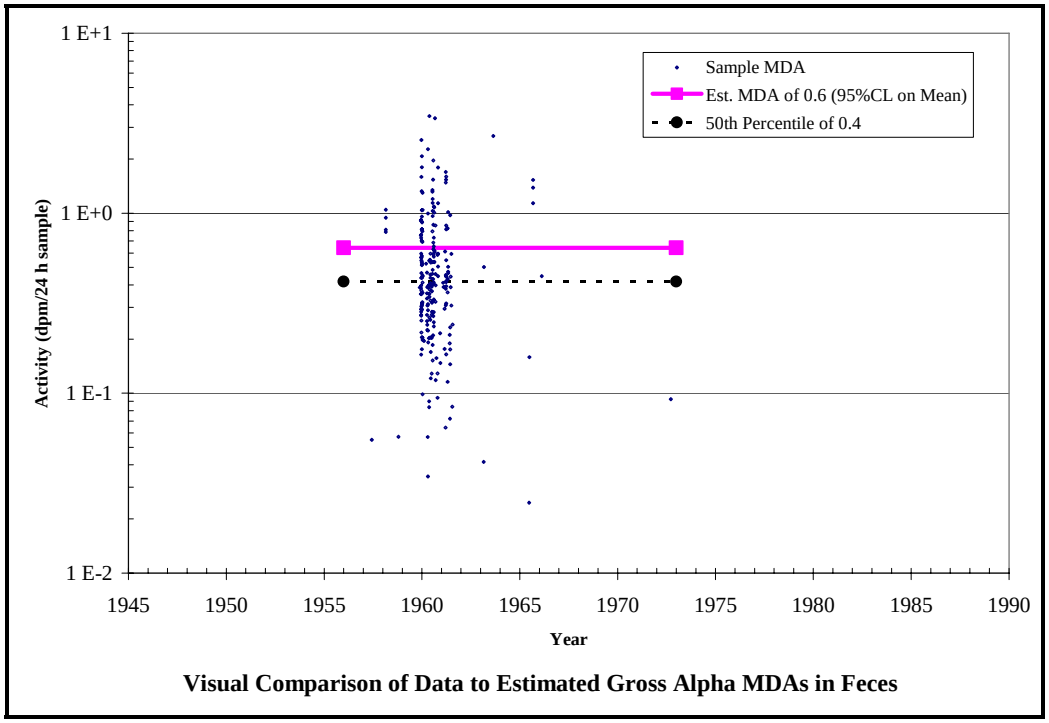


Figure A-16. MDA data for gross alpha in feces (part 1 of 3).

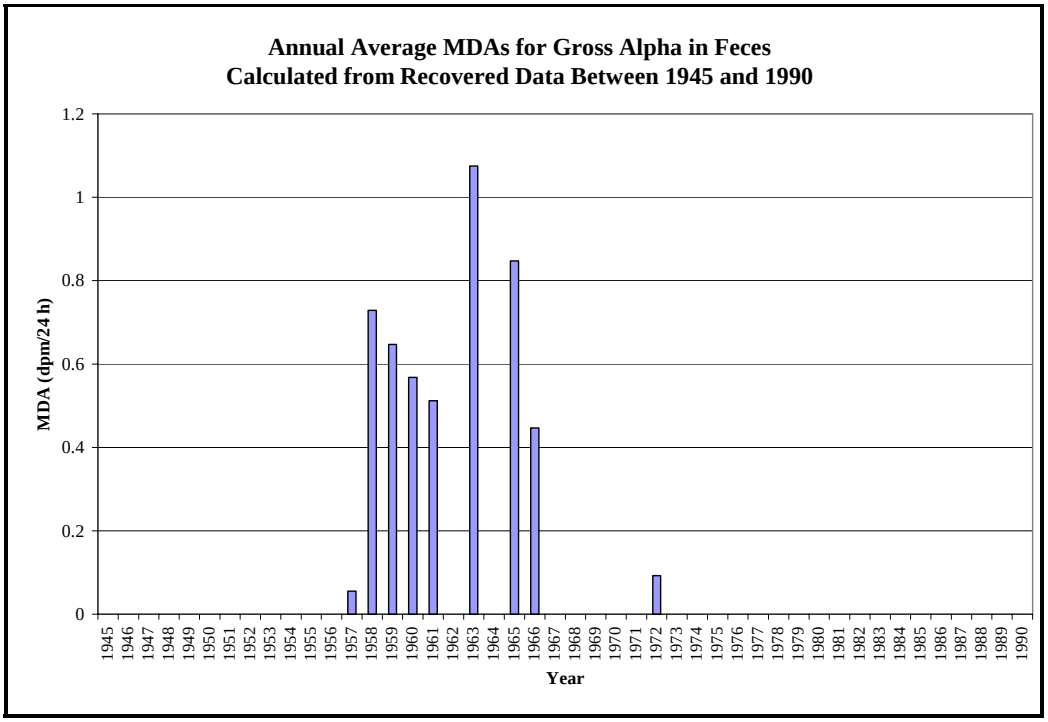


Figure A-16. MDA data for gross alpha in feces (part 2 of 3).

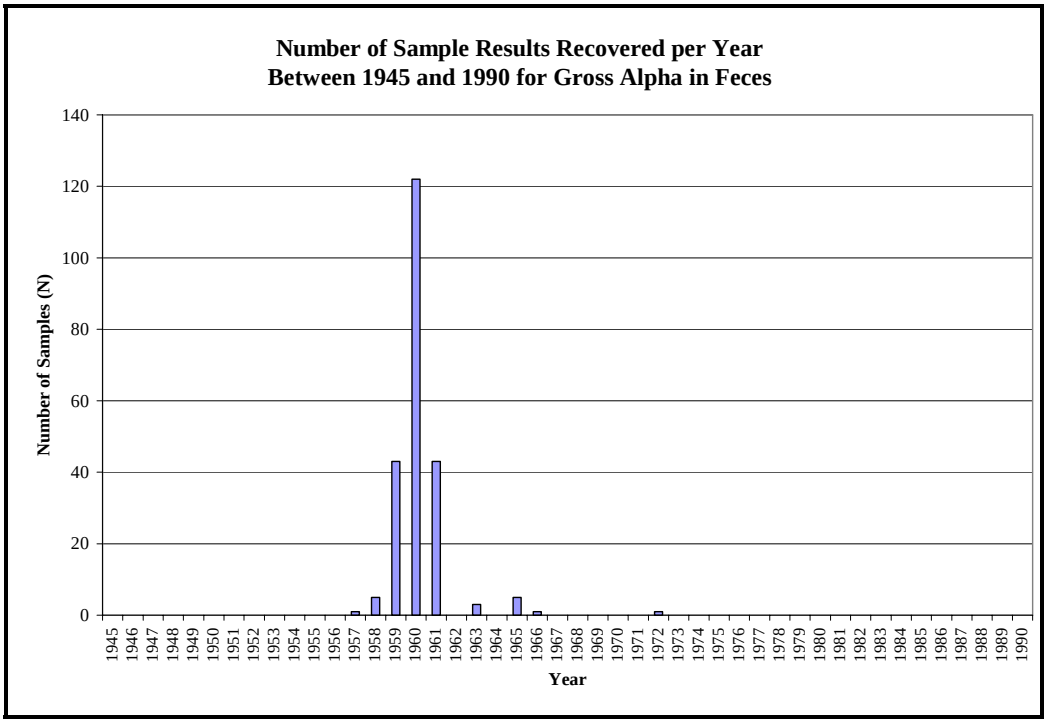


Figure A-16. MDA data for gross alpha in feces (part 3 of 3).

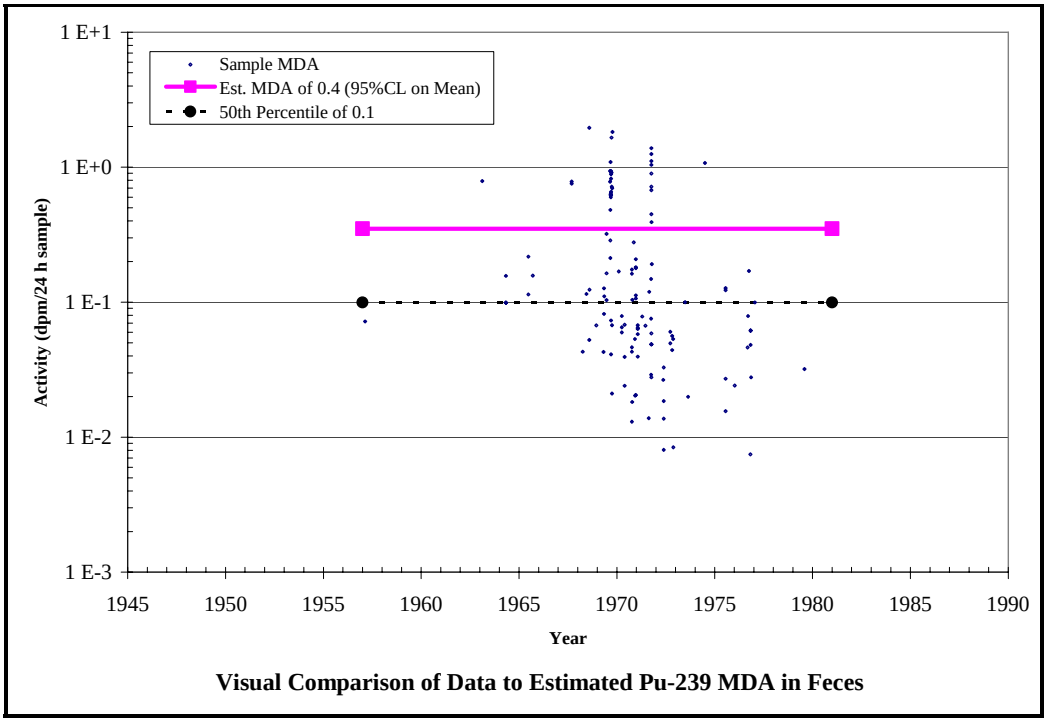


Figure A-17. MDA data for Pu-239 in feces (part 1 of 3).

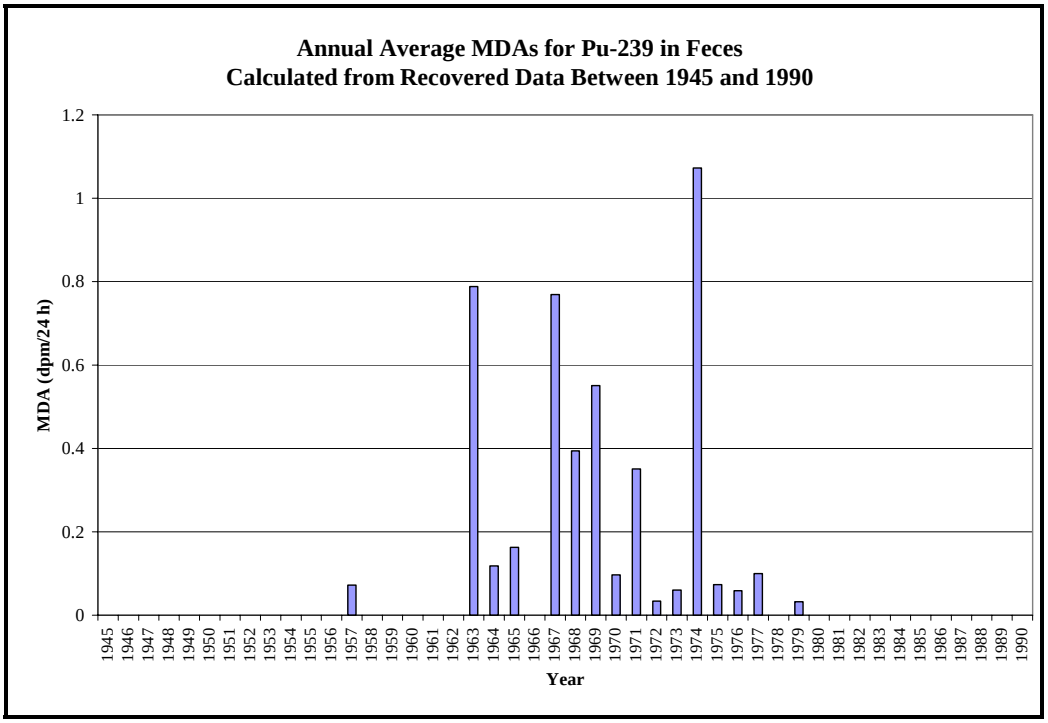


Figure A-17. MDA data for Pu-239 in feces (part 2 of 3).

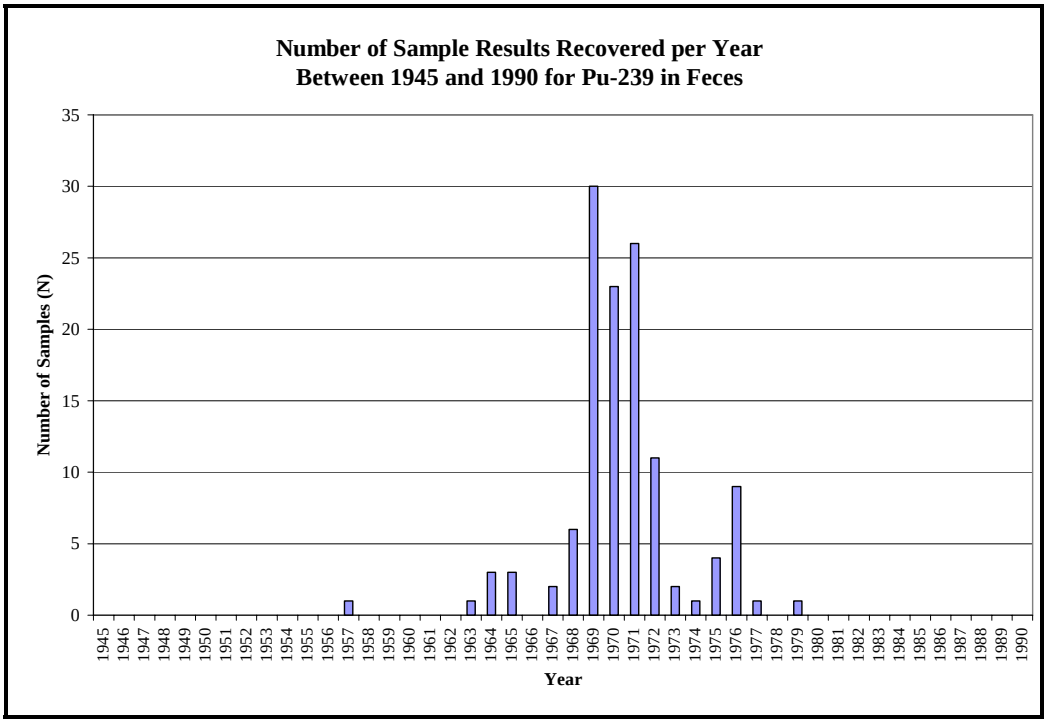


Figure A-17. MDA data for Pu-239 in feces (part 3 of 3).

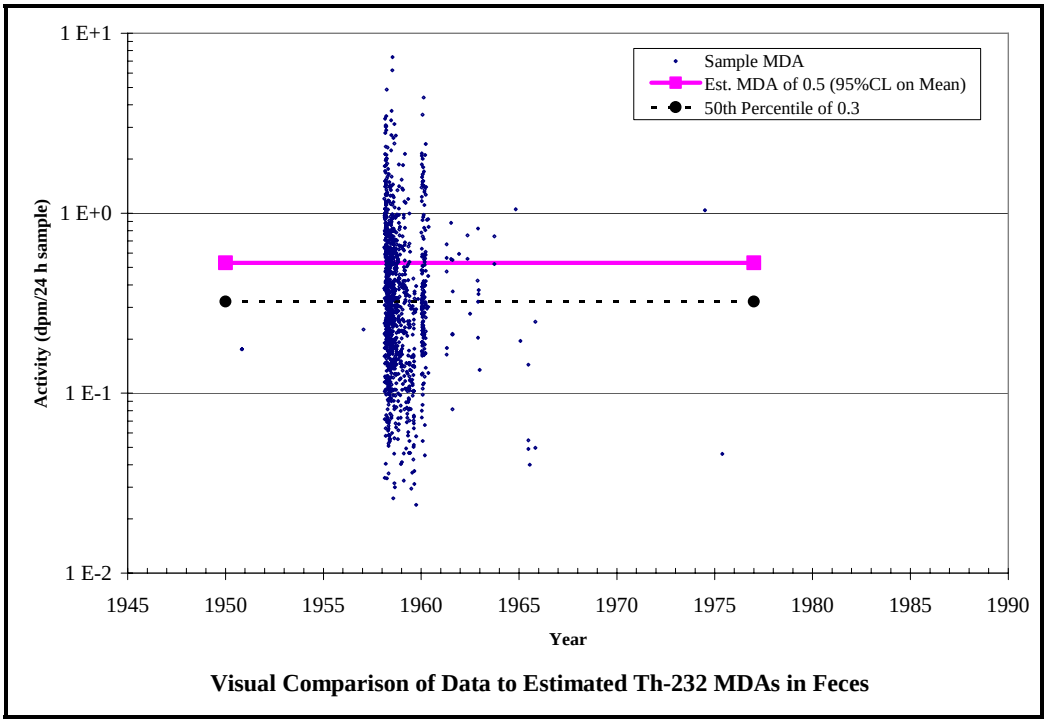


Figure A-18. MDA data for Th-232 in feces (part 1 of 3).

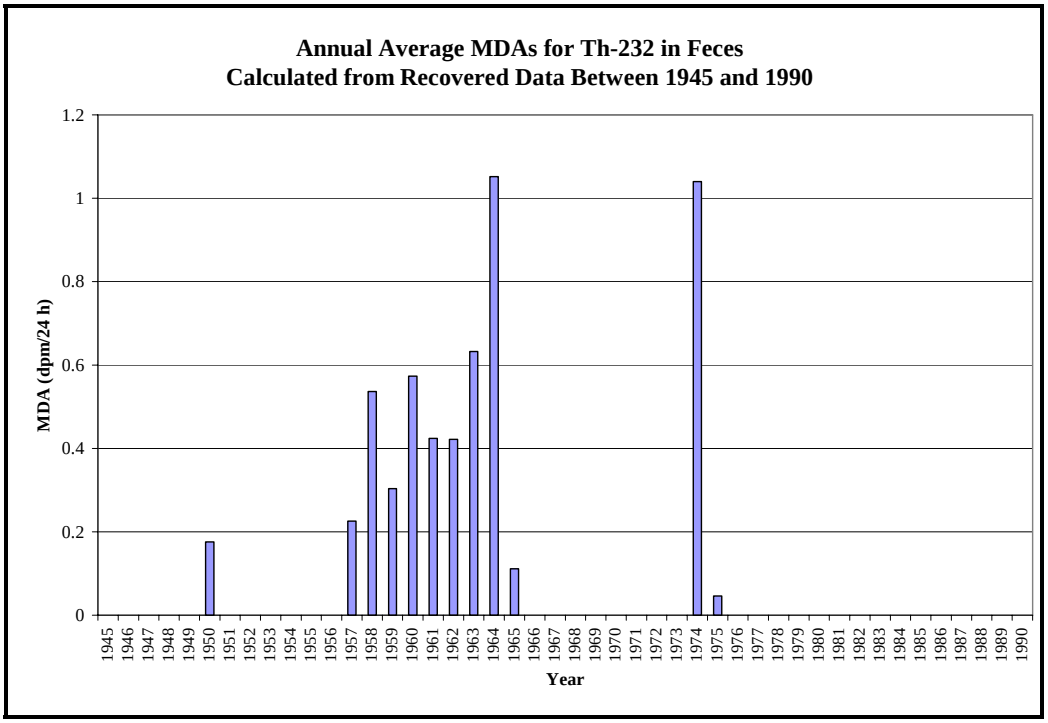


Figure A-18. MDA data for Th-232 in feces (part 2 of 3).

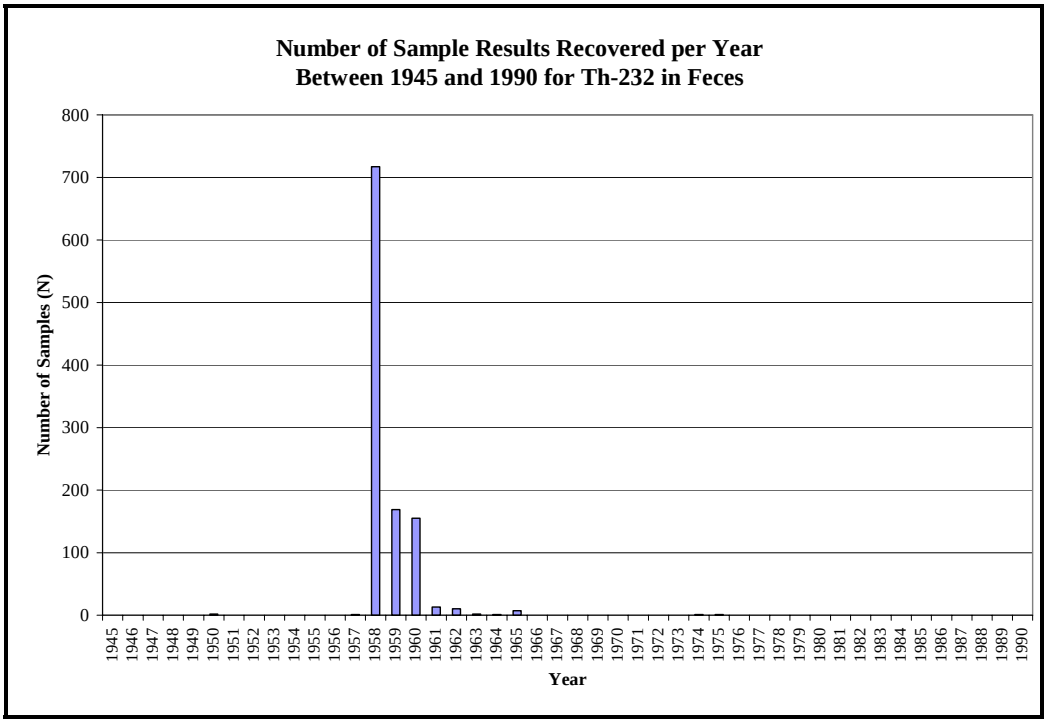


Figure A-18. MDA data for Th-232 in feces (part 3 of 3).

ATTACHMENT B
ASSESSMENT OF INTERNAL DOSE AT ORNL: 1944 TO 1947

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Information about internal dose at ORNL from 1943 to 1947 is somewhat limited, with no bioassay results appearing in the DOE worker files. ORAUT (2005a) currently extends back to 1944 for use as an overestimate at the site because tolerance levels are well documented and there is strong evidence (memoranda and monthly reports) that exposure was controlled based on the results of the samples. These memoranda include weekly reports on the airborne activity levels in various areas, as well as precautions taken or to be taken in the future. Examples of these reports include Bloss (1944a) to Bloss (1944g).

ORAUT (2005a) generally provides an overestimate for a number of reasons. Exposure at the maximum permissible air concentration is assumed for 40 hr per week, 50 weeks per year. While exposure to concentrations greater than this is likely over short periods for some occupations, the overriding control factor was to limit the average exposure over longer timeframes. In addition, the nuclide that results in the largest dose is assumed to comprise 100% of the intake; the list of possible nuclides includes some that are very unlikely to be present in significant quantities. During the early years of ORNL, several of the nuclides on the list were just being discovered or were yet to be discovered. The assumed nuclide might change from year to year to ensure that the largest possible dose is assigned in each year until the date of diagnosis.

Although not included in the individual files, there are limited urine sample results from ORNL in this period. About 150 results for urine plutonium content were found; all were submitted in 1945. It is postulated that these are not in the database because they precede the routine sampling program at the site; they were collected as part of a study to determine “the amount of plutonium, if any, that may be fixed in the bodies of any chemists at Clinton Laboratories” (Wirth 1945a). Urine samples were collected weekly for about 6 months at both Clinton (the original name for ORNL) and the Met Lab in Chicago. The entire data set was fitted to a lognormal distribution (Figure B-1).

The geometric mean of this distribution is somewhat large (1.7 pCi/1400 mL), and the fit to the results indicates that there are possibly two distributions. If this value is assumed to be due to a constant, chronic intake of Type M ^{239}Pu beginning in early 1944, an airborne concentration of 8.1×10^{-11} µCi /mL is indicated. This is almost three times larger than the tolerance level of 3×10^{-11} µCi /mL in effect at the time. If Type S is assumed, the airborne concentration is expected to be 2 orders of magnitude larger than the tolerance level.

In a memorandum dated May 11, 1945, it was reported that the findings appeared to be above control levels, which resulted in a more rigid technique of collection and system of collecting samples being implemented on April 25 (Wirth 1945b) . Results in the first 2 weeks of this new collection system were not substantially different from previous results, but samples in the subsequent, final 4 weeks of

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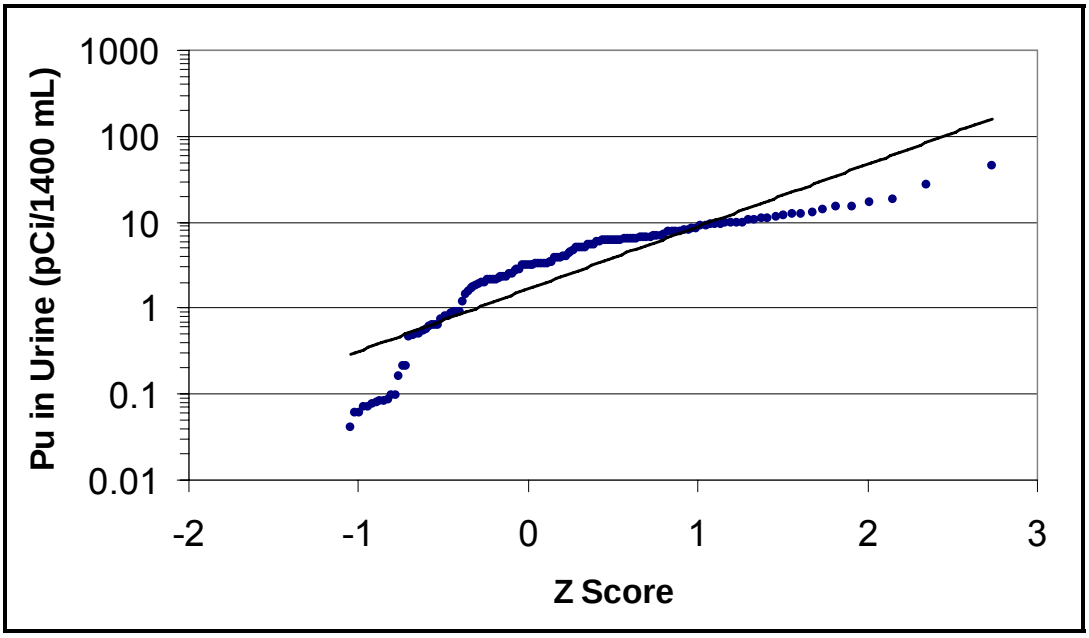

Figure B-1. Plutonium in urine samples from 1945. The geometric mean = 1.7 pCi/day, the GSD = 5.3, and r^2 = 0.79.

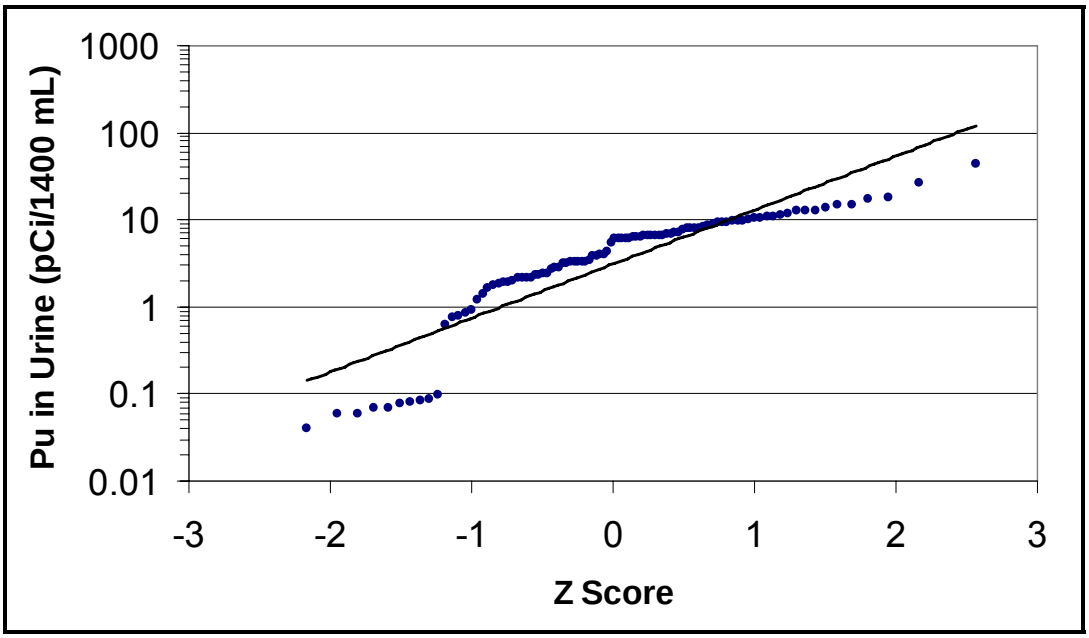

Figure B-2. Plutonium urine sample data through April 18, 1945. The geometric mean = 3.1 pCi/day, the GSD = 4.2, and r^2 = 0.82.

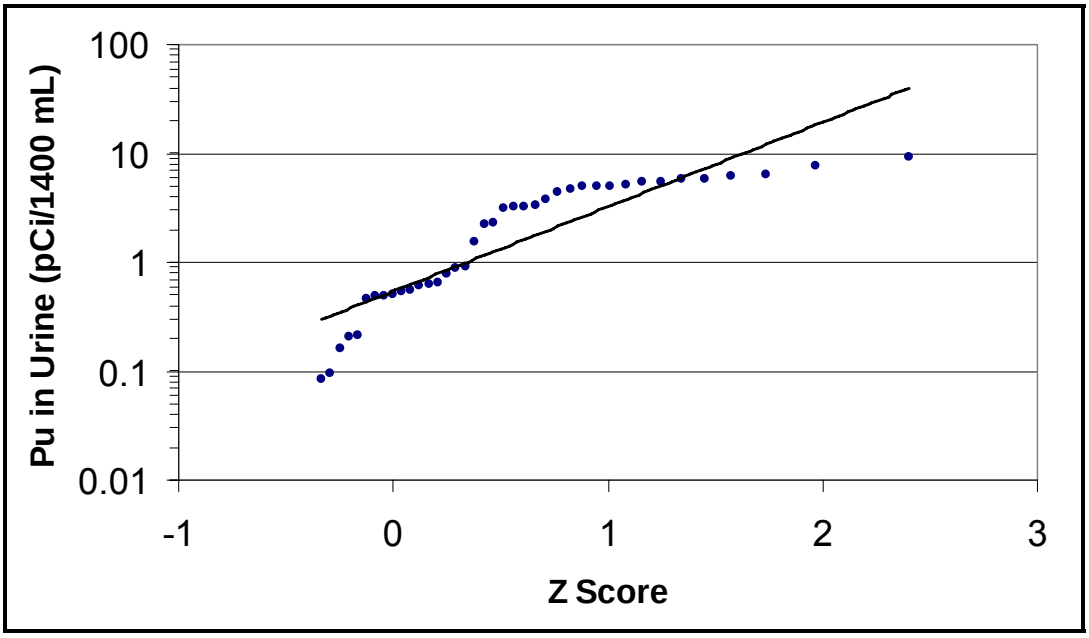

Figure B-3. Plutonium urine sample data for April 25-May 30, 1945. The geometric mean = 0.54 pCi/day, the GSD = 6.0, and r^2 = 0.80.

the study showed little detectable activity. Two separate lognormal fits were performed on these data, before and after the implementation of stricter controls. Figures B-2 and B-3 show the outcomes.

The geometric mean before April 25, 1945, is 3.1 pCi/day, while after this date it drops to 0.54 pCi/day. The arithmetic average of the results from April 25 through the end of the study is about 25% of the average during the previous weeks, and the average from the final four weeks is about 3% of the earlier weeks. These comparisons indicate that the concerns about the collection system before April 25 may have been justified.

In addition to the urine samples, 1483 alpha air-monitoring results were available for 1944 through 1947 at ORNL. The two largest values, both many orders of magnitude larger than the tolerance level in use at the time, were excluded from the analysis because one appeared to be a miscalculation (based on the recorded alpha activity) and the other was collected inside of a hot cell. Results were fit to a lognormal distribution (Figure B-4). The geometric mean is 5% of the tolerance level applied in ORAUT (2005a), and the 95th percentile is equal to the tolerance level. Notes indicating that assault masks had been worn were present for just under 10% of the samples; many of the samples were collected inside of hoods.

Given this information, ORAUT (2005a) appears to be a reasonable upper bound because the 95th percentile of the air-monitoring results coincides with the value applied in that document. Assault masks were used in many situations, and many of the results exceed the activity to which the individual would have been exposed because samples were collected inside of hoods. There is conservatism built into ORAUT (2005a) by the inclusion of alpha emitters with large dose coefficients that were likely not present or present only in very small quantities.

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Because it is not known who wore the assault masks and where specific individuals worked, it is difficult to postulate a smaller airborne concentration. Urine sample results are somewhat ambiguous because activity was detected in them but they were likely contaminated during collection or analysis.

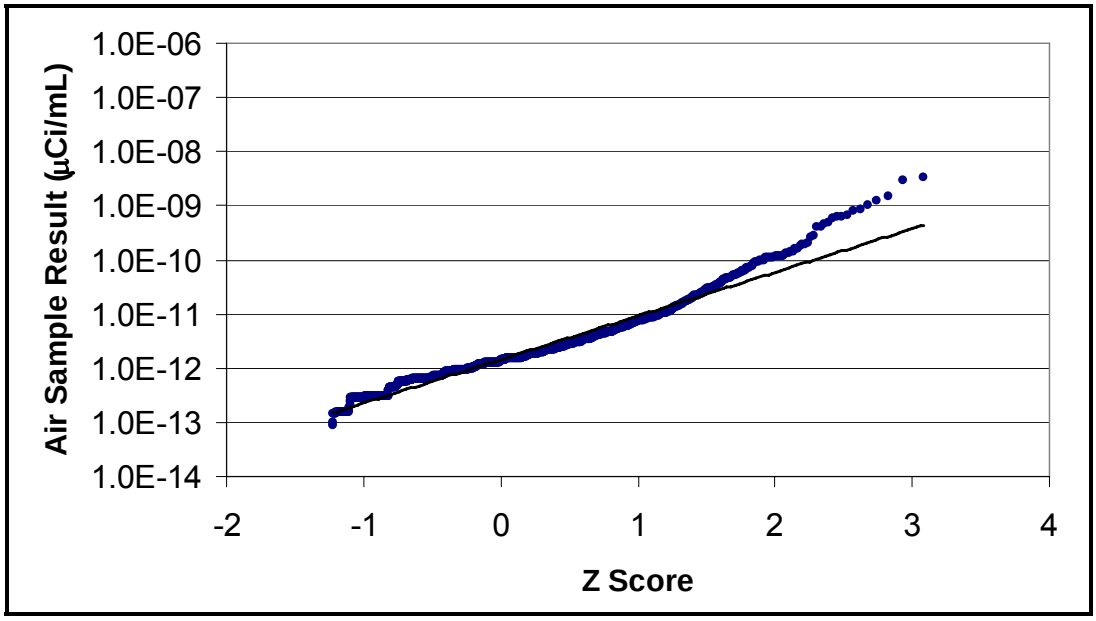

Figure B-4. Alpha air-monitoring results for 1944–1947. The geometric mean = 1.4×10^{-12} µCi/mL, the GSD = 6.35, and $r^2 = 0.97$.

As a best estimate for individuals likely to have a potential for exposure, ORAUT (2005a) values will be used but the nuclides will be limited to those likely to be present in significant quantities (plutonium and uranium isotopes), thus eliminating unnecessary conservatism. Due to the extent and quality of the alpha air-monitoring program, this approach is considered to be applicable to beta/gamma-emitting radionuclides at ORNL during this period.