07-18-11P01:15 RCVD

July 13, 2011

SEC Petition
Division of Compensation Analysis and Support
NIOSH
4676 Columbia Parkway, MS-C-47
Cincinnati, OH 45226

Re: **Employee:**
Employee SS#:
Petitioner:
Survivor:

To Whom It May Concern:

Enclosed please find a Special Exposure Cohort (SEC) Petition for the Oak Ridge National Laboratory (X-10) concerning January 1, 1943 through December 31, 1952. I have also enclosed a Petitioner Authorization Form, signed by which authorizes as the petitioner for this SEC Petition.

I have additionally enclosed continuation pages, with additional information concerning section F.1, and attachments referenced in section F.1 of this petition.

I trust that you will ensure this petition is filed. Thank you for your attention to this matter.

Very truly yours,

Enclosures

Special Exposure Cohort Petition
under the Energy Employees Occupational
Illness Compensation Act

U.S. Department of Health and Human Services
Centers for Disease Control and Prevention
National Institute for Occupational Safety and Health

OMB Number: 0920-0639 Expires: 09/20/2013

Special Exposure Cohort Petition — Form B

General Instructions on Completing this Form *(complete instructions are available in a separate packet)*:

Except for signatures, please **PRINT** all information clearly and neatly on the form.

Please read each of Parts A — G in this form and complete the parts appropriate to you. If there is more than one petitioner, then each petitioner should complete those sections of parts A – C of the form that apply to them. Additional copies of the first two pages of this form are provided at the end of the form for this purpose. A maximum of three petitioners is allowed.

If you need more space to provide additional information, use the continuation page provided at the end of the form and attach the completed continuation page(s) to Form B.

If you have questions about the use of this form, please call the following NIOSH toll-free phone number and request to speak to someone in the Division of Compensation Analysis and Support about an SEC petition: 1-877-222-8570.

If you are:			
	☐ A Labor Organization,	Start at D	on Page 3
	☐ An Energy Employee (current or former),	Start at C	on Page 2
	☐ A Survivor (of a former Energy Employee),	Start at B	on Page 2
	☑ A Representative (of a current or former Energy Employee),	Start at A	on Page 1

A Representative Information — Complete Section A if you are authorized by an Employee or Survivor(s) to petition on behalf of a class.

A.1 **Are you a contact person for an organization?** ☐ Yes (Go to A.2) ☐ No (Go to A.3)

A.2 **Organization Information:**

Name of Organization: ______________________

Position of Contact Person: ______________________

A.3 **Name of Petition Representative:**

Mr./Mrs./Ms. First Name Middle Initial Last Name

A.4 **Address:**

Street Apt # P.O. Box

City State Zip Code

A.5 **Telephone Number:** (____) ____-________

A.6 **Email Address:** ______________________

A.7 ☑ Check the box at left to indicate you have attached to the back of this form written authorization to petition by the survivor(s) or employee(s) indicated in Parts B or C of this form. An authorization

Name or Social Security Number of First Petitioner: ______________________

Special Exposure Cohort Petition — Form B

B Survivor Information — Complete Section B if you are a Survivor or representing a Survivor.

B.1 **Name of Survivor:**

Mr./Mrs./Ms. First Name Middle Initial Last Name

B.2 **Social Security Number of Survivor:**

B.3 **Address of Survivor:**

Street Apt # P.O. Box

City State Zip Code

B.4 **Telephone Number of Survivor:** (   )

B.5 **Email Address of Survivor:**

B.6 **Relationship to Employee:** ☐ Spouse ☐ Son/Daughter ☐ Parent ☐ Grandparent ☐ Grandchild

Go to Part C.

C Employee Information — Complete Section C UNLESS you are a labor organization.

C.1 **Name of Employee:**

Mr./Mrs./Ms. First Name Middle Initial Last Name

C.2 **Former Name of Employee** (e.g., maiden name/legal name change/other):

Mr./Mrs./Ms. First Name Middle Initial Last Name

C.3 **Social Security Number of Employee:**

C.4 **Address of Employee** (if living):

Street Apt # P.O. Box

City State Zip Code

C.5 **Telephone Number of Employee:** (   ) -

C.6 **Email Address of Employee:**

C.7 **Employment Information Related to Petition:**

C.7a Employee Number (if known):

C.7b Dates of Employment: Start 1945 End 1975

C.7c Employer Name: E.I. DuPont (for Univ. of Chicago), Monsanto Chemical Company, Union Carbide

C.7d Work Site Location: Oak Ridge National Laboratory (X-10)

C.7e Supervisor's Name:

Go to Part E

Name or Social Security Number of First Petitioner:

D Labor Organization Information — Complete Section D ONLY if you are a labor organization.

D.1 **Labor Organization Information:**

____
Name of Organization

____
Position of Contact Person

D.2 **Name of Petition Representative:**

____

D.3 **Address of Petition Representative:**

____
Street Apt # P.O. Box

____
City State Zip Code

D.4 **Telephone Number of Petition Representative:** (____) ____-________

D.5 **Email Address of Petition Representative:** ______________________________

D.6 **Period during which labor organization represented employees covered by this petition** (please attach documentation): Start ____________ End ____________

D.7 **Identity of other labor organizations that may represent or have represented this class of employees** (if known):

____

Go to Part E.

Name or Social Security Number of First Petitioner: ______________________________

E Proposed Definition of Employee Class Covered by Petition — Complete Section E.

E.1 **Name of DOE or AWE Facility:** Oak Ridge National Laboratory (X-10)

E.2 **Locations at the Facility relevant to this petition:**

All areas and buildings within the perimeter of the X-10 plant.

E.3 **List job titles and/or job duties of employees included in the class. In addition, you can list by name any individuals other than petitioners identified on this form who you believe should be included in this class:**

All contractor employees, subcontractor employees, and Atomic Energy Commission employees who were monitored or should, have been monitored for any of the various radionuclides and fission products present at the X-10 plant during the proposed time period

E.4 **Employment Dates relevant to this petition:**

Start January 1943 End December 1952

Start ________ End ________

Start ________ End ________

E.5 **Is the petition based on one or more unmonitored, unrecorded, or inadequately monitored or recorded exposure incidents?:** ☐ Yes ☑ No

If yes, provide the date(s) of the incident(s) and a complete description (attach additional pages as necessary):

Name or Social Security Number of First Petitioner: ________

F Basis for Proposing that Records and Information are Inadequate for Individual Dose — Complete Section F.

Complete **at least one** of the following entries in this section by checking the appropriate box and providing the required information related to the selection. You are not required to complete more than one entry.

F.1 ☑ I/We have attached either documents or statements provided by affidavit that indicate that radiation exposures and radiation doses potentially incurred by members of the proposed class, that relate to this petition, were not monitored, either through personal monitoring or through area monitoring.

(Attach documents and/or affidavits to the back of the petition form.)

Describe as completely as possible, to the extent it might be unclear, how the attached documentation and/or affidavit(s) indicate that potential radiation exposures were not monitored.

Specifically, concerning all radionuclides, for the Oak Ridge National Laboratory (X-10):

There is little or no in vitro bioassay sampling data available for reconstructing the internal dose for the years 1943-1953. There are no in vivo records available for reconstructing the internal dose for the years 1943-1953. Concerning in vitro sampling, the lack of availble samples is documented for all radionuclides other than plutonium 239, uranium, and cesium, in the graphs provided in Attachment A in ORAUT-TKBS-0012-5. These graphs provide the proof that very few urine samples were found for all other radionuclides present at X-10.

F.2 ☐ I/ We have attached either documents or statements provided by affidavit that indicate that radiation monitoring records for members of the proposed class have been lost, falsified, or destroyed; or that there is no information regarding monitoring, source, source term, or process from the site where the employees worked.

(Attach documents and/or affidavits to the back of the petition form.)

Describe as completely as possible, to the extent it might be unclear, how the attached documentation and/or affidavit(s) indicate that radiation monitoring records for members of the proposed class have been lost, altered illegally, or destroyed.

Name or Social Security Number of First Petitioner: ____________________

F.3 ☐ I/We have attached a report from a health physicist or other individual with expertise in radiation dose reconstruction documenting the limitations of existing DOE or AWE records on radiation exposures at the facility, as relevant to the petition. The report specifies the basis for believing these documented limitations might prevent the completion of dose reconstructions for members of the class under 42 CFR Part 82 and related NIOSH technical implementation guidelines.

(Attach report to the back of the petition form.)

F.4 ☐ I/We have attached a scientific or technical report, issued by a government agency of the Executive Branch of Government or the General Accounting Office, the Nuclear Regulatory Commission, or the Defense Nuclear Facilities Safety Board, or published in a peer-reviewed journal, that identifies dosimetry and related information that are unavailable (due to either a lack of monitoring or the destruction or loss of records) for estimating the radiation doses of employees covered by the petition.

(Attach report to the back of the petition form.)

G Signature of Person(s) Submitting this Petition — Complete Section G.

All Petitioners should sign and date the petition. A maximum of three persons may sign the petition.

[redacted] 7-13-2011
Signature Date

Signature Date

Signature Date

Notice: Any person who knowingly makes any false statement, misrepresentation, concealment of fact or any other act of fraud to obtain compensation as provided under EEOICPA or who knowingly accepts compensation to which that person is not entitled is subject to civil or administrative remedies as well as felony criminal prosecution and may, under appropriate criminal provisions, be punished by a fine or imprisonment or both. I affirm that the information provided on this form is accurate and true.

Send this form to:

SEC Petition
Division of Compensation Analysis and Support
NIOSH
4676 Columbia Parkway, MS-C-47
Cincinnati, OH 45226

[illegible]

Public Burden Statement

Public reporting burden for this collection of information is estimated to average 5 hours per response, including time for reviewing instructions, gathering the information needed, and completing the form. If you have any comments regarding the burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, send them to CDC Reports Clearance Officer, 1600 Clifton Road, MS-E-11, Atlanta GA, 30333; ATTN:PRA 0920-0639. Do not send the completed petition form to this address. Completed petitions are to be submitted to NIOSH at the address provided in these instructions. Persons are not required to respond to the information collected on this form unless it displays a currently valid OMB number.

Privacy Act Advisement

In accordance with the Privacy Act of 1974, as amended (5 U.S.C. § 552a), you are hereby notified of the following:

The Energy Employees Occupational Illness Compensation Program Act (42 U.S.C. §§ 7384-7385) (EEOICPA) authorizes the President to designate additional classes of employees to be included in the Special Exposure Cohort (SEC). EEOICPA authorizes HHS to implement its responsibilities with the assistance of the National Institute for Occupational Safety (NIOSH), an Institute of the Centers for Disease Control and Prevention. Information obtained by NIOSH in connection with petitions for including additional classes of employees in the SEC will be used to evaluate the petition and report findings to the Advisory Board on Radiation and Worker Health and HHS.

Records containing identifiable information become part of an existing NIOSH system of records under the Privacy Act, 09-20-147 "Occupational Health Epidemiological Studies and EEOICPA Program Records. HHS/CDC/NIOSH." These records are treated in a confidential manner, unless otherwise compelled by law. Disclosures that NIOSH may need to make for the processing of your petition or other purposes are listed below.

NIOSH may need to disclose personal identifying information to: (a) the Department of Energy, other federal agencies, other government or private entities and to private sector employers to permit these entities to retrieve records required by NIOSH; (b) identified witnesses as designated by NIOSH so that these individuals can provide information to assist with the evaluation of SEC petitions; (c) contractors assisting NIOSH; (d) collaborating researchers, under certain limited circumstances to conduct further investigations; (e) Federal, state and local agencies for law enforcement purposes; and (f) a Member of Congress or a Congressional staff member in response to a verified inquiry.

This notice applies to all forms and informational requests that you may receive from NIOSH in connection with the evaluation of an SEC petition.

Use of the NIOSH petition forms (A and B) is voluntary but your provision of information required by these forms is mandatory for the consideration of a petition, as specified under 42 CFR Part 83. Petitions that fail to provide required information may not be considered by HHS.

Name or Social Security Number of First Petitioner: ______________________________

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Special Exposure Cohort Petition
under the Energy Employees Occupational
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U.S. Department of Health and Human Services
Centers for Disease Control and Prevention
National Institute for Occupational Safety and Health

OMB Number: 0920-0639 Expires: 09/20/2013

Special Exposure Cohort Petition — Form B

Appendix — Petitioner 2

[illegible]

Use this Appendix for Petitioner 2.

This appendix form is to be used as needed. Petitioner 2, or his or her representative, should complete the parts applicable to him or her.

Refer to the General Instructions on completing petitioner information for Parts A, B, or C.

<u>If you need more space to provide additional information</u>, use the continuation page provided at the end of the form and attach the completed continuation page(s) to Form B.

Except for signatures, please **PRINT** all information clearly and neatly on the form.

If you are:	☐ An Energy Employee (current or former),	Start at C
	☐ A Survivor (of a former Energy Employee),	Start at B
	☐ A Representative (of a current or former Energy Employee),	Start at A

A Representative Information — Complete Section A if you are authorized by an Employee or Survivor(s) to petition on behalf of a class.

A.1 **Are you a contact person for an organization?** ☐ Yes (Go to A.2) ☐ No (Go to A.3)

A.2 **Organization Information:**

Name of Organization

Position of Contact Person

A.3 **Name of Petition Representative:**

Mr./Mrs./Ms. First Name Middle Initial Last Name

A.4 **Address:**

Street Apt # P.O. Box

City State Zip Code

A.5 **Telephone Number:** ( ) -

A.6 **Email Address:**

A.7 ☐ Check the box at left to indicate you have attached to the back of this form written authorization to petition by the survivor(s) or employee(s) indicated in Parts B or C of this form. An authorization form for this purpose is provided.

[illegible]

Name or Social Security Number of First Petitioner:

Special Exposure Cohort Petition
under the Energy Employees Occupational
Illness Compensation Act

U.S. Department of Health and Human Services
Centers for Disease Control and Prevention
National Institute for Occupational Safety and Health

OMB Number: 0920-0639 Expires: 09/20/2013

Special Exposure Cohort Petition — Form B

Appendix — Petitioner 2

B Survivor Information — Complete Section B if you are a Survivor or representing a Survivor.

B.1 **Name of Survivor:**

Mr./Mrs./Ms. First Name Middle Initial Last Name

B.2 **Social Security Number of Survivor:** ______________________

B.3 **Address of Survivor:**

Street Apt # P.O. Box

City State Zip Code

B.4 **Telephone Number of Survivor:** (____) ___-________

B.5 **Email Address of Survivor:** ______________________

B.6 **Relationship to Employee:** ☐ Spouse ☐ Son/Daughter ☐ Parent ☐ Grandparent ☐ Grandchild

C Employee Information — Complete Section C.

C.1 **Name of Employee:**

Mr./Mrs./Ms. First Name Middle Initial Last Name

C.2 **Former Name of Employee** (e.g., maiden name/legal name change/other):

Mr./Mrs./Ms. First Name Middle Initial Last Name

C.3 **Social Security Number of Employee:** ______________________

C.4 **Address of Employee** (if living):

Street Apt # P.O. Box

City State Zip Code

C.5 **Telephone Number of Employee:** (____) ___-________

C.6 **Email Address of Employee:** ______________________

C.7 **Employment Information Related to Petition:**

C.7a Employee Number (if known): ______________________

C.7b Dates of Employment: Start ____________ End ____________

C.7c Employer Name: ______________________

C.7d Work Site Location: ______________________

C.7e Supervisor's Name: ______________________

Name or Social Security Number of First Petitioner: ______________________

Special Exposure Cohort Petition
under the Energy Employees Occupational Illness Compensation Act

U.S. Department of Health and Human Services
Centers for Disease Control and Prevention
National Institute for Occupational Safety and Health

OMB Number: 0920-0639 Expires: 09/20/2013

Special Exposure Cohort Petition — Form B

Appendix — Petitioner 3

Use this Appendix for Petitioner 3.

This appendix form is to be used as needed. Petitioner 3, or his or her representative, should complete the parts applicable to him or her.

Refer to the General Instructions on completing petitioner information for Parts A, B, or C.

If you need more space to provide additional information, use the continuation page provided at the end of the form and attach the completed continuation page(s) to Form B.

Except for signatures, please **PRINT** all information clearly and neatly on the form.

If you are:	☐ An Energy Employee (current or former),	Start at C
	☐ A Survivor (of a former Energy Employee),	Start at B
	☐ A Representative (of a current or former Energy Employee),	Start at A

A Representative Information — Complete Section A if you are authorized by an Employee or Survivor(s) to petition on behalf of a class.

A.1 **Are you a contact person for an organization?** ☐ Yes (Go to A.2) ☐ No (Go to A.3)

A.2 **Organization Information:**

Name of Organization

Position of Contact Person

A.3 **Name of Petition Representative:**

Mr./Mrs./Ms. First Name Middle Initial Last Name

A.4 **Address:**

Street Apt # P.O. Box

City State Zip Code

A.5 **Telephone Number:** (   )   -

A.6 **Email Address:**

A.7 ☐ Check the box at left to indicate you have attached to the back of this form written authorization to petition by the survivor(s) or employee(s) indicated in Parts B or C of this form. An authorization form for this purpose is provided.

Name or Social Security Number of First Petitioner: ____________________

Special Exposure Cohort Petition
under the Energy Employees Occupational
Illness Compensation Act

U.S. Department of Health and Human Services
Centers for Disease Control and Prevention
National Institute for Occupational Safety and Health

OMB Number: 0920-0639 Expires: 09/20/2013

Special Exposure Cohort Petition — Form B

Appendix — Petitioner 3

B Survivor Information — Complete Section B if you are a Survivor or representing a Survivor.

B.1 **Name of Survivor:**

____
Mr./Mrs./Ms. First Name Middle Initial Last Name

B.2 **Social Security Number of Survivor:** ______________

B.3 **Address of Survivor:**

____
Street Apt # P.O. Box

____
City State Zip Code

B.4 **Telephone Number of Survivor:** (____) ____-________

B.5 **Email Address of Survivor:** ______________

B.6 **Relationship to Employee:** ☐ Spouse ☐ Son/Daughter ☐ Parent ☐ Grandparent ☐ Grandchild

C Employee Information — Complete Section C.

C.1 **Name of Employee:**

____
Mr./Mrs./Ms. First Name Middle Initial Last Name

C.2 **Former Name of Employee** (e.g., maiden name/legal name change/other):

____
Mr./Mrs./Ms. First Name Middle Initial Last Name

C.3 **Social Security Number of Employee:** ______________

C.4 **Address of Employee** (if living):

____
Street Apt # P.O. Box

____
City State Zip Code

C.5 **Telephone Number of Employee:** (____) ____-________

C.6 **Email Address of Employee:** ______________

C.7 **Employment Information Related to Petition:**
C.7a Employee Number (if known): ______________
C.7b Dates of Employment: Start ______________ End ______________
C.7c Employer Name: ______________
C.7d Work Site Location: ______________
____
C.7e Supervisor's Name: ______________

Name or Social Security Number of First Petitioner: ______________

Special Exposure Cohort Petition
under the Energy Employees Occupational
Illness Compensation Act

U.S. Department of Health and Human Services
Centers for Disease Control and Prevention
National Institute for Occupational Safety and Health

OMB Number: 0920-0639 Expires: 09/20/2013

Special Exposure Cohort Petition — Form B

Appendix — Continuation Page

Continuation Page — Photocopy and complete as necessary.

F-1 (cont.)

ORAUT-TKBS-0012-5 reflects that there was only one urine sample for Am- 241 in 1952, no Cm-244 urine samples were found until 1964, no gross alpha samples found until 1953, two gross beta samples were found in 1952, no H-3 samples were found until 1961, no Pm-147 samples were found until 1959, no rare earth samples were found until 1951 and only about 30 samples were found for each year from 1951-1952, no Sr-90 samples were found until 1950 with only one sample in 1950 and less than ten samples for each year 1951-1952. Appendix B, of ORAUT-TKBS-0012-5, confirms that there were "...no bioassay results appearing in the DOE worker files" from 1943-1947. Appendix B was attached to ORAUT-TKBS-0012-5 to justify utilizing the ORNL air sampling data to extrapolate internal dose. Although the air sampling data has been utilized for dose reconstructions, as justified in Attachment B of ORAUT-TKBS-0012-5, the air sampling was only for 1944-1947 and was only performed for alpha contamination. ORAUT-TKBS-0012-5, Attachment B justifies utilizing the alpha air monitoring data for beta contamination. Dose reconstructions cannot accurately be performed for X-10 workers utilizing this very limited alpha air sampling dataset and extrapolating it to additional years and to the beta emitting radionuclides. Attachment B, of ORAUT-TKBS-0012-5, references ORAUT-OTIB-0018, Rev. 01, Internal Dose Overestimates for Facilities With Air Sampling Programs for authority to apply air monitoring data to the X-10 internal dose. ORAUT-OTIB-0018 can only be utilized for "Claims for which it is likely that the covered employee had no significant intakes of particulate radioactive material." ORAUT-OTIB-0018 can only be utilized for "Intakes of particulate radioactive material only." (page 8 of 35) Confounding the previous dose reconstructions are the claimants who should have been assigned co-worker dose, because they had recorded external dose which indicated they should be assigned co-worker internal dose. Since there was no bioassay monitoring data available until 1951, and then only a limited amount, NIOSH has been assigning ambient dose for claimants and stating that the ambient dose is higher than the co-worker dose. ORAUT-OTIB-0034, Internal Dosimetry Coworker Data for X-10 documents that the co-worker dataset begins in 1951. There is simply no means to assign co-worker data prior to 1951. I have previously noted the deficiency of the internal dose from 1951-1953, which is being assigned with little or no bioassay data for the vast majority of radionuclides found at X-10 during that time frame.

An accurate dose reconstruction cannot be performed for X-10 workers from 1943-1953. NIOSH has previously performed dose reconstructions for claimants during that time period, using assumptions based on a very limited dataset.

Attach to Form B if necessary.

Name or Social Security Number of First Petitioner: ______________________

Continuation Page — Photocopy and complete as necessary.

I am attaching a copy of AECU-817, LECTURE NOTES; HEALTH PHYSICS TRAINING LECTURES, 1948-1949, September 29, 1950, Health Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee. This appears to be a series of lectures, for student Health Physics Technicians at ORNL. This document provides great insight into the Health Physics practices at ORNL, during the mid 1940's. Page 6 of this document, under Personnel Monitoring, states "External exposures are measured with the use of the pocket meter (ion chamber) and film meter. Internal exposures are determined in a limited way by a study of body fluids." All throughout this document it is evident that ORNL Health Physicists did not know the extent of radiation damage to tissues, simply because no bioassay procedures were in place. Pages 29-34 provides the rational for converting air sampling to tissue dose, under the section entitled "Relationship between Concentration of a Radioisotope in Tissue and Dosage Rate." This lecture provides several calculations to derive tissue dose from air samples and includes the assumption that a very certain amount of the quantity of the radioisotope is passed from the body by physiological elimination. Key to these calculations are the ability to determine C (= number of curies of the radioisotope in the tissue) and E (= energy per disintegration, measured in Mev). Page 32 provides insight into the difficulties of determining C & E for these equations. Page 32 states: "The principle difficulty involved in making estimates of dosage rates due to radioisotopes within the tissue is that of obtaining reliable values for C and E. It has already been indicated that the above equation is valid only if the material is uniformly distributed through the tissue. In many cases the extent to which this is true is difficult to estimate. Further difficulty arises from inadequate knowledge of rapidity and percentage of absorption into the tissues and the rates of elimination from the tissues." "For many radioisotopes, values of E are only roughly known." "Sometimes, however, the disintegration scheme is quite complex, and, more frequently than not, it is not definitely known." On pages 34-36 we find a lecture concerning "CONTAMINATION AND DUST PROPOGATION." This lecture provides some insight into the importance of particle size and surface area. This is stated on the bottom of page 34 as "In other words, relatively small amounts of material will give large amounts of particulates." Page 36 continues to stress the fact that particle size is important in regard to whether the particle is retained in the lung. Another telling statement, on page 36 is "The lack of pertinent biological data has been such that it is impossible to properly evaluate the actual physiological hazard caused by the inhalation of insoluble radioactive particles." Page 61, under the section entitled "CALIBRATION TECHNIQUE" in regard to

Name or Social Security Number of First Petitioner: ____________________

calibrating their detection instrumentation, states "We do not at the present time calibrate for beta. There are many problems involved in the calibration of beta, such as various energy ranges, etc., which make the problem much more complicated than gamma calibration. It is recognized that beta calibration is desirable, and at the present time people in the Research Section are working on a method for calibration of beta." This proves that accurate measurements of the beta emitting radionuclides was impossible in 1949, and any measurements of beta must be considered inaccurate. Page 69 begins the lecture entitled "PERSONNEL MONITORING PROCEDURES" and states "Effective personnel monitoring involves measurements of both external and internal radiation. External radiation is monitored at present by pocket meters (ionization chambers) and film meters. Internal radiation presents two problems in that we are faced with both radiation and toxic effects. Internal radiation is measured by a study of body fluids (this method is discussed in Lecture 6.3)." On page 74 we find lecture 6.3 entitled "MEASUREMENTS ON INTERNAL EXPOSURE." This particular lecture is very short and consists of only five paragraphs. Paragraph five states "Since the organization of a group for body fluids analysis at Clinton Laboratories in July 1945, the main concern has been the development of procedures of chemical isolation of radioactive elements, followed by the perfecting of these procedures to a point where they were usable. The development of instruments which are reliable has been one of the major problems. The quantity of any radioactive material to be found in either urine or feces is exceedingly small. Therefore the counting instruments must be dependable over a long period of time." Although it is not specifically stated, it is clear that ORNL had no reliable means of analyzing biological samples in 1949. The lack of biological data prohibited ORNL from accurately assessing exposures, and this is a prevailing message throughout this lecture series.

This leads us back to determine the accuracy of ORAUT-OTIB-0018, Rev. 1, which allows NIOSH dose reconstructors to deviate from the stated applicible dates of 1953 the present for this technical information bulletin, and apply this TIB to ORNL from 1944-1952 for both alpha and beta emitting radionuclides. This TIB is used to assign internal dose when air sampling was the only method for determining dose. It basically assumes that since ORNL had a program in place to limit exposures by means of a rigorous air sampling scheme, internal doses can be reconstructed from these limiting concentrations. Although ORNL did perform air sampling, and did report data in regard to limiting exposures, AECU-817 does reflect that ORNL was still struggling with instrumentation, calibration, and lack of any biological

Name or Social Security Number of First Petitioner: ____________________

samples to determine the effectiveness of the air sampling program at ORNL. These lectures reveal that ORNL was in its infancy, concerning extrapolation of air sampling to damage to biological tissues, instrumentation to determine the accuracy of their sampling program, and the determination of the particle size of the contaminants. It is not logical to accept all of the air samples accuracy, and those air samples to extrapolated to internal dose, when it is clear that the ORNL methods were exceptionally crude, by today's standards.

This SEC petition has selected an end date of December 31, 1952, due to the fact that ORAUT-OTIB-0018 was developed to apply from 1953-the present. Although an exception to that date of application was specifically made for ORNL, it is becoming clear that ORNL exception was granted without closely reviewing ORNL's crude methods for gathering and analyzing air samples. It is very clear that co-worker internal dose cannot be assigned prior to 1951. It is very clear that all radionuclides were not addressed in either air sampling or the very limited number of in vitro samples performed prior to 1953. As previously mentioned a majority of radionuclides, present at ORNL were not considered in bioassay sampling until sometime later than 1953. Attachment B, for ORAUT-TKBS-0012-5, Rev. 1, states that there were "1483 alpha air-monitoring results available for 1944-1947 at ORNL." This is apparently the evidence that shows that ORNL had a rigorous air sampling program in place and allowed ORAUT-OTIB-0018 to determine internal dose for workers at ORNL. 1483 air samples reflects only 1.02 air samples per day during this time period. NIOSH is basically assigning internal dose, for ORNL workers, based upon 1483 questionable air samples performed from 1944-1947, and extrapolating that data to cover the entire period from 1943-1953, when ORAUT-OTIB-0018 does not cover the numerous radionuclides for which no sampling was performed, and also specifically does not cover radioactive gases (such as tritium which was prevalent at ORNL), and does not cover workers who were likely to have incurred exposures of particulate radioactive material (page 8 of 35 in ORAUT-OTIB-0018).

It is not feasible that 1.02 air samples daily (1483 from 1944-1947) could possibly provide the necessary data to limit everyday exposures to even the sampled radionuclides, which I assume to be plutonium, uranium, and possibly strontium. Added to this extremely limited amount of air sampling is the fact that these 1944-1947 air samples are extrapolated to cover all years from 1943-1953, then they are again extrapolated to determine internal dose. Internal co-worker data is also not available prior to 1951, according to ORAUT-OTIB-0034, so this very limited dataset of air samples is utilized for any co-worker doses that must be addressed in dose reconstructions.

It is time for NIOSH to finally realize that dose reconstructions cannot be accurately performed for workers at ORNL from January 1, 1943 to December 31, 1952, and accept this class to the Special Exposure Cohort.

Attachments:

ORAUT-TKBS-0012-5 Oak Ridge National Laboratory Occupational Internal Dose
ORAUT-OTIB-0018 Internal Dose Overestimates for Facilities with Air Sampling Programs
ORAUT-OTIB-0034 Internal Dosimetry Coworker data for X-10
AECU-817 Lecture Notes; Health Physics Training Lectures, 1948-1949, Health Physics Division, Oak Ridge National Laboratory, Oak Ridge, TN

Name or Social Security Number of First Petitioner: ______

Special Exposure Cohort Petition
under the Energy Employees Occupational Illness Compensation Act

U.S. Department of Health and Human Services
Centers for Disease Control and Prevention
National Institute for Occupational Safety and Health

OMB Number: 0920-0639 Expires: 09/20/2013

Petitioner Authorization Form

Use of this form and disclosure of Social Security Number are voluntary. Failure to use this form or disclose this number will not result in the denial of any right, benefit, or privilege to which you may be entitled.

Instructions:

If you wish to petition HHS to consider adding a class of employees to the Special Exposure Cohort and you are NOT either a member of that class, a survivor of a member of that class, or a labor organization representing or having represented members of that class, then 42 CFR Part 83, Section 83.7(c) requires that you obtain written authorization. You can obtain such authorization from either an employee who is a member of the class or a survivor of such an employee. You may use this form to obtain such authorization and submit the completed form to NIOSH with the related petition. **Please print legibly**.

For Further Information: If you have questions about these instructions, please call the following NIOSH toll-free phone number and request to speak to someone in the Division of Compensation Analysis and Support about an SEC petition: 1-877-222-8570.

Authorization for Individual or Entity to Petition HHS on Behalf of a Class of Employees for Addition to the Special Exposure Cohort

I, ____________________
Name of Class Member or Survivor

Street Address of Class Member or Survivor Apt. # P.O. Box

City, State, Zip Code of Class Member or Survivor

do hereby authorize:

Name of Petitioner

Address of Petitioner Apt. # P.O. Box

City, State and Zip Code of Petitioner

to petition the Department of Health and Human Services on behalf of a class of employees that includes:

Name of Class Member (employee, not the employee's survivor)

for the addition of the class to the Special Exposure Cohort, under the Energy Employee's Occupational Illness Compensation Program Act (42 U.S.C. §§ 7384-7385).

In providing this authorization, I recognize that the petitioner named above will have all the rights of a petitioner as provided for under 42 CFR Part 83.

Signature of Class Member or Survivor Date 7-8-2011

Name or Social Security Number of First Petitioner:

Public Burden Statement

Public reporting burden for this collection of information is estimated to average 3 minutes per response, including time for reviewing instructions, gathering the information needed, and completing the form. If you have any comments regarding the burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, send them to CDC Reports Clearance Officer, 1600 Clifton Road, MS-E-11, Atlanta GA, 30333; ATTN:PRA 0920-0639. Do not send the completed petition form to this address. Completed petitions are to be submitted to NIOSH at the address provided in these instructions. Persons are not required to respond to the information collected on this form unless it displays a currently valid OMB number.

Privacy Act Advisement

In accordance with the Privacy Act of 1974, as amended (5 U.S.C. § 552a), you are hereby notified of the following:

The Energy Employees Occupational Illness Compensation Program Act (42 U.S.C. §§ 7384-7385) (EEOICPA) authorizes the President to designate additional classes of employees to be included in the Special Exposure Cohort (SEC). EEOICPA authorizes HHS to implement its responsibilities with the assistance of the National Institute for Occupational Safety (NIOSH), an Institute of the Centers for Disease Control and Prevention. Information obtained by NIOSH in connection with petitions for including additional classes of employees in the SEC will be used to evaluate the petition and report findings to the Advisory Board on Radiation and Worker Health and HHS.

Records containing identifiable information become part of an existing NIOSH system of records under the Privacy Act, 09-20-147 "Occupational Health Epidemiological Studies and EEOICPA Program Records. HHS/CDC/NIOSH." These records are treated in a confidential manner, unless otherwise compelled by law. Disclosures that NIOSH may need to make for the processing of your petition or other purposes are listed below.

NIOSH may need to disclose personal identifying information to: (a) the Department of Energy, other federal agencies, other government or private entities and to private sector employers to permit these entities to retrieve records required by NIOSH; (b) identified witnesses as designated by NIOSH so that these individuals can provide information to assist with the evaluation of SEC petitions; (c) contractors assisting NIOSH; (d) collaborating researchers, under certain limited circumstances to conduct further investigations; (e) Federal, state and local agencies for law enforcement purposes; and (f) a Member of Congress or a Congressional staff member in response to a verified inquiry.

This notice applies to all forms and informational requests that you may receive from NIOSH in connection with the evaluation of an SEC petition.

Use of this form is voluntary. Failure to use this form will not result in the denial of any right, benefit, or privilege to which you may be entitled.

Name or Social Security Number of First Petitioner: ______________

UNITED STATES ATOMIC ENERGY COMMISSION

AECU-817

LECTURE NOTES; HEALTH PHYSICS TRAINING LECTURES, 1948-1949

September 29, 1950

Health Physics Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee

Technical Information Service, Oak Ridge, Tennessee

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LECTURE NOTES; HEALTH PHYSICS TRAINING LECTURES, 1948-1949

PREFACE

The following notes were issued in conjunction with a series of lectures given during the fall of 1948 to persons working in the field of health physics at the Oak Ridge National Laboratory, either as employees of the Health Physics Division or as trainees assigned from agencies outside the laboratory. Some of these notes were written by the lecturers, while others were taken by a stenographer and edited by the lecturer. As nearly as possible, the notes were issued concurrently with the lectures, and greater stress was placed on getting the material into the hands of the students than on the quality of preparation.

For the purpose of reproduction by the Technical Information Division of the Atomic Energy Commission, some deletions have been made by the AEC Declassification Branch, and minor corrections or changes have been made by the Health Physics Division. However, except for deletions, the notes are essentially in the form written or approved by the lecturer. Since the demand for the notes makes it expedient to release them without thorough editing, it should be understood that some of the material contained may not accurately reflect the policies or the considered thinking of the Health Physics Division, Oak Ridge National Laboratory.

The lecture series to which these notes apply was planned and coordinated by T. H. J. Burnett of the Health Physics Division. While most of the lectures were given by members of the Health Physics Division, Lecture 4.4 was given by a member of the Instrument Department and Lecture 5.10 by a member of the Physics Division. Unit 7 consisted of a series of lectures by representatives of other divisions of the laboratory. Since these dealt primarily with the work of the laboratory, no notes were issued.

UNIT 1. INTRODUCTION*

1.1 A BLUEPRINT FOR THE DEVELOPMENT OF A HEALTH PHYSICIST

(Place of Health Physics in Industry)

By K. Z. Morgan†

Dr. Morgan started his lecture by expressing a desire to meet and know the students better. He assured them that they should feel free to call upon him for any advice or problem they might have.

The first point was to give the students a knowledge of what our problem is and how it has been developed. Dr. Morgan then outlined a brief history of radiation.

X rays were discovered in 1895 by Roentgen, and in 1896 Becquerel discovered the activity of uranium. That very year Grubble, a manufacturer of X-ray tubes in Chicago, reported an erythema on his hands. This was the first recorded case of radiation damage. Following that recognition of radiation injury of the man's hands, there have been many cases of radiation damage recorded. Various types of radiation damage were discovered later, but most of the early damage was observed in medical doctors, radiologists, dentists, and radium technicians, and there was damage in varying degrees. The damage that was first observed was the erythema, but it was soon recognized that the blood frequently sustained a more serious damage. Unfortunately radiation damage occurs in a rather insidious manner. If you walk up too close to a hot plate, you recoil with pain. On the other hand, there is no sixth sense to indicate that we are receiving a radiation exposure which will account for damage or even death, and such an exposure may be received without our knowing it. The first symptom of a serious overexposure is drowsiness the next day, perhaps followed by illness 3 or 4 days later. Doses of as much as 100,000 r (roentgens) would kill a person in a matter of minutes. The last stages of a case following a fatal radiation exposure (of, for example, 600 r) would result in much suffering. The medical profession is only able to ease the pain of such an unfortunate individual.

The object of Health Physics is not treatment after exposure, but to make certain that no one receives exposures of consequence.

Other effects of radiation include the following: The body may be made more susceptible to some diseases. It has been found to be rather definite that if a person is exposed to radiation, it reduces his expected life. For example, if a person receives our currently proposed tolerance level (which is 0.3 r/week) for 35 years, it has been calculated that he would lose about 1.5 years of his expected life. We are not sure of this exact figure, but we are sure that radiation does reduce the expected life span on the average. Radiation causes cancer, and ordinarily cancer does not show up until 10 or 15 years after the exposure. It would be necessary to see these persons 10 or 15 years after certain radiation exposures in order to see that any damage had been done. Genetic effects may not show up for several generations. Thus we see that in most cases it is not possible to observe radiation effects immediately.

This was the sort of picture of the past that we had when the Plutonium Projects began in 1942. The medical men, in looking over the records, found that man, in working with 2 lb of radium, had produced a great deal of damage; some died from it, and others suffered because of the radiation. A. Compton, Fermi, Stone, and others actually considered whether it was advisable to go ahead with this

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*This series of lectures was sponsored by the Radiation Survey-Monitoring Section of the Health Physics Division, Oak Ridge National Laboratory and conducted under the direction of T. H. J. Burnett, Technical Adviser.

†Notes taken by C. Scott; edited by K. Z. Morgan.

project, assuming that so many persons had been killed with 2 lb of radium (this was all the radium that had been produced at that time). They asked how many persons would be expected to die as a result of pile operations with the equivalent of millions of pounds of radium. Fortunately it was decided to go ahead with the Plutonium Projects, but to put forth an unprecedented effort for radiation protection in addition.

Development in pile research and operation of piles has developed simultaneously with the field of Health Physics. Initially it was thought that Health Physics would be a medical problem, and, as a result, we reported to the medical group here when Clinton Laboratories began operations. Later it was found that if we did our part appropriately, there would be little need for medical advisors on problems of damage.

One by one the Health Physics Divisions throughout the Plutonium Projects have obtained maturity and have become full-size divisions. This has been true primarily at the Argonne National Laboratory, Hanford Works, and at Oak Ridge National Laboratory (ORNL). At the University of Chicago the first Health Physics Division was organized under the direction of E. O. Woolan, who is now Assistant Director of the Physics Division here. J. C. Hart, R. R. Coveyou, a few others, and myself joined that group in 1943. H. M. Parker and C. Gamertsfelder had joined it in the latter part of 1942. During the fall of 1943, Parker, Coveyou, Gamertsfelder, Hart, and myself came to ORNL (then Clinton Laboratories), and we set up a Health Physics Division and began a training program for personnel who were scheduled to move out to Hanford. It was our opinion at that time that in a Health Physics organization all that was needed was a few senior men to serve in an advisory and consultation capacity, and all the survey work would be done by Operations. We have had to change that philosophy.

Radiation is actually no more dangerous than any other hazard so long as safety rules are complied with. The only difference is that we do not have a physical sense which tells us when we are receiving an exposure.

We were made to realize from our early experiences that operations and research personnel often became so impressed with the work they were doing that they were inclined at times to ignore certain hazards. Research personnel who thought they knew all the answers were sometimes the worst, because they had the attitude that it was their necks if they wanted to take chances.

With the help of the Manhattan District and the University of Chicago, our local contractor, we developed a new philosophy in that we actually did do a great deal of the service work out in the field. In addition to pocket meters and film badges (our only service function then), it was decided to train a group of Health Physics surveyors in the use of instruments and interpretation of instrument readings and see that radiation protection measures were enforced. A month or so after operations got under way at ORNL, this new philosophy was developed, and it has been held ever since.

During the past year our philosophy has changed again somewhat. We have been convinced that we have been doing too much of the scrub-up type of work for operations and research and that when they make a mess, the best way to impress on them that they should be more careful is to let them clean it up themselves. At Chicago, during the past two years, they have had a special clean-up squad to wash up radiation messes. Here and at Hanford we feel that a chemist or top-notch operations man should do his own clean-up work, and therefore spills do not occur very often. Therefore we have encouraged the research personnel and operations to do these routine jobs wherever possible, and we have been pleased to have their full cooperation. If equipment is to be decontaminated, we show them how to use the instruments. We hope to have the research and operations personnel so trained and equipped with instruments in the near future that they can do their own clean-up work. This is not doing Health Physics out of a job, since it gives us more time for other things that are more important to be done.

The problems presented by such an operation as we have here are somewhat different from the problems of the hospital radiologist. Here we not only have alpha, beta, gamma, and X rays, but we also have fast neutrons and thermal neutrons. In addition to these particles, we have the recoil fission elements and the neutrinos. We also have epithermal neutrons. However, we do not have much knowledge of the biological damage associated with neutrinos or epithermal neutrons.

Dr. Morgan then outlined the principal objectives of the Health Physics students:

1. Make a Study of Various Radiation Problems. One of these problems is shielding. In order to protect personnel by shielding, it is necessary to know something about the nature of radiation and how to shield one from this radiation. If you are working in the pile building, you know that you have to put up a certain shield to stop the neutrons and would prefer to use light material such as paraffin to slow down or stop the neutrons. You might mix paraffin with iron (for fast neutrons) or boron (to capture thermal neutrons), but then, if you left it at that, you would run into trouble because fast neutrons react with iron and slow neutrons with hydrogen to produce gamma radiation. It would then be necessary, in addition, to surround the paraffin mixture with lead or heavy material to stop the gamma rays which are produced by slowing down and capture of neutrons. Shielding of radiation can be quite a complex problem.

2. Develop Methods of Preventing or Minimizing Radiation Exposures. Along with the idea of making the study of radiation, the next step is making a study of how to prevent radiation exposure. This is our primary objective. Shielding is not the only method of preventing radiation exposures. For example, using psychological means or "horse sense" in dealing with people may be an important element in obtaining cooperation and preventing overexposure to radiation. In preventing radiation exposures you will find quite a challenge. If you develop any new methods of minimizing radiation exposure, bring it to the attention of the supervisor so that he can pass it on to others.

3. Understand the Operation and Interpretation of Radiation-detection Instruments. I do not mean by this that each of you needs to become an expert in electronics. A few of you may go on in electronics or some other branch of physics. I hope that quite a few of you will take additional college work and advance in these fields. A large majority will certainly not become specialists in electronics, but I do hope that each of you will understand the fundamentals of instrument usage, how the instruments operate, their limitations, etc. Film meters, for example, have certain inherent limitations. Understand what they are good for and what they are not good for. Know why a film meter may give a different reading from a Cutie Pie; if a supervisor wishes to know why these two meters read differently, give the whole radiation story to the supervisor, and I am sure that you will gain the complete respect of the people with whom you work.

4. Understand the Rules and Procedures for Radiation Protection. There is no substitute for this. I have to study the rules and procedures every day or so, but I suggest, to begin with, that you obtain a copy of the manual of rules and procedures for working with radioactive material and practically memorize the rules and procedures for the area in which you are working.

As you acquire a knowledge of all operations relative to Health Physics, you become a senior man and take on more responsibility. I have had this question asked: Here is a fellow in Health Physics working side by side with someone else. Both are doing the same type of work, but one is making 30 per cent more than the other. Why is this? The answer is that the person who is receiving 30 per cent more not only knows how to do this particular job, but he knows how to carry out Health Physics functions on 10 other operations. He is a much more useful man, although he may be doing just the same type of work as a less experienced man on a particular job. Look forward to familiarizing yourself with all the varied Health Physics survey functions. If you have research interests, add to your education, and there is no reason why you cannot go on and become a research Health Physicist. If you wish to be a supervisor, make a study of that field. There is plenty of chance for advancement in Health Physics.

5. Maximum Permissible Levels of Exposure. You should know the maximum permissible level of exposure for every operation with which you are associated, or know how to find them in a hurry. It is not enough to know that the present level is 1/10 r/day and the new level is 3/10 r/week. You want to know the permissible contamination on the hands, when a person should wear gloves, and when he should wear complete protective clothing or just a lab coat. Never use the word "tolerance." Replace it with "maximum permissible exposure." No radiation is tolerable. We all take some radiation exposure and can never get away from the natural radiation of the earth, but we should not tolerate any unnecessary radiation exposure.

6. Become Thoroughly Familiar with Employee Relations and Supervisory Relations. This will be discussed further in later lectures. Later on you will be able to attend supervisory lectures. Health Physics occupies a rather delicate position as far as management is concerned. We tie into

all operations at various levels. You are directing the radiation-protection procedure on a job, and yet on this job you probably find people working who are assigned to it from a number of divisions or departments. I am sure you have found out already that you do not go up to a plumber and tell him to get away from what he is doing before he has obtained a slight overexposure. In practically all cases we work with supervision and try to offer advice and recommendations to the supervisor in the area in which we work, and he passes these recommendations on down to the people under him. If damage may result to the plumber from what he is doing, then try to remove him from the area in which he is working as quickly and quietly as possible. Otherwise, go to the supervisor. If no cooperation is available, contact your own supervisor, and he will see that the case is taken care of. Do not cut channels except in serious emergencies where life is endangered. Offer advice where and when it will do the most good and in a manner that will be accepted. This is a challenge to each individual. Some are born with a good personality and a knack of getting along with people, but others must develop it.

7. Have a General Knowledge of the Total Laboratory Operation, with a Working Knowledge of the Various Operating Techniques in Your Particular Assigned Area. This is also a matter of psychology, in that you must be able to create opportunities to learn of the operation without taking up someone's time or aggravating him.

Previously I pointed out the fact that if you become a good Health Physicist here, you are not necessarily limited to ORNL or to Oak Ridge. There is a great need for Health Physicists in many walks of life and in all countries of the world. Wherever you have people working with radioactive materials or instruments that protect from radiation, you find a need for Health Physicists. The greatest need has been, up to now, on the AEC projects. Altogether, we have trained in our division roughly 150 persons in Health Physics, some for periods of one year or more, and others for months, weeks, or days. We have sent trained personnel to Mound, Hanford, Los Alamos, and general AEC operations. Many opportunities in the AEC operations still exist, and these operations are quite anxious to get additional personnel. Our division is conducting at this time a Health Physics training program for 10 persons from other laboratories.

As time goes on, the need for Health Physicists and/or Radiological Physicists in hospitals becomes more urgent. There is going to be a cancer research program at our local hospital, and a project is starting in Knoxville which will need Health Physicists on the staff. Various universities are taking this job seriously Purdue, Chicago, M. I. T., and others are sending men here for training, and they are going back to set up their own nuclei of organizations. The steel industry, using C^{14} and radioactive iron, is anxious to get started in tracer work and to obtain Health Physicists. The petroleum industry has indicated that it would like to send people here for training. The Army, Navy, and Air Force need senior and junior Health Physicists The Public Health Service will need quite a number of persons trained in the field of Health Physics. Some senior men from the Public Health Service are in training here now. TVA has a senior man here in training. The transportation industry will be anxious to get a few senior men trained in Health Physics because they are responsible for seeing that all shipments of radioactive material are made safely. Insurance companies are interested and have inquired about the possibility of organizing a school for insurance specialists. An effort is already under way to give safety engineers some elementary but fundamental training in Health Physics. Many research organizations are interested in securing Health Physicists.

1.2 HEALTH PHYSICS ORGANIZATION AT OAK RIDGE NATIONAL LABORATORY

By J. C Hart*

Mr. Hart's lecture was primarily concerned with explaining the organization of the AEC and the laboratory to the students, with the aid of diagrams and charts which he had previously put on the blackboard. He spent some time in explaining the various job titles and specific differences in research and operations.

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*Notes taken by C Scott; edited by J. C. Hart.

A department breakdown was given as follows:

It is the primary function of the Survey-Monitoring Section to prevent radiation damage in any of its various forms to living matter. The present Survey-Monitoring organization was developed along these lines with a design which would make individual groups independent, with a minimum of overlapping functions. It has been agreed that four distinct groups are essential to achieve this objective. They are as follows:

Survey. It is the responsibility of the Survey group to assay or survey conditions throughout the laboratory for the purpose of establishing tolerance levels, contamination control, etc. It can be said that the successful control of the exposure of personnel to radioactive material and resultant control over contamination lies with the Survey group. With instruments designed to aid in the quantitative and qualitative analysis of various types of radiation, the surveyor aids the laboratory operation by keeping informed of potential damage which any given operation might produce.

Area Monitoring. In defining the duties of the Area Monitoring group, it can be said that this group is interested in radioactive matter as associated with things. An example of this is a contaminated chair, wastebasket, or piece of laboratory apparatus. Other examples, in a broader sense, are air, earth, and water as related to area contamination.

Currently this group consists of three major units. Field Analysis is interested in sampling of soil surrounding the X-10 area and analyzing contents of the White Oak Creek drainage system and on down the Clinch River valley. Incidents of high concentrations of contamination are investigated by this group when found, and proper precautions are instituted. Meteorological Service is interested in air contamination locally and regionally. For operational convenience the laboratory personnel which provides analyses on samples from the field is under the same supervision as is the meteorological group. Two other units of operation, currently under common supervision, are known as Laundry Monitoring and the Counting Facility, employing a staff of technicians skilled in the use of the Geiger counter. The Laundry Monitoring group checks all clothing issued by the various operating departments for protection against contamination. Before the garments are washed the garments are then divided into two groups: those which are hot and those which are free of contamination. Hot clothes are given a special wash designed to remove most types of contaminants, after which the garments are rechecked, and, when thoroughly decontaminated, these are put back into circulation. The Counting Facility could perhaps function in any of the Survey-Monitoring groups or even exist as an independent group. However, since the majority of samples originate with the other units within Area Monitoring, the Counting Facility has been placed there.

Personnel Monitoring. As the name implies, the Personnel Monitoring group is interested in the quantitative doses of radiation which involve people. This group attempts to determine the extent of exposure to radiations from external sources and, to a limited extent, to make some attempt at determining internal exposure. External exposures are measured with the use of the pocket meter (ion chamber) and film meter. Internal exposures are determined in a limited way by a study of body fluids.

Instruments Group. In a larger sense, the Instruments group is a service group catering to the three above-mentioned groups. It is the responsibility of this group to provide for the procurement, maintenance, and development of radiation-detection instruments.

Procurement requires an up-to-date study of available instruments and an accurate prediction of the needs within the groups in order that adequate instruments are available to them. Routine maintenance is the major function in that radiation-detection instruments require lengthy and frequent calibrations. Development is activated in proportion to the birth of new ideas. The Instruments group currently consists of three units: a calibration unit utilizing four technicians performs precise calibration techniques using beta, gamma, and alpha rays of varying intensities and energies; a repair unit consisting of three instrument mechanics performs periodic examinations on the instruments for the purpose of detecting defects; and a development group consisting of one engineer, two mechanics, and one technician is in constant search for minor improvements on the instruments currently in use.

UNIT 2. FUNDAMENTAL PHYSICS

2.1 ATOMIC STRUCTURE

By Ross S. Thackeray*

An atom is far smaller than anything which can be observed with direct means such as a microscope. As the result, knowledge of atomic structure is limited to a series of models which seem to account for the results of indirect experiments. Even though large numbers of models have been proposed and then discarded as the result of further experiments, we shall proceed as though we had complete, definite knowledge.

The first concepts of atomic structure pictured an atom as a hard, solid sphere somewhat like a ball bearing. This picture was used for a few years and seemed to work reasonably well until Rutherford performed his experiment (Fig. 2.1) with alpha particles at the Cavendish laboratory in Cambridge, England. This experiment consisted in the bombardment of a thin gold foil with alpha particles. The foil was so thin that most of the alpha particles passed through. However, it was found that in penetrating the foil these particles were directed from their original path by a process which came to be known as "scattering" (the assumed picture of scattering behavior from per cent distribution is shown in Fig. 2.2). The scattered alpha rays were detected with scintillation screens, and the angular distribution of the scattered rays was determined. From these data Rutherford drew the conclusion that the foil was not made up of closely packed solid spheres, but instead was made up of spheres with small positively charged dense centers surrounded by a large volume of virtually empty space.

Some idea was known of the distance between centers of atoms in thin gold film, since X-ray experiments had already given that information. The size of the positive nucleus, demonstrated by the alpha-ray experiments, was very much smaller than the distance between the atoms (Fig. 2.2).

More information about the extranuclear structure of the atom was obtained by such experiments as foil bombardments, using high-speed electrons. It was found that an incident electron could eject one or more other electrons from the target material (see Fig. 2.3). Knowing that an atom as a whole had to be electrically neutral, Rutherford then proposed that the nucleus was surrounded by a field of electrons which were very much lighter than the nucleus and carried a negative charge.

Prior to the discovery of the neutron the fundamental atomic particles known were only two in number. To fit the observed atomic mass and charge data, it was proposed that the nucleus contained heavy, positively charged particles called "protons" and that there were approximately twice as many protons in the nucleus as there were electrons in the external orbits (see Fig. 2.4). Since the atom is electrically neutral, it was also assumed that there were sufficient electrons contained in the nucleus to make the sum of the negative charges in the atom equal the total positive charge.

Experiments carried on at the Curie Institute in Paris and in various European universities led to the discovery of a particle with no electric charge and with mass equal to that of the proton. This particle, now called the "neutron," removed the necessity for assuming that electrons exist in the nucleus. It is now known that the nucleus contains a number of protons equal to the number of orbital electrons, and the rest of the nuclear mass consists of neutrons. Since all atoms are assumed to be made up of various combinations of these three fundamental particles, we now have available a systematic method for their classification. If we assume a unit of mass equal to the weight of the proton

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*Notes taken by C. Scott, edited by R. S. Thackeray.

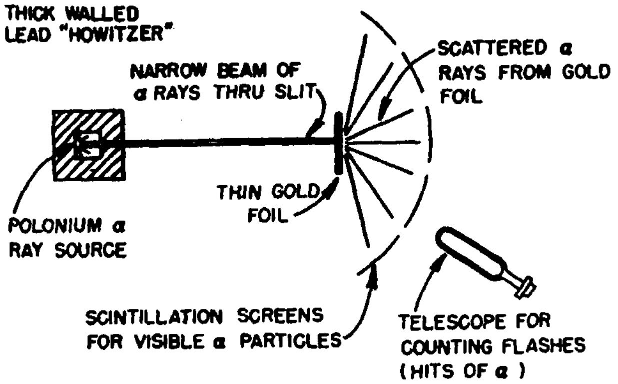

Fig. 2.1—Rutherford experiment.

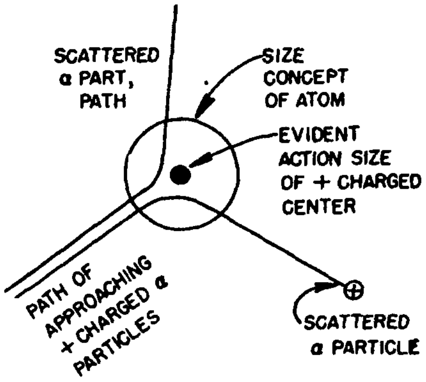

Fig. 2.2—Assumed picture of scattering behavior from % distribution.

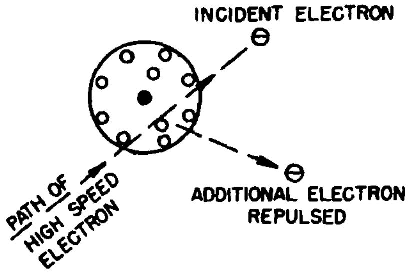

Fig. 2.3—Evidence of electrons as components of atoms.

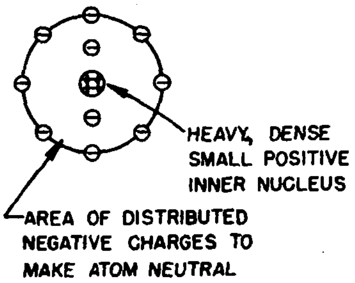

Fig. 2.4

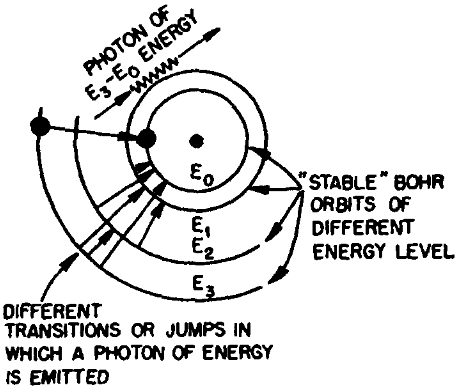

Fig. 2.5

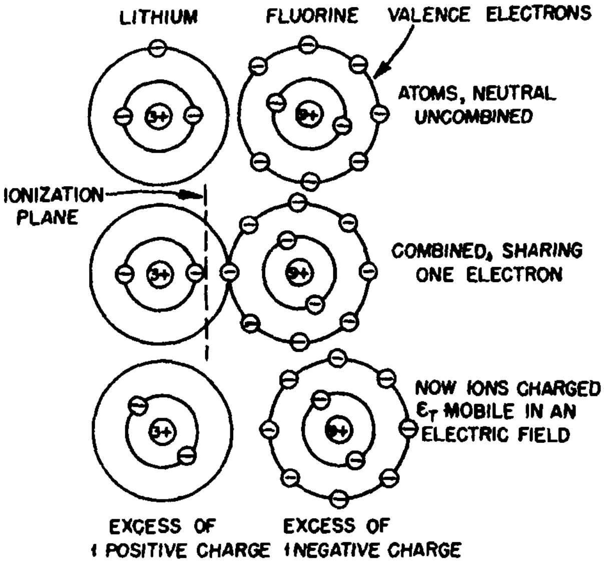

Fig. 2.6

and neglect the comparatively small mass of the electron, the mass of the atom as a whole becomes equal to the number of particles in the nucleus. Thus the atomic weight of the normal helium atom is 4, since the helium nucleus contains 2 protons and 2 neutrons. The number of protons in the nucleus, which is equal to the number of orbital electrons in the atom, determines the so-called "atomic number," which is usually designated by "Z." One fact of interest which has been observed is that quite a range of elements have an atomic weight "A," which is approximately equal to twice the atomic number.

Niels Bohr made a great contribution to our knowledge of orbital electron structure outside the nucleus. Bohr's work was an initial application of Planck's quantum hypothesis. Planck's theory was that energy cannot be divided continuously until arbitrarily small amounts of energy are obtained, but, instead, energy comes in small units which cannot be further subdivided.

Following Rutherford's work it had been assumed that electrons could occupy any point external to the nucleus. This implies that the electron could have any total energy less than a maximum characteristic of the nucleus in question. The quantum hypothesis, of course, prohibits this and states that the electron can have only certain definite energies. It is to be understood that this is a negative energy, since work would have to be done on the electron to move it away from the attraction of the nucleus.

Electrons, which are at a given distance from the nucleus, settle into orbits about the nucleus for which the electrical attraction is balanced by the centrifugal force acting on the electron. The older ideas would have permitted any ciruclar orbit to exist. Bohr's theory stated that only specific orbits are permissible (see Fig. 2.5). A generalization of Bohr's idea made by the German physicist, Sommerfeld, actually allows several elliptical orbits for each of the circular Bohr orbits. In the elliptical orbit the electron has the same total energy, that is, kinetic energy plus potential energy, as does the electron in the equivalent circular orbit.

As we have said, each of the Bohr orbits plus their associated elliptical orbits correspond to a certain total energy for the electron. An electron which reduces its total energy is known to radiate the difference in energy in the form of electromagnetic waves. If for some reason an electron is forced to move from a stable orbit close to the nucleus to one further out, it will then, in a short time, return to its original position. To move the electron outward, an external force is required. This does work upon the particle, giving the additional energy. In returning to its original position, in a lower energy state, the added energy is lost. This appears in the form of light or X rays. Since each orbit has a definite energy associated with it, each of the electron jumps will cause the radiation of a photon of a definite energy. The frequency or "color" of the electromagnetic radiation is determined from the energy of the transition by the expression $E = h\nu$, where E is the energy difference between the initial and final orbits, h is Planck's constant, and ν is the frequency.

The chemical properties of a given atom are entirely determined by the electron structure and are independent of the nucleus. There are preferred numbers of electrons in each of the orbits about the nucleus. Two electrons will fill the first shell, eight the second, and eighteen the third, etc. An outer shell which is nearly filled, but still incomplete, seeks to add electrons. An outer shell which has only one or two electrons seeks to give those electrons away. This is the basis of simple chemical binding in the formation of a molecule (see Fig. 2.6). The atom which desires to give electrons shares them with an atom which is seeking to complete a shell. Electrons which are available for such sharing are known as "valence electrons."

Electrons can be added to or subtracted from normal atoms in other ways; that is, an electron may be knocked out of an orbit, and the atom left in a net positive electrical state, or an atom may complete a shell from a supply of free electrons and become electrically negative. Such charged atoms are known as "ions." Ionization is frequently found where gases have been partly bombarded with radiations or in solutions where chemical molecules have dissociated, with the shared electrons going entirely into nonmetal atoms and leaving the metal atom in a positive state. Such charged atoms will migrate in an electrical field owing to the electrical attraction and repulsion and can be collected as in an electrolytic cell or ionization chamber.

It is obviously quite possible for an atom to retain the same total number of electrons and protons and therefore retain the same chemical characteristics while the mass of the nucleus is changed by adding neutrons. This gives rise to the concept of isotopes in which the atoms of a chemical element can have a variety of atomic weights without changing the nature of the element itself.

2.2 NUCLEAR STRUCTURE

By Ross S. Thackeray*

We shall have to study the nuclear structure without having a convenient picture of the dependence of energy on position such as is available in atomic structure. There is no experimental information about what goes on inside the nucleus. It is possible, however, on the basis of analogy with atomic structure, to construct a theory which helps to explain what is known as the "nucleus."

In the last lecture we studied the idea of a nucleus constructed of positive and neutral particles with negatively charged particles moving in planetary orbits around the heavy core. If we restrict ourselves to the consideration of the Bohr model of the atom, we have an expression for the total energy of an electron moving in an orbit of radius r. This expression is

$$E = \frac{-e^2}{2r}$$

This shows that as you go closer and closer to the nucleus, the negative energy of the electron becomes greater.

Let us now discuss the situation when there are two possible orbits which will be numbered 1 and 2 (see Fig. 2.7). We have then

$$E_1 = \frac{-e^2}{2r_1}$$

and

$$E_2 = \frac{-e^2}{2r_2}$$

where e is the electronic charge and r_2 is greater than r_1. In this situation E_2 is greater than E_1 since E_2 corresponds to a smaller negative energy. In order to transfer an electron from orbit 1 to orbit 2, positive energy must be added. This can be done by bombardment with radiation, excitation in an arc, and other means. An electron which has been lifted from its "home" orbit will be attracted to return by the unlike charge on the nucleus. If it falls back to the state of higher negative energy, the difference in the energy levels E_1 and E_2 will be available for radiation as a photon of electromagnetic energy. This difference in energy is radiated as a photon of which the wavelength or "color" is related to the energy by the equation

$$\Delta E = h\frac{c}{\lambda}$$

in which h is Planck's constant, c is the velocity of light, and λ is the wavelength.

The identity and characteristic of the fundamental particles used in building up heavy nuclei were determined as the combined result of many individual experiments. We can outline the way in which one such characteristic was determined as an illustration of the method of procedure. As a stream of electrons bombards hydrogen gas it will strip electrons from some of the hydrogen nuclei. The charged hydrogen ions resulting can be accelerated electrostatically and run as a beam into a magnetic field. If the charge on the particle, the velocity on the particle, and the strength of the magnetic field are known, the mass of the particle may be calculated from the curvature in its path due to the magnetic field. This process is the basis of the mass spectrometer (Fig. 2.8).

No particle mass has been observed which is a submultiple of the mass of the hydrogen nucleus, nor has any submultiple of its charge been observed. This leads to the conclusion that the hydrogen nucleus is a single fundamental particle, and as such it is known as a "proton." The other fundamen-

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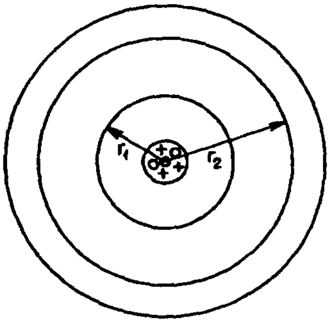
*Notes taken by C. Scott; edited by R. S. Thackeray



$$E_2 = -\frac{e^2}{2r_2} \qquad E_1 = -\frac{e^2}{2r_1}$$

$$E_2 - E_1 = \frac{-e^2 r_1 + e^2 r_2}{2r_2 r_1} = \frac{e^2}{2r_2 r_1}(r_2 - r_1)$$

$$E_2 - E_1 = \Delta E = h\frac{c}{\lambda}$$

Fig. 2.7

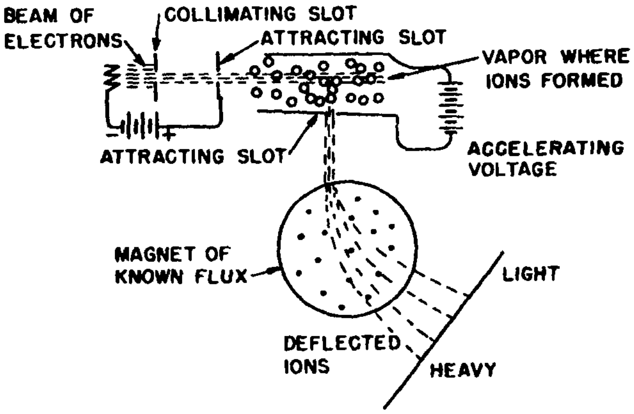


Fig. 2.8

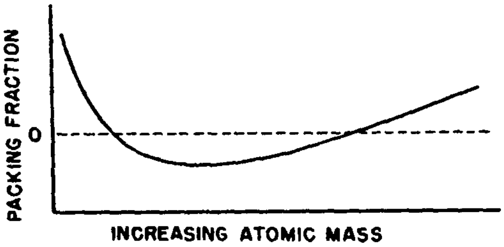


Fig. 2.9

tal particle in the nucleus is the "neutron," whose discovery was announced by James Chadwick of England as the ultimate result of a set of experiments which had been done in many parts of Europe.

When free neutrons were originally observed, it was thought that they were very energetic high penetrating gamma rays. Experiments finally showed them to be particles almost exactly equal in mass to the proton, but with no electrical charge. Protons and neutrons taken together in definite number and frequently varying proportions make up all the nuclei of known chemical elements. There are, however, some arrangements in proportions of these fundamental particles which produce nuclei that are more stable and longer lasting. It is these nuclei which are found in nature.

At first sight, it might seem that the mass of any given nucleus ought to be some integral multiple of the proton mass. However, this is not the case, since in being bound to the nucleus as constituent, the particle loses a small amount of its total mass proportional to the binding energy which holds the particle in the nucleus.

According to Einstein's statement of the equivalence of mass and energy

$$E = mc^2$$

The energy loss experienced by a free particle when it becomes part of the nucleus appears with a change in mass equal to the energy change divided by the quantity c^2, where c is the velocity of light. If the actual mass of a nucleus is subtracted from the mass of an equal number of free protons, a difference obtained is called the "mass defect" and is proportional to the total binding energy holding the nucleus together. If the mass defect is divided by the total number of particles in the nucleus, one obtains a quantity, called the "packing fraction," which is proportional to the binding energy per particle in the nucleus (the approximate curve shape in a packing fraction is shown in Fig. 2.9). Nuclei whose packing fractions have the lowest possible values are, in general, most stable.

It might be well to add a word about the atomic mass unit. This unit is not based on the mass of a free neutron or proton. It is based instead on the atomic weight of the most abundant isotope of oxygen divided by 16. Since, from physical experiments, the mass of the oxygen atom is very slightly less than the mass of 16 individual protons, it may be assumed that there are 16 fundamental particles in an oxygen nucleus, and the atomic weight divided by 16 will represent an average mass of each individual particle within the nucleus. The number obtained in this way will be smaller than the mass of the free proton, because each of the particles in the oxygen nucleus has lost a small fraction of its mass in the form of binding energy.

It is possible for a nucleus to exist in several different states analogous to the various states in which the electrons around the nucleus may be arranged. There is no picture of orbits which can apply here. However, experimental sources state that a nucleus may change from one energy state to another, with the excess energy being emitted as an extremely high-energy electromagnetic photon called a "gamma ray."

2.3 TRANSMUTATION AND NUCLEAR REACTIONS

By R. S. Thackeray

In the last two lectures a great mass of material was presented hurriedly. In order to crystallize agreement on the fundamental points on which great emphasis is to be placed, we shall briefly review some of the points covered.

The characteristics of the fundamental particles about which we have talked are listed in Table 2.1.

The portion of this lecture which was devoted to review also covered a mechanism by which the orbital electrons can radiate photons whose energy is characteristic of the difference in energy between stable electric states. Suppose that a heavy object, on roller bearings to eliminate friction, could be pushed to the top of a hill. One can imagine that such an object would be difficult to move up the incline and that its velocity would never become very great. If the object's velocity was always close to zero, it would never have a great momentum and could not do appreciable work on any obstruction which it might strike. All the effort that went into moving this heavy load would disappear in the raising of the body to a greater height. However, if the body were allowed to descend under the force of gravity in a frictionless slide down the hill, it would acquire a great velocity and would do work against any obstruction in its path.

Table 2.1

Name of particle	Electrical charge, esu	Mass, g	Mass, amu
Proton	$+4.8 \times 10^{-10}$	1.67×10^{-24}	1
Neutron	0	1.67×10^{-24}	1
Electron	-4.8×10^{-10}	9.1×10^{-28}	$\frac{1}{1840}$
Positron	$+4.8 \times 10^{-10}$	9.1×10^{-28}	$\frac{1}{1840}$
Neutrino	0	~0	~0

A similar situation prevails in the orbital structure of the atom. If an electron in an inner shell is lifted to an outer shell by energy supplied by a photon or another electron, that energy disappears in the work of moving the electron against the attraction of the nucleus. If, however, the electron goes back to its original position, the energy used to raise it to an outer orbit becomes available and is radiated as an electromagnetic photon of energy (see Fig. 2.10 for an interaction diagram for photons).

Other concepts which were reviewed were the definition of atomic mass, atomic number, and isotopes.

There was a brief discussion of nuclear reactions at the end of this period. Various methods of obtaining high-speed subatomic particles were introduced, but the student is referred to reference texts and literature for the mechanisms and operating characteristics of these methods.

The nucleus is held together by amazingly great forces, and it requires very energetic bombardment to disrupt the nucleus. The ionization potential for the various electrons of a normal atom requires energy on the order of 10 to 15 ev. To cause the same sort of disturbance in the nucleus, it would require energy on the order of several million electron volts.

There are a variety of common nuclear reactions, but the most frequently used in work with piles are those in which neutrons are absorbed in an atomic nucleus and some secondary particles are emitted. One can characterize such exchanges as n,α; n,2n; n,χ; and n,p. This symbolism means that a neutron is absorbed by the target nucleus and that alpha particles, neutrons, gamma rays, or protons are ejected. In any case in which the ejected particle carries a charge, it changes the atom from one element to another. This process is called "transmutation."

Another concept which is frequently met with in our work is that of cross section. A cross section can be given for a single process or a combination of processes, and it signifies the relative probability of the occurrence of the process in question. Thus if we write sigma (σ) as the symbol for the cross section, a high value of σ_s indicates that the material with this cross section will scatter neutrons from their orbital paths very efficiently. Similarly, a low value for σ_a indicates that few of the neutrons entering a thin slab of the appropriate material will be absorbed in the material and that most of them will escape. σ_t corresponds to the combined probability of scattering and absorption and indicates the total loss of neutrons from a beam due to the insertion of an elementary thickness of material.

2.4 CHAIN REACTION AND THE PILE

By W. H. Ray

The chain letter was given as an example of chain reaction. The fundamental basis of chain reaction is the release of several units of something, each of which may stimulate or trigger a repeat of that same action.

In chemistry, combustion is an example of chain reaction. In the example of a match, there are chemical compounds which will unite with each other and liberate heat, but not unless heated. You scratch the match to start the chain reaction with heat produced by friction. The heat released by the chemicals uniting then ignites the match stick, and combustion will continue, heat releasing more heat, until the chain reaction is stopped for lack of fuel or because the heat is carried away faster than it is generated, such as when you blow relatively cold air past the match to extinguish it.

When a uranium atom is split there result two atoms in addition to the neutrons released. The sum of the mass of these two atoms plus the neutrons will be less than that of the uranium atom from which they came. The loss in mass represents energy which we observe as the kinetic energy of the particles. The majority of atoms do not fission into equal fragments; therefore there results a broad spectrum of fission-product elements or isotopes (see Fig. 2.11 for curve of the abundance distribution of fission-product elements).

Nuclear chain reaction may take place where we have atoms (or, rather, nuclei) which may crack apart and in so doing eject more than one neutron each (on the average there is more than one neutron per fission). Each neutron has a chance of creating an additional fission if it is captured by an atom that will split open (Fig. 2.12 shows a symbolic chain reaction and neutron multiplication).

There is quite a problem of utilizing the neutrons that are ejected, that is, of getting fissionable nuclei to capture the neutrons so that they can split open and release more neutrons. In general, nuclei accept neutrons more readily if they are not going by too fast. Neutrons, without electrical charge, can come closer to a nucleus than ions, and the longer a neutron remains in the vicinity of a nucleus, the more likely it is to be captured. If a neutron is going by rapidly, it is less likely to be accepted.

Neutrons react with nuclei along their trajectory in several ways (Fig. 2.13): (1) pass by nucleus, (2) elastic collision, (3) inelastic collision (energy loss by neutron), and (4) capture (may fission or may be radioactive).

If a neutron recoils from a heavy nucleus, there is not much loss in energy, but if it collides with a light nucleus, the kinetic energy may be shared. Sometimes the neutron is captured by a nucleus and transmutation results. In special cases a fission results. Usually the new nucleus is unstable and adjusts itself by an energy loss that is often accompanied by an ejection of a particle. Where this is delayed, we call the new nucleus "radioactive." A rare case would be where a nucleus accommodates an extra neutron without some further adjustment.

The likelihood that a bombarding particle will have a certain reaction with a target substance may be represented by what is called the "cross section" of that particular target for that particular reaction with the specified particle at a specific kinetic energy (see Fig. 2.14).

Particles incident normally on a unit area of target will react (in the specific manner being considered) as if only a fraction of the area were occupied by target nuclei. The unit of cross section is termed a "barn" and is 10^{-24} cm^2.

To enhance the capture cross section of fissionable U^{235}, means for slowing down the neutrons of high kinetic energy are employed. A material called a "moderator" is introduced. Hydrogen is ideal as far as mass is concerned, but it has an appreciable capture cross section for neutrons. Deuterium, helium, beryllium, and carbon are acceptable moderators because of their low capture cross sections, but the energy dissipated per collision decreases with the increasing mass.

If there are too few fissionable nuclei within mutual proximity, not enough fissioning captures will result relative to the escape of neutrons from the assembly of fissionable atoms to sustain a chain reaction. This critical mass required to permit a chain reaction can be reduced if means are instituted to reduce the escape of neutrons. Enclosing the material within a reflector, usually a wall consisting of moderator free from impurities, will return an appreciable fraction of the neutrons that otherwise would escape.

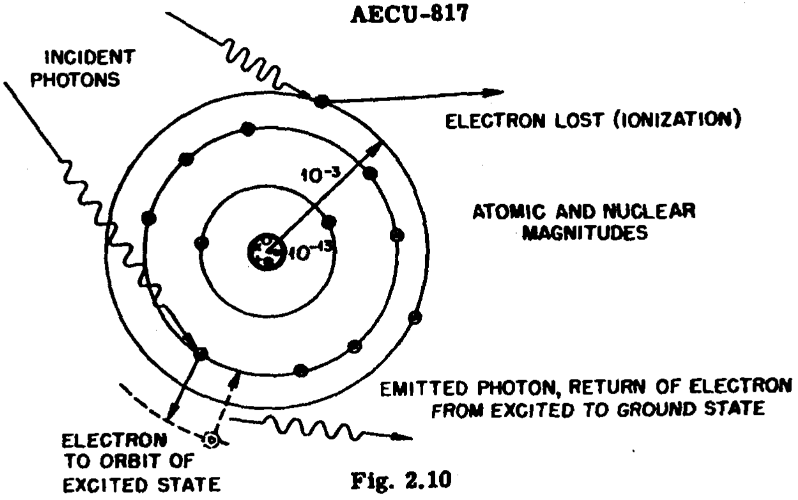


Fig. 2.10

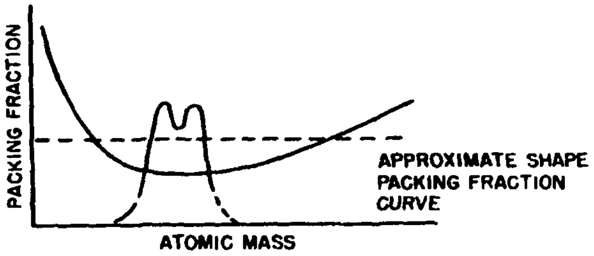


Fig. 2.11

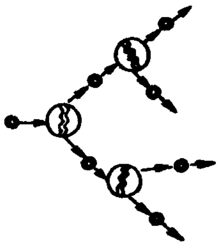


Fig. 2.12

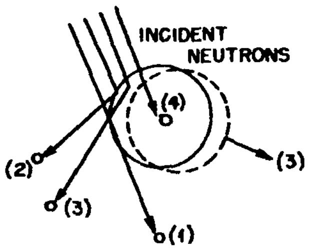


Fig. 2.13

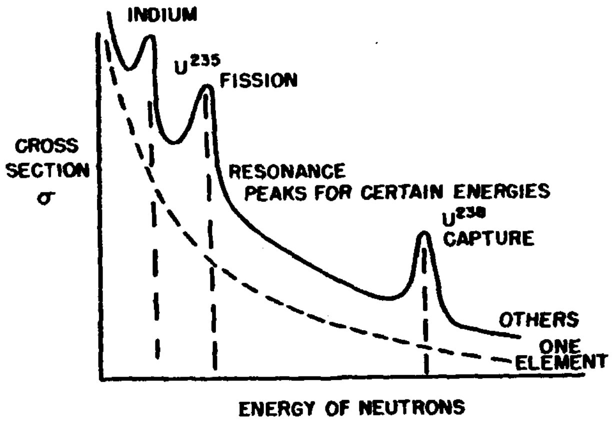


Fig. 2.14

Impurities that have appreciable capture cross section must be excluded if sustained reaction is to take place. The presence of nonfissioning U^{238} makes it necessary to be conservative with neutrons if natural uranium is to be made self-sustaining in a chain reaction. Uranium isotopically enriched in U^{235} can be made to react with remarkably less mass, inferior moderators, and with appreciable foreign materials present. The fission products that form, however, limit the reactivity of an assembly as it operates. Some of these newly generated elements have relatively high capture cross sections and, as they build up, tend to poison the reaction by robbing it of neutrons.

In order to maintain a nuclear chain reaction indefinitely we must not only replenish the "fuel" but remove the "ashes."

The ability of any assembly of fissionable material to maintain a chain reaction may be expressed as its reactivity represented by its "k" factor. The k factor is the ratio

$$\frac{\text{Number of fissions resulting}}{\text{Number of fissions supplying neutrons}}$$

Neutrons from a source may be multiplied by an assembly with $k < 1$, but when $k = 1$, a self-maintained chain reaction exists.

A pile is controlled by inserting poisons of high capture cross section such as boron or cadmium so that the reactivity may be held to 1 or reduced to less than 1 to stop the reaction ($k = 1$ at any steady power operating level, $k > 1$ paves the way for an increasing rate of reaction).

Reactivity has a temperature coefficient which we hope will protect pile operations against a catastrophe, although we do not depend on it. How temperature could affect nuclear reactions may be postulated by considering that the increased kinetic activity of moderator atoms at elevated temperatures relative to that of the massive fissionable atoms hinders the moderator's function of slowing down neutrons by recoil.

Control of a pile is feasible because an appreciable fraction of the fission neutrons is released after a momentary delay. Should the k of an assembly be raised to a point where reaction without the delayed neutrons would take place, control would be lost. We can imagine a critical assembly abundant enough in neutrons to sustain fission captures without the moderation and thrifty conservation of neutrons described above.

A pile is useful because it has an abundance of neutrons. Incidental to the reaction there is a considerable neutron flux over a generous range of energies. This is useful in bombardment of other atoms to produce transmutations. It made practical the production of plutonium by converting U^{238} to U^{239} which decays in two rather rapid stages to Pu^{239}. Similar transmutations are the heart of our current radioisotope industry.

The nuclear chain reaction releases energy which is dissipated in absorbing materials and thus converted to heat. Present proposals to utilize this atomic energy postulate the use of heat engines. In the future we may realize more direct application of the kinetic energy imparted to the particles resulting from fission.

2.5 RADIOACTIVITY DEFINED

By W. H. Ray

The discovery of radioactivity in 1896 by the Frenchman, Becquerel, is a classic example of a finding incidental to an objective investigation which is of greater significance than the objective. In seeking a material that would fluorescent readily when excited with X rays (discovered the previous year). Becquerel was checking some salts of uranium which he had previously observed would fluoresce under ultraviolet irradiation. Pausing in his work, he stowed the salts in the same drawer with some photographic material. When he later developed the emulsions which had been wrapped in light tight paper enclosures, he was surprised to find that they had been affected. Thinking this penetrating radiation resulted from previous excitation of the salts, Becquerel ran a series of experiments to check the response quantitatively. He not only found that salts prepared in darkness had the same property, but that this radioactive characteristic was associated always with uranium.

The romance of the investigation, the discovery of additional radioactive substances, and the development of our understanding about the various radiations make stimulating reading. It is recommended that the student become familiar with these recent activities of modern science. Here we shall attempt only to review briefly the knowledge resulting from this interesting work.

Radioacitvity is spontaneous transmutation. Some nuclei, whether found in nature or artificially prepared, are unstable and transmute themselves into nuclei of different composition (always with less energy). In some instances the new arrangement is even less stable than the initial, and several transmutations may follow successively before a stable nucleus that will remain permanently results.

We know of nothing that will affect the stability of a particular nuclear arrangement except bombardment to transmute it artificially. This transmutation (or disintegration) involving the loss of energy does not occur gradually, but all at once, for any particular nucleus. Among a number of identical radioactive nuclei, disintegration events will occur strictly at random; therefore computations involve a probability that a nucleus or a group of nuclei may disintegrate in a specified interval of time and must be treated by statistical methods. Such mathematics is scheduled to be treated in a subsequent lecture. The probability that a specific percentage of any number of a particular kind of nuclei will disintegrate in a unit of time is solely determined by the kind of nuclei they are. For some kinds, 50 per cent may be transmuted in a fraction of a second, other kinds may have half-lives as long as 10^{10} years.

There are several things which may be given off by a nucleus in radioactive transmutation. Energy may be lost by quanta of electromagnetic radiation. More than 1 quantum may be released per disintegration, but these have distinct energy values characteristic of the disintegration of the kind of nucleus transmuting.

Nuclei of elements having atomic numbers greater than 82 may eject the equivalent of a nucleus of an helium atom, the alpha particle. The kinetic energy of such a particle is characteristic of the disintegration involved. If quanta of gamma radiation accompany the alpha, the energy is relatively low. The kinetic energy of the alpha particle is several Mev. Such a transmutation lowers the atomic number of the nucleus by two and reduces the mass by four units.

Gamma quanta are most commonly found accompanying radioactive transmutations which eject an electron (or beta particle) from the nucleus. (Some nuclei of atomic number less than 82 emit a positron rather than an electron.) However, there are a few radioactive atoms that release a beta and no gamma. Beta transmutations increase and positron transmutations decrease the atomic number of the nucleus by 1 with slight change in mass. The mass of the unit electrical charge e is equivalent to about 1/2 Mev. The kinetic energy of these ejected electrons has a maximum value characteristic of the nucleus from whence the electron came. However, most identical disintegrations supply less than this maximum energy, since the distribution of kinetic energy is over a broad spectrum extending to zero.

Because of the apparent identity of all daughters of such a transmutation, it is believed that disintegrations imparting less than the characteristic maximum kinetic energy to the ejected unit charge dissipate the balance of the energy some other way during the disintegration. The neutrino is postulated to carry away this energy. The ascribed properties of no mass and no charge render it undetectable. Therefore some investigators claim that it does not exist. At present we are not concerned regarding its effect on health because of its inability to produce ions.

Protons and/or neutrons are not emitted in radioactive decay. They may be released only during or immediately subsequent to nuclear bombardment.

Except for potassium and rubidium, radioactive elements occur (to an appreciable extent) in nature only above atomic number 80 (all elements greater than 82 are radioactive). There are three long chains of natural radioactive isotopes terminating in stable isotopes of lead. Isotopes in one chain are never descendants from another chain. Since all transmutations (radioactive) involving a change of mass are in increments of four atomic mass units, the nuclei of all members of a chain differ in mass only by integral multiples of the mass of an alpha particle. This suggests that the helium nucleus is a sort of prefabricated unit in nuclear structure. On this basis, a fourth chain of radioactive descendants might be expected. Nuclear bombardment can effect transmutation from one chain to another and hence into the chain absent in nature. By this means many of the decay schemes of this missing chain have been demonstrated. In addition, the chains have been extended by transmutation to elements of higher atomic number than the 92 maximum found in nature. In Table 2.2 the three chains found in nature plus a new one called the "Neptunium Chain," the missing (4n + 1) series.

Table 2.2

Chain	Name	Prime ancestor	Terminal isotope
4n + 0	Thorium	${}_{90}Th^{232}$	${}_{82}Pb^{208}$
4n + 1	Missing		${}_{83}Bi^{209}$
4n + 2	Uranium	${}_{92}U^{238}$	${}_{82}Pb^{206}$
4n + 3	Actinium	${}_{92}U^{235}$	${}_{82}Pb^{207}$

2.6 HOW TO COMPUTE HALF-LIFE AND HALF-THICKNESS

By T. H. J. Burnett

"Half-life" is the length of time (in seconds, hours, days, or years) required for one-half the amount of radioactive material being considered to disintegrate (or "decay"). This is a constant for each specific isotope, and this decay rate is unaffected by the amount of isotope present or by any other factor.

"Half-thickness" is the thickness of some material which will (by interaction with the radiation) reduce the intensity of incident radiation to one-half.

Mathematical relations which enable us to compute these values are formulas:

$I = I_0 e^{-\lambda t}$ for half-life and decay calculations
λ = decay constant (given in units of reciprocal time, as "per hour," etc.)
t = time elapsed from initial time at which
I_0 = original intensity

$I = I_0 e^{-\mu x}$ for half-thickness and absorption calculations
μ = absorption coefficient (given in units of reciprocal distance, as "per centimeter," etc.)
x = thickness of material which will cut down
I_0 = incident intensity
I (in both cases) = resultant activity or intensity

This type of mathematical relation is called a "function" where for specific λ or μ the intensities are a function of time t or thickness x. A function may be defined by a formula, by a table of values, by a graph, etc.

A graphical method of solving these types of problems was presented in principle and by examples.

Brief derivations of these relationships might be as follows:

Consider λ as the probability of disintegration in unit time of the atoms present where N is the number of such atoms.

Then λN is the fraction decaying per unit time, which may also be considered as the rate of decrease of N, or dN/dt, per unit time.

Therefore

$$\frac{dN}{dt} = -\lambda N$$

a differential equation of which a solution is

$$N = Ae^{-\lambda t}$$

Putting in boundary conditions $N_0 = N$ at $t = 0$, A is found equal to N_0, from which

$$N = N_0 e^{-\lambda t}$$

or using I for the amount of radioactive material present instead of N atoms

$$I = I_0 e^{-\lambda t}$$

If the mean free path of a radiation in a material is l and a very thin slice Δx is taken from a slab x thick, then the probability of interaction (and removal) in one slice Δx is $\Delta x/l$ with the probability of transmission,

$$1 - \frac{\Delta x}{l}$$

If there are n slices in the x thickness of material, $\Delta x = x/n$ and the probability of transmission through n slices is

$$\left(1 - \frac{x}{nl}\right)^n$$

Carrying this to a limit as $n \to \infty$ and $\Delta x \to 0$, we get the over-all probability as: e^{-x}/l which, using μ in place of $1/l$ and an incident intensity I_0, gives

$$I = I_0 e^{-\mu x}$$

Repeated practice in using the simple graphical methods of solution is urged.

2.7 FUNDAMENTAL PARTICLES AND UNITS OF MEASUREMENT

(Activity Evaluation)

By Ross S. Thackeray

In previous lectures we have covered the fundamental particles and their arrangement in atomic structure. Others have spoken of the way in which atoms disintegrate in natural activity. We shall now discuss radiations from active atoms, and we shall find that all the particulate radiations are actually free fundamental particles. There are a number of types of radiation which we shall take up in order.

Alpha particles (or stripped helium atoms)
Neutrons
Protons
Beta rays (or electrons)
Gamma rays (or x rays)

The alpha particles are helium nuclei consisting of two protons and two neutrons bound tightly together. The external electrons, normally associated with helium atoms, are stripped away in a process called "double ionization."

Neutrons can occur as radiation and have the same properties as when bound in nuclei, that is, they have no charge, and they have a mass equal to 1 amu.

Protons or stripped hydrogen atoms also appear as radiation having one plus charge and a mass of 1 amu.

Beta rays are nothing more than rapidly moving electrons freed from their orbits and streaming through space.

Gamma and X rays are electromagnetic photons of very high energy. They are completely identical, and the difference in name only signifies that gamma rays arise from the changes in the nuclear energy levels, while X rays are caused by similar shifts in electron energies.

It may happen that a gamma ray making its way through the orbital shells of the atom will strike an electron and eject it as a fast-moving particle. The energy of the gamma photons will be given to the electron as kinetic energy which will be equal to the energy of the photon minus the ionization potential. This process is called "internal conversion."

The process of beta decay liberates a high-speed electron, but the mechanism is something entirely different. In a beta-active atom a neutron breaks up and gives a proton plus an electron. The electron is given considerable energy and moves rapidly away from the nucleus. There is a very slight change, approximately 1/2000 amu, in the mass of the nucleus so that the weight, in atomic mass units, of the nucleus remains essentially unchanged. However, the number of neutrons has decreased by one, and one proton has been added. This raises the atomic number of the atom 1 unit and completely changes its chemical characteristics.

We can pause here to consider the difference in experimental measurements on internal conversion electrons and discuss beta particles. Although the two radiations are completely identical on a particle for particle basis, we find that the beta particles from a given isotope are emitted with a large spread of energies. On the other hand, the internal conversions of electrons all have the energy of the gamma ray which liberated them. Any given isotope will emit only a very few definite energies of gamma radiation, and therefore, all the internal conversion electrons will have similar discrete energies.

Alpha decay is somewhat similar and causes a loss of 4 amu in the nuclear mass. It decreases the atomic number of the material by two units, thus causing a transmutation.

Free neutrons and protons are usually found as the results of neutron bombardment of some target material.

In attempting to specify the strength of a given source of radiation, we usually speak in units of curies. The curie was originally defined as the rate of disintegration of the radon in equilibrium with 1 g of radium. This, however, applied only to a radon source. To generalize this unit, it is now arbitrarily defined as 3.7×10^{10} disintegrations/sec. If it is desired to express the strength of any source in terms of this unit, divide the measured number of disintegrations per second for the source by 3.7×10^{10}. Since the curie is a very large unit of activity, we frequently use submultiples. The millicurie (mc) is c/1000 or, similarly, 10^{-3} c, and the microcurie (μc) is c/1,000,000 or 10^{-6} c.

2.8 INTERACTION OF RADIATION WITH MATTER

By W. H. Ray

Radiations are absorbed by matter by processes which impart energy to the absorber. The cases of concern to Health Physics are those in which ionization is involved in the process.

Beta particles dissipate their kinetic energy by knocking orbital electrons from their respective atoms. An average of 32 ev is required for the formation of each ion pair. When the beta particle has lost its kinetic energy, it penetrates no further.

Alpha particles produce ions similarly, but because of their relatively lower velocity and greater charge they are able to produce more ions per unit length of path.

Photons impart their energy to respective electrons in the absorbing medium which in turn produce ionization in a manner similar to that of beta particles. There are three methods by which a photon may impart its energy to an electron:

1. Photoelectric process
2. Compton scattering
3. Pair production

Each method has a probability of occurring as the photon proceeds through the absorbing medium. Consequently there is a probability that some will penetrate any absorber thickness which may be established. The probability depends on the orbital electron density encountered and the energy of the photon.

Fast neutrons dissipate their energy by collision with nuclei in an absorber. Nuclei of low mass are accelerated upon effective collision to an extent that they produce ions in a manner similar to that of alpha particles. Since the energy shared per collision diminishes as the neutron's kinetic energy is dissipated, an exponential type of absorption is characteristic of this process.

Thermal neutrons have no excess of kinetic energy to dissipate. Energy is released when thermal neutrons are captured by nuclei in the absorbing medium. In addition to photons and/or particles released in this capture, energy may be released subsequently if the new nucleus is radioactive.

UNIT 3. APPLICATION OF PHYSICS TO HEALTH PHYSICS

3.1 THE ROENTGEN AND ITS DERIVATIVES

By Ross S. Thackeray

The first purpose of this discussion is to make clear that the roentgen is not a unit of radiation. It does not describe directly either the number of particles in the beam or their energy. A physicist has little use for this unit since it tells nothing about the rays themselves or what they are likely to do.

The roentgen is a unit of radiation exposure based on the effect of X or γ radiation on the air through which it passes. It is also important to note that the unit applies only to X and γ radiations in air. Any attempt to use the unit for other radiations or for dosages within a medium other than air implies an extension of the present definition.

The roentgen is currently defined as "that quantity of X or γ radiations such that the associated corpuscular emission (secondary electrons) per 0.001293 g of air (which is equal to 1 cc at 0°C and 760 mm Hg pressure) produces in air ions carrying 1 esu of quantity of electricity of either sign."

We can consider the various parts of this definition in the light of what we know radiation can do to the material which absorbs it. X or γ photons of low energy will separate electrons from atoms by the process of photoelectric absorption. In this way the photon disappears, and a high-speed electron is ejected from the atom. This process has caused one ionization, and the quantity of electric charge separated by this process is too small to measure, even when repeated many times in a beam of radiation. However, an amplification of the amount of charge separated occurs because each photoelectron will cause between 10,000 and 100,000 ionizations along a short track. If the multiple ionization by secondary electrons is repeated frequently as in a beam, there is soon a charge available which is easily measured with an electroscope.

As the energy of the radiation is increased, other processes begin to absorb more of the photons. These processes, called "Compton scattering" and "pair production," also give rise to secondary electrons which cause large numbers of ionizations for each ionization caused directly by the photons.

From this it is easy to see that the roentgen unit considers the ionizations caused only by secondary particles ejected from some known volume of air. Their ionized tracks, however, may go outside of the known volume, and it is important that their total ionization be collected wherever it occurs.

Many units have been proposed for comparing exposures within a material other than air and for measuring dosages of radiations other than X or γ. Some of these have been based on the equivalence of the biological effect of a given amount of the other radiations to 1 r of X or γ. This equivalence, however, is not useful, since the effects are too different to compare. The failure of the biological equivalence led to consideration of some sort of physical equivalence.

One roentgen of X or γ rays corresponds to the absorption of 83 ergs of energy by each gram of bombarded air. Dr. H. M. Parker has based his unit, the rep (roentgen equivalent physical), on this fact. The rep is defined as that amount of any ionizing radiation from which tissue will absorb energy to the extent of 83 ergs/g.

It will be noticed that Parker has actually used the energy absorption per gram of air per roentgen for the absorption in tissue. Tissue will absorb energy to the extent of approximately 93 ergs/g/incident roentgen, and therefore the rep actually embodies about a 12 per cent factor of safety.

Another equivalent unit, which bears a closer relation to the roentgen, has recently been proposed independently by Dr. K. Z. Morgan and some British investigators. This unit is called the "acre," or air cavity roentgen equivalent, by Morgan and simply the "J unit" by the British. It makes use of a hypothetical air cavity inserted in the tissue at the point at which it is desired to know the dosage. One acre corresponds to the absorption of 83 ergs of energy per gram of air in the assumed cavity. If the cavity is restricted to a very small size, the radiation going through the cavity, and particularly the secondary electrons, will be equal to what would have penetrated the tissue or other material which the air replaced. While the unit is designed primarily for biological work, it is perfectly possible to define a concrete, uranium, steel, or lead acre without loss of consistency.

With these units, the roentgen, rep, and acre, it is possible to correlate the effects of radiation on both living and inanimate material, either with calculated internal dosages or with external measurements of the exposure. It is also possible to relate the calculated dosage to the external measurement if sufficient information about the radiation involved is available.

3.2 PERMISSIBLE RADIATION EXPOSURE

By Lucille Leibowitz

Permissible radiation exposure is limited by the tolerance dose. The first approach to any discussion of the subject is a strict definition of the terminology.

The term "dose" is referred to as the energy absorbed in the tissue.

The "dosage rate" is the dose per unit time. There was a tendency in the earlier literature to use the words "intensity" and "dosage rate" interchangeably. This is very confusing, since intensity is measurement of energy flux, while dosage rate is the energy absorption per unit time.

The "tolerance dose" is the dose to which the body can be subjected without production of harmful biological effects. The fact that all radiation produces damage cannot be overemphasized. For this reason it is advisable that persons should strive to keep radiation exposure to a minimum and think of maximum allowable rather than tolerance amounts of radiation.

Actually we are exposed continually to background radiation due to cosmic rays and the radioactive content of the air; however, such low levels are not known to be harmful to man, although it does produce certain biological changes in small organisms.

The awareness for the need of a tolerance dose arose from the discovery of severe skin dermatitis among those working with X rays. Becquerel, who carried a tube of radium in his vest pocket for demonstration purposes, developed a reaction of the underlying skin.

In 1902 Rollins attempted to formulate some idea of a tolerance dose. He suggested that if a photographic plate is not fogged in 7 min, the radiation was not of harmful intensity. This dose corresponds to 10 to 20 r/day for soft X rays in modern terminology.

In 1903 damage to the reproductive organs of animals was observed which could be traced to overexposure to radiation. In 1904 the blood-forming organs were recognized as sensitive indicators of radiation damage. The death of Madame Curie is attributed to aplastic anemia, which was undoubtedly caused by radiation damage to the blood system.

The fact that dermatitis was the first known reaction to overexposure led to the concept of a skin erythema dose. This dose was given as the threshold for undesirable skin reactions. The skin, however, varies in sensitivity over the body. It also varies with the individual, so that no satisfactory limit could be set. In addition, dermatitis is a late reaction of radiation, and the more sensitive cells could be severely damaged by a dose which would have no effect on the skin.

In 1915 Russ delivered a paper before the British Roentgen Society in which he encouraged more precaution in the handling of X rays and radioactive materials. His recommendations were ignored during the war when protection was sacrificed for expediency. After the war repercussions occurred in the form of death to many prominent radiation workers in this country and in Europe. The unfavorable publicity warrented action.

The American Roentgen Ray Society formed a committee in 1920 to recommend protective measures. At about the same time, the British X-ray and Radium Protection Committees published their recommendations for radiation safety. The reports were similar, but they dealt with protective materials for use in X-ray and radium laboratories rather than tolerance limits.

In 1925 the International Congress of Radiology in London considered the problem, but no action was taken until 1928 at the Stockholm Meeting where definite proposals were adopted and an International Committee on X-ray and Radium Protection was formed. There was no reference to a tolerance dose, but they stated that the known effects to be guarded against were injuries to the superficial tissues, derangement of internal organs, and changes in the blood.

In 1934 the tolerance dose was suggested as 0.2 r/day. The basis for this figure was established from the opinions of various individuals, who published their concepts during the period from 1925 to 1932. These opinions were not substantiated by experimental evidence, but merely by observing radiation workers who showed no obvious effects. The daily dose to which they were exposed was calculated, and a generous safety factor was applied.

As an outgrowth of the International X-ray and Radium Commission, the United States formed an advisory committee on X-ray and radium protection, which published its first proposals in the Bureau of Standards Handbook. The first tolerance dose set by this committee was 0.2 r/day, which was later changed to 0.1 r/day for X and γ radiation. This limit is widely used today, although it is by no means the final word.

There is no legal status of radiation safety recommendations. The complications of such an arrangement were recognized, but the American Advisory Committee feels that public opinion can be just as effective as word of law. Moreover, by not having a legal status, recommendations could remain flexible and be readily changed to suit changing conditions. Certain states have set up, through their Departments of Labor and Industry, rules to cover the safety of workers engaged in luminous-dial painting, but in no case are the recommendations actually law on the statute books.

The difficulties in establishing a tolerance dose are due to varying biological factors. Different tissues have different reaction to the same dose. Different organs react in different ways. Each species of animals varies in response to radiation damage. Within the species there is a variation in strain; however, this is more a matter of degree than of type. This obviously makes it rather difficult to extrapolate to human applications from animal experiments.

Generally speaking, the less specialized tissues are more vulnerable to radiation than the complex cells. For example, the nervous cells seem fairly radioresistant, while the more primitive cells of the reproductive or lymphatic systems are early indicators of radiation damage. The rapidly growing and active cells tend to be more radiosensitive than cells in the resting stage.

Although the actual mechanism whereby the cell is damaged is not known, the result is attributed to a combination of a direct action on the cell and an indirect action on the environment of the cell. The radiation damage will therefore be dependent on the environmental conditions. If, for example, the cell has a diminished oxygen supply at the time of irradiation, the effects are less pronounced. There is experimental evidence that change in acid-base relationship affecting permeability of membranes can increase radiosensitivity in certain tissues. Physical factors, such as heat, cold, and previous radiation, also affect the response of some cells.

There are two kinds of effects observed, the threshold and the nonthreshold. In the case of a threshold effect, the damage is not measurable by present methods until a certain dose is administered. The skin and blood-forming organs respond to radiation in this manner. For the nonthreshold effect there is no initial dose that must be exceeded before an effect is obtained. This is true of the influence of radiation upon the germ plasm of lower organisms.

The reversibility of radiation effects is important, particularly in occupation exposure. Reversibility implies that the tissue may be returned to its normal state after exposure is discontinued. This faculty depends on the regenerative properties of the tissue. Some tissues, such as the skin and the blood-forming elements, are endowed with a special type of repair mechanism. Other tissues, such as the brain and the kidney, do not have this property, and any repair is by the formation of a scar, which does not take over the function of the original tissue which it replaces.

In order for an effect to be reversible, it must not produce injury beyond the normal capacity for regeneration. The total dose and the total time affect the powers of regeneration. A certain dose may be safe if administered over a long enough period of time; however, if given as an acute dose, it may damage the tissue beyond repair.

The early signs of damage due to external radiation are lassitude, fatigue, and variation in the number of white blood cells. The red blood cell decrease is a late effect of overexposure, leading to fatal anemia.

Internal radiation refers to radiation received by the ingestion or inhalation of radioactive substances. The dose of internal radiations is measured by the concentration of radioactivity in the excreted body fluids, and in the case of inhalation it is measured by the concentration in the expired air. The late effects of continued inhalation of excessive amounts of radon have been realized by the high incidence of lung cancer among workers in the uranium mines of Bohemia.

There have been several attempts to calculate the lung dosage, but unfortunately the validity of these results cannot be checked because the reports on the Schneeberg miners did not include the site of origin of the cancer. However, on the basis of these calculations, the American Bureau of Standards recommended a permissible tolerance concentration of radon as 10^{-14} curie/cc of air.

There are other radioactive gases, such as radon, mesothorium, xenon, argon, krypton, and iodine, which must be considered hazardous. The tolerance value for radioactive iodine in the atmosphere is based on the selective absorption and deposition of iodine in the thyroid gland. Permissible atmospheric concentration has been tentatively placed at 1.0×10^{-13} curie/cc over 24 hr per day. In the case of the noble gases, argon, krypton, and xenon, the hazard is due primarily to beta and gamma radiation from gas outside the body.

Internal radiation from ingestion and deposit of radioactive material in the body was first encountered among radium-dial painters. Although only a few micrograms were retained in the body, this proved to be a lethal dose. The continued bombardment of the bone marrow led to fatal anemia or production of malignant bone tumors. The actual path of radioactive materials in the body is very complex, and, as yet, no conclusive estimates can be made. The experience with the radium poisoning in the luminous-dial industry has led the Bureau of Standards to establish 0.1 μg of radium as the limit which can be deposited in the body without resulting in later damage.

The skin is composed of two layers of tissue, the epidermis which consists of the protective layer and the dermis which carries the nutrient vessels, accessory organs, and nerve supply. The thickness of the epidermis varies over the body, but in general it ranges from 0.07 to 0.12 mm. On the palms of the hands it averages 0.8 mm, while on the soles of the feet it is as thick as 1.4 mm. The hands can take a greater dose than the whole body tolerance.

The results of radiation are produced in the dermis as damage to the capillaries, venules, and arterioles supplying the skin. The mechanism is thought to be identical with the erythema produced by ultraviolet irradiation of sunlight.

The thickness of the epidermis is of importance in relation to the penetrability of alpha and low-energy beta radiation. The alpha particles of uranium have a range in tissue of about 0.5 mm and therefore do not produce any skin injury. When the energy of the alpha particles is greater, such as those from a cyclotron, the penetration is sufficiently great to produce erythema.

It must be remembered that erythema results as an injury from relatively acute doses of radiation. As the time over which the dose is administered is lengthened, the tolerance increases.

The reproductive organs may sustain damage either to the germ plasm or to the cells which carry the germ plasm. To obtain a permanent sterilization of the female ovary requires 400 to 600 r delivered within the ovary. Sterilization in the male is produced by 800 to 1000 r in the testes. However, sterility in the male is reversible, and although there may be temporary sterility at lower levels, fertility is usually regained after a period of time.

In 1927 Muller demonstrated that the mutation rate of the fruit fly could be accelerated by exposure to X rays. Radiation-induced mutations have been found to have characteristics which bear on the practical consideration of radiogenetic changes possibly associated with occupational exposure. The most important of these is that there is a linear relationship between dose and increase in mutation rate. There is no threshold effect—the cumulation of exposure is therefore additive. The magnitude of the effect is independent of the wavelength and dosage rate of exposure. These genetic studies of the fruit fly have led to considerable discussion concerning their application to man in relation to the tolerance dose. Actually, very little is known about genetics in man, and still less is known about the effects which might be produced by exposure to radiation. As yet there is no evidence that the present generations of radiation workers have produced offspring which differ from those of the general population. If experimental findings in the lower organisms are accepted as valid for man, then one can only escape some degree of radiogenetic effect by avoidance of all radiation exposure.

Recent animal experiments have indicated that long-continued radiation shortens the over-all life span of the animals. On the basis of this consideration, plans are now being formulated for reducing the tolerance dose to 0.3 r/week. The fact that radiation is used extensively in medicine must also be considered, and it is suggested that radiation workers inform the doctor of the conditions under which he is working.

The estimation of the dose received by the body is normally founded on the reading at one region. There are certain conditions under which such a reading may be unreliable. In the case of directional radiation, for example, where radiation is aimed at a specific part of the body, the whole body will not receive the same dose. This dose may be large enough so that if the whole body were subjected to it, the damage would be intolerable, yet the area to which radiation is directed may not be affected.

For emergency cases, certain tolerances have been defined to provide for permissible exposure above that of the daily tolerance dose. In the case of fire, for example, individual exposure will be limited to a total dose of 5 r to be acquired at intensities not exceeding 20 r/hr. If the intensity is 10 r/hr, the fireman will be permitted to stay in the area for 1/2 hr. If the intensity is only 2.5 r/hr, he will have an exposure time of 2 hr.

For cases of extreme emergency, not generally anticipated in industrial operations, a man twenty-one years of age, not previously subjected to significant radiation exposure, may receive a single whole-body exposure of 25 rep provided he receives no other significant exposure before the age of forty-five. For persons over forty-five, a single whole-body exposure of 50 rep may be considered permissible provided no other significant exposures are received in a lifetime. If a person received a single whole-body exposure of 25 rep before reaching forty-five, he would be permitted a single whole-body exposure of only 25 rep after the age of forty-five.

The permissible tolerance doses for a 24-hr day used in practice at ORNL at present are listed in Table 3.1.

Table 3.1

Type of radiation*	Roentgen	Rem	Rep
X-ray	0.1	0.1	0.1
Gamma	0.1	0.1	0.1
Beta		0.1	0.1
Fast neutron		0.1	0.02
Thermal neutron		0.1	0.02 – 0.1
Alpha		0.1	0.01

*Considered from standpoint of internal effects.

The maximum permissible level for plutonium in the atmosphere is tentatively set at 5×10^{-10} μg/cc or 3×10^{-11} μc/cc for an 8-hr working day, 6 days per week, for a one-year period. The object is the prevention of the accumulation of a total of more than 1 μg of plutonium in the body during a person's life.

The air-contamination tolerance level for the more hazardous and common beta-gamma radioisotopes, such as iodine, strontium, barium, etc., is considered to be about 10^{-7} μc/cc of air.

Working-time calculations are based on the period over which the dose is administered. If a man has worked in an area where the intensity is 0.1 r/hr, he will be limited to 1 hr of work in that area and will not be permitted to be exposed to any further radiation over the 24-hr period. If the intensity of the area is 0.01 r/hr, the man may be continually exposed over an 8-hr period or even 10 hr if necessary.

When two or more different types of radiation contribute to the dose, the total dose is determined by the combined effect. For example, if a man is in an area in which he is exposed to neutron flux and gamma radiation at the same time, he will be permitted to work for 1 hr if the neutron dose is 0.05 rem/hr and the gamma dose is also 0.05 rem/hr. If the neutron dose was 0.05 rem/hr and the gamma dose was 0.02 rem/hr, he would be able to work approximately 1 hr and 20 min in the area.

Some people have radioactive substances fixed in their bodies. Such people are limited to a dose which is the difference between the tolerance dose and the dose from the amount fixed in the body. A man with a concentration of plutonium in his bones amounting to 0.06 rem/day will be limited to a tolerance dose of 0.04 rem/day from external exposure.

The above discussion applies to total-body exposure. The only tolerance defined for an individual part of the body in industrial workers is a dose of 1.0 rep/week for the hands.

3.3 PERMISSIBLE CONTAMINATION CONCENTRATIONS

By W. H. Ray

Contamination concentration is usually thought of as radioactivity per unit of surface area and should have curies cm^{-2} as a fundamental unit of measure. When specifying scrap material for salvage, curies g^{-1} or curies cm^{-3} may be more significant. Volumetric determinations involve laboratory analysis of samples.

The earth's crust has been estimated to contain 1.3×10^{-12} g of Ra and 1.6×10^{-5} g of Th on the average per gram of matter. This would indicate 3.6×10^{-11} curie/g (or 79 dis/min/g) if all the members of only these two naturally radioactive decay chains were considered, i.e., 46 dis/min alpha plus 33 dis/min beta. It is evident that the concentration varies widely from the above average. Some ores of pitchblende may comprise 84 per cent uranium. Therefore we are ever beset with a background of radioactive contamination above which we must work.

Reasons for limiting radioactive contamination are based on both health and technical objectives. The basis for health considerations is 300 mrem/week. Contamination should be limited to prevent overirradiation to personnel. Control of contamination is also imperative to keep radioactive sources out of the anatomy. The mechanism of tactile transfer of contamination is as impressive as King Midas' touch. To keep things out of the mouth is simple, but to keep an environment comfortably occupiable, immaculate techniques must be instituted. It is reasonable to expect that surface contamination may become air-borne as a result of erosion. Practically no correlation between surface activity and concentration of air-borne contamination has been observed thus far. The possibility of inhaling transient "slugs" of radioactivity lead one to measures of absolute prevention.

Consequently the permissibility of any radioactive contamination is a matter involving professional appraisal of the specific situation considering the above factors. Sometimes quantities have been expressed for specific situations, and these values have been unadvisedly published. Such values are derived from the basic limitation of 300 mrem/week by an appraiser's estimate of the conditions pertaining to a special example and should be considered in this light. As a suggestion they are less reliable to apply to another situation than reconsideration from the basic fundamentals to arrive at an appropriate recommendation.

For the various reasons stated above, contamination concentrations recommended are usually very low, sometimes expressed as none. The determination of small values of randomly occurring events above an appreciable background of also randomly occurring events has practical limitations. An indication of twice background takes no more time to determine than background and is almost certainly positive. As a consequence twice background has been suggested as a permissible contamination limit where the intent was zero. Because background is not a universal value, some specification as to the normal value intended should accompany such a requirement.

Before less than twice the background was suggested, it was evident that contamination limits were usually suggested at levels dependen upon the sensitivity of available instruments. As sensitivities have increased, the trend has been downward; therefore we may expect the twice background level to be the limit being approached.

Technical reasons for limiting radioactive contamination concentration are based upon avoidance of experimental errors due to activity turning up inappropriately. It has been said that the loss of 10^{-3} of the activity from an experiment will foul the results. This is a function of the total radioactivity, but all those interested in sensitive radiation detection and measuring equipment do not want to see radioactive backgrounds increased above the inevitable concentrations in nature.

3.4 PERMISSIBLE INHALATION CONCENTRATIONS

By Forrest Western

Introduction

In a discussion of permissible inhalation concentrations of radioactive materials, it is convenient to classify the material as follows:

A. Gaseous material
  1. Noble gases
  2. Chemically active gases
B. Particulate materials
  1. Insoluble
  2. Soluble

In such a classification it is to be borne in mind that there are wide variations in the chemical activities of gases and in the solubility or insolubility of particulate materials.

Except for the possible case of noble gases which do not rapidly decay into chemically active radioactive products and for *insoluble particulate materials, the basic problem in the determination* of permissible concentration may be broken up into two parts: (1) What concentration of a given radioisotope can be tolerated in a particular tissue or organ of the body? (2) What concentration of a given radioisotope in the atmosphere will give a tolerance concentration of the isotope in the critical tissue of the body? It is the purpose of the following discussion to provide a basis for making quantitative estimates of the answers to these questions, making use of certain simplifying assumptions and appropriate biological information.

Under normal circumstances it is believed that dosages to body tissues from beta-gamma-emitting radioisotopes within the tissues should not exceed those permitted from external sources of radiation when measured in equivalent units.

As defined by the Fifth International Congress of Radiology (Chicago, 1937), 1 r is "the quantity of X or gamma radiation such that the associated corpuscular emission per 0.001293 g of air produces in air ions carrying 1 esu of quantity of electricity of either sign." From its definition, the roentgen is a measure of the amount of energy, per unit of mass, absorbed in air from the radiation. In Appendix A it is shown that the exposure if air to 1 r of radiation results in the absorption by the air of 5.23×10^7 Mev of energy per gram of air, or 83.8 ergs of energy per gram of air.

General Relationships and Units of Measurement

In general, when a quantity of a radioisotope is fixed in the tissues of a particular organ of the body, the rate of dosage to the tissues of that organ depends on the concentration of the isotope in the organ, on the energy per disintegration of atoms of the isotope, and, to a lesser extent, on the composition of the tissue. Customarily, concentrations are measured in curies per gram, energies of disintegration are measured in millions of electron volts (Mev), and dosages are measured in rep.

The dependence of energy absorption upon tissue composition arises in two ways. For the more energetic radiation, loss of energy to the tissue is largely through Compton scattering, for which the loss per electron of tissue is only slightly dependent upon the composition of the material. Because of the large hydrogen content of soft tissues, there are approximately 10 per cent more electrons per gram than in air, and therefore the absorption of energetic radiation in soft tissue is about 10 per cent higher than in air. For the less energetic gamma and X radiation, a considerable fraction of the loss of energy to matter is through photoelectric absorption. Roughly, the loss per electron, in the ranges in which we are most interested, may be considered to be proportional to the fourth powers of the atomic numbers of the atoms to which electrons are attached. Certain of the body tissues, particularly the skin and the bones, contain elements of higher atomic number which result in higher absorption of energy from soft gamma or X radiation than occurs in either air or soft tissue.

Perhaps the variation of energy per gram per roentgen of dosage should be considered as a defect of the roentgen as a unit of dosage. In any event, when considering dosage from radioisotopes distributed through body tissues, it is more convenient and, so far as is known, just as significant to measure dosages in terms of energy absorbed per gram of tissue. To afford a rough comparision between these two methods of expressing dosage, the term "roentgen equivalent physical," customarily referred to as the rep, has been widely used. As commonly used, 1 rep corresponds to the absorption in tissue of the same quantity of energy (83.8 ergs/g) as would be absorbed in air from a dosage of 1 r. (Since this lecture was given the rep has been defined in terms of a larger energy unit. The new rep corresponds more nearly to the energy absorption in tissue when the energy absorption in a small air cavity in the tissue is 83.8 ergs/g. At ORNL the rep is defined in terms of an energy absorption of 95 ergs/g of tissue.)

It should be obvious from the preceding discussion that a tissue dosage of 1 rep is generally only roughly equivalent to a tissue dosage of 1 r. However, a practical approach to the problem is to assume as a working basis that tissue tolerance can be expressed in terms of energy per gram. (This assumption is as well grounded as the assumption that tolerance for penetrating radiation can be expressed in roentgens.) It is customarily assumed, also, that an exposure of 0.1 rep/24-hr day constitutes a tolerance exposure. These assumptions are made in subsequent discussion.

One curie of a radioisotope is defined as a quantity of the isotope such that it disintegrates at the rate of 3.7×10^{10} atoms/sec. This is equivalent to a rate of disintegration of 3.2×10^{15} atoms/24-hr day. If the average energy per disintegration is E Mev, the energy liberated in a 24-hr day by 1 curie of the radioisotope is 3.2×10^{15} E Mev. Since 1 Mev = 1.6×10^{-6} erg, this is equivalent to 5.1×10^{9} E ergs.

Relationship between Concentration of a Radioisotope in Tissue and Dosage Rate

For simplicity, let us consider a tissue of uniform composition, in which the atoms of a radioisotope are uniformly distributed, and of which any linear dimension is large compared to distance in the tissue required for the radiations from the atoms of the isotope to be effectively absorbed. In such a case, except near an edge of the tissue, any small portion of the tissue will absorb just as much energy as the radioactive atoms in that portion radiate (see Appendix B). If the radioisotope is uniformly distributed through the tissue, in most cases this relationship gives a good approximation to the rate of absorption of energy in the tissue. If the tissue has a small linear dimension, the above assumption leads to a computed value for the absorption of radiation which is larger than the actual value.

From the above relationship, the dosage rate resulting from a radioisotope uniformly dispersed through a tissue is computed as follows:

Let C = number of curies of the radioisotope in the tissue
E = energy per disintegration, measured in Mev
W = weight of tissue, measured in grams
R = dosage, measured in rep
t = time, measured in days

Then the total energy radiated per day measured in Mev is 3.2×10^{15} CE, and the energy per gram per day is 3.2×10^{15} CE/W.

Since 5.23 Mev/g is required to give a dosage of 1 rep, the dosage rate in rep per day is

$$\frac{R}{t} = \frac{3.2 \times 10^{15} \text{ CE/W Mev/g/day}}{5.23 \times 10^{7} \text{ Mev/g/rep}}$$

$$= 6.1 \times 10^{7} \text{ CE/W rep/day}$$

Permissible Concentration of a Radioisotope in the Atmosphere for Constant Exposure over a Long Period of Time

If the daily atmospheric concentration of a particular radioisotope is maintained at a value which is roughly constant for a period of time long compared to its effective half-life in the critical tissue of the body, the rate of loss of the isotope from the body will equal the rate at which it is taken in from the atmosphere. We use this relationship as follows:

The rate at which the atoms of a radioisotope disintegrate is proportional to the number present. After chemical elements are fixed in body tissues, the rate of elimination also is generally approximately proportional to the amount present. The rate of radioactive decay of an isotope may be described by giving the fraction of its atoms disintegrating per unit of time or by giving the length of time for half of any quantity of the isotope to disintegrate. The former is called the "disintegration constant" of the radioisotope, and the latter is called the "radioactive half-life."

Let T_r equal the radioactive half-life, in days, and f_r equal the fractional decrease per day

$$f_r = \frac{0.7}{T_r}$$

and T_e equal the elimination half-life, in days, and f_e equal the fraction eliminated per day

$$f_e = \frac{0.7}{T_e}$$

As a result of these two processes, the effective fractional decrease per day is

$$f = f_r + f_e$$

This daily decrease gives rise to an effective half-life of the radioisotope in the tissue,

$$T = \frac{0.7}{f}$$

Since

$$\frac{1}{T} = \frac{f}{0.7} = \frac{f_r + f_e}{0.7}$$

$$\frac{1}{T} = \frac{1}{T_r} + \frac{1}{T_e} \qquad \text{or } T = \frac{T_r\,T_e}{T_r + T_e}$$

If C is the number of curies of the radioisotope in the tissue, the rate of reduction is

$$\frac{dC}{dt} = fC = -\frac{0.7C}{T}$$

If we take the permissible dosage rate for a beta-gamma–emitting radioisotope to be 0.1 rep/day, the number of curies which will give this dosage rate is, from the preceding section,

$$C = \frac{0.1W}{6.1 \times 10^7 E} = 1.64 \times 10^{-9}\,\frac{W}{E}$$

The rate at which this quantity of radioisotope in the tissue is being reduced by a combination of radioactive decay and physiological elimination is

$$f \times 1.64 \times 10^{-9}\,\frac{W}{E} = \frac{0.7}{T} \times \frac{1.64 \times 10^{-9}W}{E} = 1.15 \times 10^{-9}\,\frac{W}{TE}$$

Under the condition of equilibrium assumed at the beginning of this section, this is also the permissible rate of accretion of the radioisotope in the tissue.

To determine the permissible concentration of the radioisotope in the atmosphere, we must know the volume of air breathed per day and the fraction of the inhaled radioisotope which finds its way to and is retained by the critical tissue.

Let F equal the fraction of radioisotope taken into the body (in this case, into the lungs) that reaches and is retained by the tissue (subject, of course, to reduction at the rate of fC per day), V equal the volume of contamination air breathed per day, and Q equal the curies of radioisotope per unit volume of air. Then the rate at which the radioisotope is increasing in the critical tissue, owing to inhalation, is

$$\frac{dC}{dt} = +FQV$$

Since this is equal to the rate of decrease

$$FQV = 1.15 \times 10^{-9}\,\frac{W}{TE}$$

or, the permissible concentration of a beta-gamma-emitting radioisotope in the atmosphere is given by

$$Q = 1.15 \times 10^{-9}\,\frac{W}{FVTE} \qquad \frac{\text{curies}}{\text{unit volume}}$$

(The unit of volume in which the concentration is expressed will be the same as that used to measure V.)

As a numerical example, let us compute the permissible concentration of P^{32} in air, breathed 8 hr/day, 7 days per week. For this isotope the critical tissue is bone, and the following values for the constants involved in the equation are used by K. Z. Morgan:

$W = 12{,}000$ g
$F = 0.09$
$V = 4.3 \times 10^6$ cm^3/8-hr day
$T = 12$ days
$E = 0.5$ Mev

Substituting these values into the equation above,

$$Q = \frac{1.15 \times 10^{-9} \times 1.2 \times 10^4}{0.09 \times 4.3 \times 10^6 \times 13 \times 0.5}\ \text{curies/cm}^3$$

$$= 5.4 \times 10^{-12}\ \text{curie/cm}^3$$

If the air is to be breathed 24 hr/day, obviously the permissible concentration will be only one-third as much. If one wishes to assume a 40-hr week, he may multiply the above result by 7/5.

It is, of course, assumed in the above example that the tissue in question is exposed to negligible radiation from other sources. The assumption of uniform distribution in the tissue should be carefully examined in all cases.

Evaluation of Quantities C and E Appearing in the Above Equations

The principal difficulty involved in making estimates of dosage rates due to radioisotopes within the tissue is that of obtaining reliable values for C and E. It has already been indicated that the above equation is valid only if the material is uniformly distributed through the tissue. In many cases the extent to which this is true is difficult to estimate. Further difficulty arises from inadequate knowledge of rapidity and percentage of absorption into the tissues and of rates of elimination from the tissues. While the chemical behavior of a radioisotope is the same as that of stable isotopes of the same chemical element, it must be kept in mind that one cannot always predict quantitatively the behavior of trace amounts of the element from experiments performed with relatively large amounts.

For many radioisotopes, values of E are only roughly known. Generally the isotopes in which we are interested disintegrate by the emission of a beta particle, frequently followed by the emission of one or more gamma quanta. Sometimes, however, the disintegration scheme is quite complex, and, more frequently than not, it is not definitely known. Examples of the more common cases, together with the method of determining E for each case, are given below.

1. P^{32}: maximum beta energy, 1.69 Mev; no gamma emission. In all beta spectra the energies per disintegration range from zero to the maximum in such a way that the average energy per disintegration is approximately one-third of the maximum energy observed. Since there is no gamma energy to consider in this case

$$E = \frac{1.60}{3} = 0.56 \text{ Mev}$$

2. Na^{24}: maximum beta energy, 1.39 Mev; gamma energies, 2.76 and 1.38 Mev. Measurements by K. Siegbahn, Phys. Rev., 70: 127 (1946), indicate that the emission of each beta particle is followed by 1 gamma quantum of each of the energies given above. Assuming all energy to be absorbed in the tissue

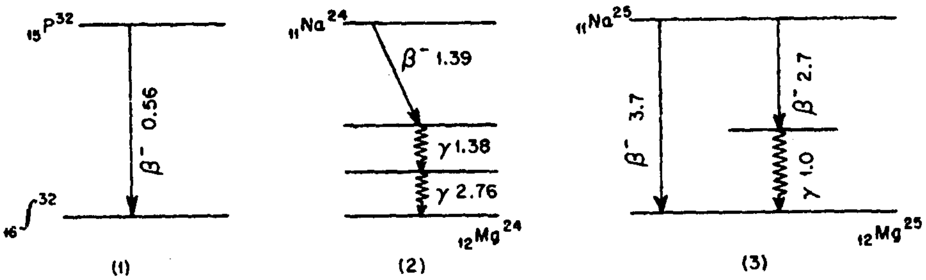
$$E = \frac{1.39}{3} + 2.76 + 1.38 = 4.6 \text{ Mev}$$

In this case our assumption that all the energy is absorbed in the tissue gives too high a value for E, since quanta of these energies are very penetrating.

3. Na^{25}: maximum beta energies, 3.7 and 2.7 Mev; gamma energy, 1.0 Mev. Measurements by Bleuber and Zunti, Helv. Phys. Acta, 19: 421 (1946) and 20: 195 (1947), indicate that 55 per cent of the disintegrations involve the emission of only a beta particle with maximum energy of 3.7 Mev, while the remaining 45 per cent involve a beta with maximum energy of 2.7 Mev followed by a 1.0 Mev gamma.

$$E = 0.55 \times \frac{3.7}{3} + 0.45 \times \left(\frac{2.7}{3} - 1.0\right) = 1.53 \text{ Mev}$$

The disintegrations schemes described above are customarily illustrated by diagrams as given in Fig. 3.1.



(1) (2) (3)

Fig. 3.1

Permissible Concentrations of Alpha-emitting Materials

Where it is believed that the behavior of alpha-emitting radioisotopes in the body is sufficiently well known to use the above method, the calculation is performed in the same manner except that the effect of the alpha radiation is considered to be 10* times as great, per Mev, as that of beta and gamma radiation. This factor may be introduced into the above equations by multiplying the alpha contribution to E by 10. (There is some consideration of using a higher factor than 10, but this has not been accepted.)

Factors Determining Permissible Concentrations

From the above equations it is apparent that permissible concentrations are lowered by small mass of critical tissue, high retention in the tissue, long effective half-life, and high energy of disintegration. In general, permissible concentrations for materials deposited in the bones are low because of the comparatively long effective half-life. It may be observed that, if the elimination half-life and the radioactive half-life are greatly different, the shorter of the two is the principal factor in determining the effective half-life. Thus the effective half-life of 45-day Sr^{89} in bone is 43 days, while that of 25-year Sr^{90} is 197 days.

Practice at Oak Ridge National Laboratory

It is generally impractical to consider individual isotopes in setting maximum permissible limits of concentration in the atmosphere, since the isotopic concentration of the radioactive contamination cannot generally be readily determined. It is the practice, therefore, to set the permissible limits for beta-gamma sufficiently low to take care of the worst cases to be anticipated.

In the case of alpha contamination the frequent occurrence of comparatively large quantities of short-life alpha emitters make it desirable to determine the percentage of long life activity before assessing the hazard. At the present time decay determinations are frequently made for this purpose, but a more rapid method is desirable. The Geiger-Nutall relationship between the range and the rate of decay affords a means for rapid discrimination between long and short half-life alpha emitters by use of a range analyzer.

Noble Gases

Noble gases which do not decay into radioactive materials are less hazardous from the point of view of inhalation than from external radiation.

Particulate Atmospheric Contamination

Hazards from particulate contamination which is highly soluble in the lungs and body fluids may be assessed by the methods given above.

At the present time there is insufficient biological information to assess the hazards due to inhaled insoluble particulate radioactive materials.

Appendix A

Experimental results indicate that the energy absorbed per ion pair produced by X and gamma radiation in air is approximately 32.5 ev. For a dosage of 1 r, the energy absorbed per gram of air would be

$$\frac{1\ \text{esu}}{0.00129\ \text{g}} \times \frac{2.08 \times 10^{9}\ \text{ion pairs}}{\text{esu}} \times \frac{32.5\ \text{ev}}{\text{ion pair}} = 5.23 \times 10^{13}\ \text{ev/g} = 5.23 \times 10^{7}\ \text{Mev/g} = 84\ \text{ergs/g}$$

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*The use of the value of 20 for the relative biological effectiveness of alpha radiation was recommended by the Chalk River Conference on Radiation Protection, September 29-30, 1949.

Appendix B

For a given uniform concentration of the radioisotope in a body of any given shape, the rate at which energy is produced in the body by radiation is proportional to the volume of the body, while the rate at which energy escapes from the body is proportional to the surface of the body. However, for bodies of similar shape the volumes are proportional to the cube of any linear dimension, while the surfaces are proportional to only the square of a linear dimension. Therefore, as the body is made larger, the fraction of radiated energy which escapes becomes smaller; and, by making the body sufficiently large, the fraction of the energy which escapes can be reduced to any negligible value. In such a case essentially all the radiated energy is absorbed in the same volume from which it is radiated. Except for portions of the body very near the surface, any one small element of volume is as favorably situated relative to radiating material as is any other. Therefore each element of volume must absorb radiation at the same rate that radiation is emitted within it.

References

1. K. Z. Morgan, Tolerance Concentrations of Radioactive Substances, J. Phys. & Colloid Chem., 51 (No. 4), July 1947.
2. Koval to Morgan, June 22, 1945: Determination of Particulate Air-borne Long-lived Activity.

3.5 CONTAMINATION AND DUST PROPAGATION

By H. J. McAlduff

Before speaking of radioactive dust propagation and contamination, I should like to review very briefly some of the facts that are common knowledge about nonactive particles or dust.

Humans are living almost constantly in an atmosphere of dust. The times when air is actually clean and free from various types of particles are relatively uncommon. The atmosphere that we might expect in this room might run to the order of several hundred thousand dust particles per cubic feet of air, while other locations might be higher or lower, depending on localized conditions.

When we speak of particles, or dust particles, they can be grouped in certain classification. The first is actual dust which consists of nothing more or less than solid material which has been dispersed to extremely fine sizes, the size classification being considered from the visible range with the unaided eye to less than 1 μ for solid material. These dusts are usually produced by grinding, crushing, and blasting. The size range of fume particles ranges from 0.2 to 1 μ. Fumes are produced by chemical reactions or processes, and examples would be the burning of magnesium and zinc. Smokes are produced by incomplete combustion of carbonaceous material such as oil and coal. The formation of mists and fogs is usually the result of condensation of H_2O vapor on suitable nuclei such as submicroscopic dust particles, and the size of the droplets depends on the conditions under which they are formed. Oak Ridge is an excellent locale for smokes and fogs.

General Properties of Particles

Particulates with which we are concerned in respect to size lie between the region of colloids and those which can be seen readily with the unaided eye. They are larger than those in colloidal systems and yet small enough to exhibit some of their properties and also to have properties not possessed by large masses of the substance from which they are generated.

When solids or liquids break up into finely divided particles, two important changes take place: (1) surface area is greatly enlarged, and (2) space occupied by the dispersed material is increased many times over the volume of the original mass. For instance, 1 cc of material crushed into particles 1 μ^3 in size = 10^{12} particles with a total surface area of 6 m^2. If we assume a dust concentration of 100 million particles/ft^3 of air, this 1 cc of material will be dispersed in an air volume of 10,000 ft^3. In other words, relatively small amounts of material will give large amounts of particulates.

One other factor that is involved, when a solid mass is ground up into fine material, is that the small particles adsorb a layer of air. This condition makes it very difficult to wet the particles. For these particulates to become wetted, the liquid must overcome the gaseous surface area that the small particle has adsorbed. The finer particulates in the extremely fine particle range have the power of Brownian movement. This movement causes, in many cases, a phenomenon called flocculation, in which the smaller particles, by collision, conglomerate to form masses of larger size and settle much more rapidly than fine particles. Flocculation seems to depend on the material. Generally it requires experimentation to determine whether or not a given material will flocculate. Dust formed from substances such as quartz or minerals will not generally flocculate. Certain fumes or smoke such as zinc oxide fume and tobacco smoke will flocculate extremely well.

The fact that a relatively small amount of material will give rise to a large amount of particulate matter is of importance to health physicists due to the fact that when spills occur which involve radioactive materials, the dispersal of that resulting material must be prevented. A method that has been carried out, not in regard to particular spills or localized areas of activity, is vacuum cleaning, which has been used only to the extent of removing dust accumulated over a period of time. It might be well in the future that vacuum cleaning should be carried out more often and in the vicinity of a spill, particularly after it has dried out.

Ventilation requirements in regard to the nonactive materials have consisted in installing ventilation equipment in order to keep the concentration of these materials down to certain allowable levels. These have been determined over a period of years by industrial hygienists and have worked out, for the most part, extremely well. Unfortunately past experience has indicated that ventilation requirements for the purpose of merely keeping within a certain level are not entirely satisfactory when radioactive materials are used. In the ventilation of an industrial process it is a relatively easy matter to capture the contamination and dispose of it. This is not true in the case of radioactive materials. The problem of ventilation merely leads to the problems of what we are going to do with the material when we do capture it and how to capture the material we are actually interested in, in the first place.

The distances traveled by small particles are fantastic at times. Some rough approximations made, assuming particles being discharged from a 200-ft stack, indicate that if a number of factors are neglected, such as terrain and turbulence, a 1-μ particle ejected in a 10 mph wind might possibly travel as far as Memphis or Washington, D. C.

For larger particles the distance traveled is less; a particle 100 μ in size with the same winds prevailing, would probably land within 271 ft from the stack.

The effects of the various particles in the body are varied, and disabilities therefrom may be divided into four classifications:

1. Fibrosis from such dusts as silica and asbestos.
2. Toxic reactions from such dusts as those of lead, arsenic, manganese, phosphorus, and cadmium.
3. Metal fume fever caused by inhalation of such freshly formed metallic oxides as those of zinc and magnesium. It appears that a man working in conditions where he is exposed to those metals builds up a resistance during the week and loses it during the week end. The symptoms are high fever and a feeling similar to coming down with a good case of grippe.
4. Allergies and fungus growths due to dusts of organic or bacterial origin. In many cases the dusts themselves do not produce the growth, but bacteria adhering to those particles caused the trouble.

It is possible that a fifth classification might be added, i.e., radioactive dusts or particulates, but it is necessary to be extremely conservative since so little is actually known about the action of these particles.

In view of the dust conditions to which we are normally exposed and dust conditions to which men in industry are exposed, the efficiency of the body's protective mechanisms, especially in the lungs, is indicated by the fact that during an individual's lifetime the amount of dust actually inhaled and retained is small. The filtering agency of the nostrils, the mucous membrane of the nose and throat, and the lymphatic system of drainage all serve to protect an individual against the inhalation and retention of normal dust concentrations.

When a disability due to dust actually occurs, there are generally three important factors involved: (1) the concentration of material to which the individual has been exposed, (2) the particle size distribution of the material, and (3) the length of time the individual has been exposed to the disabling material.

Medical autopsies have indicated that particles larger than 10 μ are rarely if ever found in the lungs. A perusal of the latest literature discloses that the "critical" particle size range is thought to be from about 0.4 to 2.5 μ. By "critical size" is meant that size range which is most likely to penetrate deep into the lungs and be retained.

The lymphatic system of drainage in the lungs has been mentioned. Augmenting this are phagocytic or scavenger cells that engulf particles which do penetrate deeply into the lungs and remove them to locations where they may be eliminated by absorption and excretion.

It is due to the overloading of these protective mechanisms by particles in the critical size range in large concentrations over a long period of time that disabilities from dust occur.

I am speaking now primarily of fibrotic conditions of the lungs or conditions that affect the lungs themselves. I am not referring to toxic materials which enter the body through the lungs and exert their influence on other organs of the body.

I have only succeeded in hitting the high spots, and in some cases my statements have been very broad in discussing aspects of nonactive particulates. I wish to bring them out, however, since it can be reasonably assumed that in so far as radioactive materials are concerned, their physical properties will follow some of the physical properties of nonactive materials. These should disperse and spread in the form of dust, fumes, smokes, and mists in a similar manner to nonactive particulates and, in general, follow the laws applicable to fine particulates. In other words, conditions would be expected to be similar, except that we have added radioactivity to our particulates. It is, admittedly, a very difficult subject to discuss, since there are so many conflicting opinions. The lack of pertinent biological data has been such that it is impossible to properly evaluate the actual physiological hazard caused by the inhalation of insoluble radioactive particles.

3.6 PERMISSIBLE WASTE-DISPOSAL CONCENTRATIONS

By Forrest Western

Introduction

In Lecture 3.4 a method was given for estimating the maximum permissible quantity of a beta-gamma-emitting radioisotope which, in the absence of other significant radiation exposure to the critical tissue, could be taken into the body. Changing the meaning very slightly of two of the symbols used to apply to any case we may wish to consider,

let QV = permissible daily quantity of a radioisotope taken into the body
W = weight of tissue, measured in grams
F = fraction of isotope which becomes fixed in the critical tissue
T = effective half-life of radioisotope in the critical tissue
E = average energy of radiation per disintegration

an equation derived in Lecture 3.4 becomes

$$QV = 1.15 \times 10^{-9} \frac{W}{FTE} \text{ curie/day}$$

In the quantity QV, V will generally represent a volume or mass of material per day, in which case Q will be the number of curies of the radioisotope per unit of volume or of mass.

This equation or equivalent equations have been frequently used to compute the maximum permissible concentrations of beta-gamma-emitting radioisotopes in drinking water.

While it appears unlikely that a person might spend enough time in or very near water to be concerned over external radiation, for the protection of fish, fish eggs, etc., it may be considered desirable to limit the external radiation dosage rate for bodies submerged in the water to 0.1 rep/24-

hr day. The maximum concentration of a radioisotope in water which could be permitted under this condition is found from an equation given in Lecture 3.4.

$$C = 1.64 \times 10^{-9} \frac{W}{E} \qquad \text{or, } \frac{C}{W} = \frac{1.64 \times 10^{-9}}{E}$$

In this equation C/W is the isotope concentration in curies per gram, and E is the average energy of radiation, measured in Mev.

The concentration represented by the above equation will give a radiation dosage rate of 0.1 rep/day to a small body such as a fish egg, but to a hippopotamus the surface dosage would be only approximately 0.5 rep/day, and the internal dosage would decrease with distance from the surface. It is of interest to note that for some radioisotopes in water the potential radiation hazard due to ingestion greatly exceeds the submersion hazard, while for others, due to very low absorption from the digestive system, the reverse is true.

Possible Effects of Physical and Biological Processes

Using the ORNL liquid-waste–disposal system as an example, it is generally possible to so control the concentration of the waters discharged from White Oak Lake that they fall below the permissible concentrations for drinking water and submersion given above. There have been a few occasions on which these concentrations have been exceeded for short periods of time. Since these concentrations are very greatly diluted in the Clinch River, they are felt to represent no hazard to persons (and fish) downstream. However, in many instances nature has shown remarkable ability to concentrate chemical elements by both biological and physical processes. Many plants and lower animals greatly concentrate certain chemical elements. Such concentrations are sometimes more pronounced if the elements are present in very low quantity, and especially if the organism has a diet deficiency in these or similar chemical elements, conditions which are likely to hold in the case of radioisotopes of some of the rarer elements. The most familiar example of concentration at ORNL is by algae which grow in the Settling Basin and frequently accumulate sufficient activity that a small piece will produce a significant intensity of radiation at a distance of many feet.

Experiments have shown that the clay in this area adsorbs many of our radioactive wastes very strongly. In fact, it is felt that the deposition of radioisotopes in the bed of White Oak Creek and Lake is largely due to this property of the clay which enables it, in settling out, to carry the radioisotopes with it. More effective physical concentration is accomplished by the resin columns that have developed for the almost quantitative removal by ion exchange of both positive and negative ions from solution. The same principle is used in many water-softener plants. The health physicist looks with suspicion upon any uncontrolled concentration of radioactive material, and in this case his misgivings are intensified by the knowledge that such concentrations are frequently sensitive to the degree of acidity of the solution. A solution may permit concentration over a long period of time; then a slight change in pH may release the accumulated radioisotope in a very short time, giving rise to undesirable concentrations.

It is said by some chemists that it is possible for some of the more hazardous radioisotopes, if dumped into a drain pipe, to be almost completely plated or perhaps adsorbed to the interior of the pipe, waiting for some unsuspecting plumber to come along and receive a serious overdosage.

At the present time our knowledge of the extent to which concentrations may take place in nature and the conditions under which they may represent serious hazards is too limited for us to evaluate with any certainty the hazards involved in any method of waste disposal.

UNIT 4. RADIATION-DETECTION METHODS

4.1 IONIZATION MECHANISM

By R. S. Thackeray

Ionization, or the separating of neutral atoms into two charged portions, accompanies the bombardment of any material by the radiations in which we are interested in this laboratory. The ionization associated with uncharged radiations, neutrons or photons, is caused by the secondary electrons which are set into rapid motion by the primary radiation. The radiations which consist directly of charged particles, such as alpha, beta, and proton, can cause ionization as a result of the interaction of primary radiation with the absorbing material and can also give rise to secondary electrons which will produce additional ionization.

The Wilson cloud chamber is a device for making visible the ionized tracks of radiation passing through air. Such a device gives a good picture of the effectiveness of the various radiations in producing ionization. Cloud-chamber photographs show that alpha rays give broad dense tracks, which are filled with many ions per centimeter of path length. Beta particles give tortuous, sparsely populated tracks, which are sometimes so ill defined that they are hard to follow. The tracks of protons have characteristics intermediate between those of the alpha and beta particles.

A concept which is useful in describing the radiations of the ability to ionize is to speak of their specific ionization in a certain medium. Specific ionization is defined as the number of ion pairs formed in a centimeter of track length. In this unit the alpha particle gives a value 1000 times greater than that given by the beta particle, which follows from the fact that an alpha tract is broad and densely populated. There is an empirical relationship in using this concept which has proved to be of some use as a rule of thumb. This rule states that the damage to tissue, which can be done by the various radiations, is proportional to the cubic root of the specific ionization for the appropriate radiation and medium. From this, it follows that an alpha particle with a specific ionization 1000 times greater than that of a beta particle will do damage 10 times greater than that done by a beta bombardment which yields the same energy to the tissue.

Another empirical value, which has proved of great importance, is the value describing the average energy lost by an ionizing particle per ion pair formed. We know that it requires between 10 and 15 ev to ionize a single atom. However, the moving particles lose energy by means other than ionization, and the value for electron volts lost per ion pair formed is obtained by dividing the initial energy of the particle by the total number of ions which it creates. For fast electrons or beta particles, the average energy loss per ion pair formed is 32 1/2 ev. By using this constant we can obtain the total charge to be collected in an ionization chamber simply by multiplying the total energy absorbed in the ion chamber from the incident radiation by

$$\frac{4.8 \times 10^{-10} \text{ esu}}{32.5 \text{ ev}}$$

The actual value of charge collected will, however, be somewhat lower because some of the ions will recombine to give neutral atoms before they can be collected by the ion chamber.

4.2 IONIZATION CHAMBERS

By F. J. Davis

The ionization chamber is a radiation-detection device in which the ionization produced by the radiation is not amplified within the chamber. The dimensions are not critical; some have been built as large as 1 m in diameter and others the size of thimbles. The chamber is essentially a simple instrument composed of an outer container and an insulated central electrode. A voltage is applied to collect ions produced within the chamber.

The currents produced from the ionization are usually quite small. A typical background is 2 ion pairs/cc/sec due to cosmic rays, 2 ion pairs/cc/sec due to stray gamma radiation, and 2 ion pairs/cc/sec due to natural alpha contamination of the walls. The current produced by these 6 ion pairs/cc/sec would then be

$$\frac{6 \times 4.8 \times 10^{-10}}{3 \times 10^{9}} \approx 10^{-18} \text{ amp/cc}$$

The detector may be either a method by which the rate of voltage rise across a condenser whose capacity may be the chamber itself is measured, or it may be a method by which the voltage is measured across a high resistor of the order of from 10^8 to 10^{12} ohms between the collecting electrode and ground. Since the currents are so small, good insulators such as polystyrene, teflon, or fluorothene are necessary. A guard ring should be used in most cases. This eliminates the voltage drop across the insulator between the central collecting electrode and the guard ring, thereby minimizing leakage currents.

For greater sensitivity, larger volumes or higher pressures are used. We have a chamber in the laboratory which uses a pressure of 1500 psi. The usual gas in ionization chambers is air; however, chambers for special purposes use argon, butane, nitrogen, etc.

The voltages applied to the chamber need to be high enough to collect ions before they can recombine. The usual voltage is of the order of several hundred volts. To determine the voltage to be used, a curve of intensity of radiation against ion current should be run to determine linearity. If chamber voltage is not high enough, the curve will not be linear, i.e., the ion currents for high intensities will be too low. This results from the recombination being proportional to the square of the number of ions produced.

The wall of the chamber should be smooth and free of sharp corners. The thickness of the walls is determined by the type of radiation being measured. For gamma radiation the walls should be thick to form large numbers of secondary electrons, which produce the ionization measured, but not thick enough to absorb most of the gamma. The optimum thickness is usually considered as that thickness which will just absorb the secondary electrons that are produced at the outer surface; this is of the order of a few millimeters of aluminum. At atmospheric pressure almost all the ionization is produced by secondary electrons formed in the walls, while at high pressures above 10 atm the secondaries produced in the gas itself become the main component.

For measurement of beta rays, the thinner the wall, the better to eliminate the absorption in the walls. Ionization is produced by the beta rays themselves. To get an idea of the thickness of the wall to be used for measurement of beta rays, the Feather rule is found useful. This rule states that the thickness in g/cm^2 which will just absorb all the beta rays of a certain energy is 0.54 E − 0.13, where E is the energy in Mev. This rule is good for energies above 0.7 Mev. For 1 Mev the thickness would be 0.41 g/cm^2, and the walls of a chamber through which beta rays of this energy are to be measured should be thin compared to 0.41 g/cm^2.

The range of alpha particles from radium is about 10 mg/cm^2, so that if any window is used for the measurement of alpha rays it must be thin compared to the range of the particle. In many cases the alpha source can be placed inside the chamber, thereby eliminating any window absorption.

For the measurement of alpha rays, it is often better to count pulses rather than to measure the ion current. If the ions are collected rapidly on an electrode, an electronic pulse is obtained which can be amplified by a high-gain amplifier. Ion chambers which which count the alpha particles usually use argon or nitrogen in place of air. Oxygen forms heavy ions by combining with electrons, resulting in long pulses of the order of milliseconds. Argon does not combine with the electrons, and since the electrons have a very high mobility, they are collected very fast, producing pulses less than 1-μsec duration. The sharp pulses can be amplified with greater discrimination against noise. The long pulses are of the same order of frequency as microphonics and 60-cycle disturbances; this phenomenon results in a poor signal to noise ratio.

For the counting of low alpha activity the chamber must be free of alpha contamination. Normal brass gives from 2 to 4 alpha ct/hr/100 cm^2. The best treatment is usually a fresh-turned surface untouched by any chemicals or fingers. Farabee's alpha-counting chambers have a background of about 0.1 ct/min. The chambers used here for radon have a background of about 50 ct/hr. These chambers have a wall area of about 1500 cm^2.

Chang and Eng is a special apparatus composed of two ion chambers for the purpose of measuring fast neutrons in the presence of gamma radiation. The two chambers are balanced against each other so that the electrometer registers the difference of ionization in the two chambers. One chamber is coated on the inside with about 1 mm of paraffin and filled with butane to a pressure of about 20 psi. The other chamber is filled with 30 psi of argon. The butane chamber is sensitive to fast neutrons as well as gamma radiation, while the argon chamber is sensitive to only gamma radiation. The apparatus is then adjusted so that in presence of gamma radiation the same ionization is produced in each chamber and thereby shows no indication on the electrometer; in the presence of fast neutrons, however, only one chamber shows additional ionization which is indicated on the electrometer.

For the measurement of slow neutrons, the ionization chamber is usually coated with a layer of boron.

4.3 GEIGER-MUELLER TUBES AND SCALING CIRCUITS

By G. S. Hurst

Geiger counters have been widely used for some twenty years as a very sensitive detector of ionizing particles. The first electrical counter was devised by Geiger and Rutherford in about 1900, when they were making preliminary investigations of the nature of alpha particles. This counter, known as a "point counter," consisted merely of a central electrode, the tip of which was drawn to a very sharp point, and an outer electrode of an unspecified shape. A voltage was applied across the electrodes; thus there was an extremely high electric field at the sharp point.

Geiger and Mueller, in 1928, realized that a much more efficient counter could be made by using a very fine wire as the central electrode. The electric field, being proportional to 1/r, where r is the distance from the center of the wire, is very intense in the near vicinity of the wire and is uniform throughout the length of the counter. Geiger and Mueller further emphasized that the outer electrode should be of cylindrical geometry in order to realize the highest counting efficiency possible. This type of counter bears the name of its first users.

Discharge Mechanism

This part of the discussion will apply to a counter of the type just described, filled with argon (or one of the other inert gases) to a pressure of 6 to 10 cm Hg.

With reference to Fig. 4.1, suppose the pulse size is studied as a function of the voltage across the counter. The alpha curve shows the history of the pulses produced by alpha particles, while the beta curve shows the beta-pulse history. In the ionization-chamber region only those electrons formed by the initial ionizing event are collected on the central wire. When the voltate is increased, the field becomes strong enough to cause the electrons to produce additional ions by collision; thus the pulses become larger, as shown in the proportional region. As the voltage is increased further, the limited proportional region is traversed. Finally the Geiger region is obtained, where all pulses are of the same height, regardless of the magnitude of the initial ionizing event. Notice that the size of the pulse does depend, however, on the voltage.

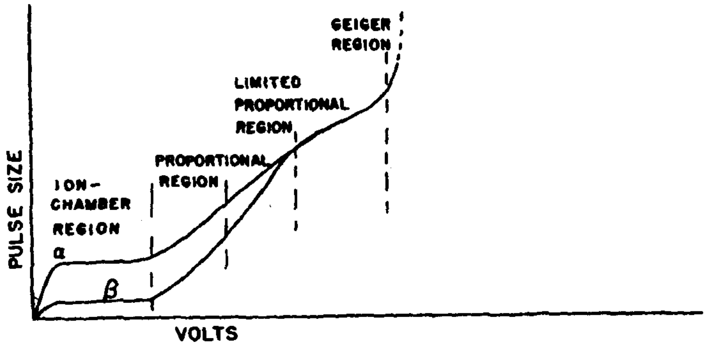


Fig. 4.1—Pulse Size

Dead Time and Recovery Time

After a counter has fired, the electrons are drawn very rapidly to the positive electrode (usually the central wire). The positive ions are drawn much more slowly toward the wall. As a result, the positive ions remain in the volume of the counter as a sheath, more or less well defined. This sheath reduces the electric field between itself and the positive wire to such a value that ionization by collision is impossible. Therefore, if a particle enters the counter during the time when this condition prevails, it will not be recorded. The counter is "dead" and remains in this state until the positive ion sheath has moved out to a certain well-defined radius. At this critical radius pulses begin to occur, but they are not of the normal height. The time required for the sheath to move the rest of the distance to the counter wall has been named the "recovery time." Each of these times is of the order of 100 μsec.

Quenching

When the positive ion sheath reaches the wall of the counter, it might have enough energy to produce photons as a result of the positive charge being neutralized with a negative charge residing on the wall. Furthermore, electrons might be ejected out into the counter volume. These photons or electrons, or both, would then start the ionization process again. Then there would prevail the unhappy circumstances of obtaining several counts for a single initial ionizing event. This condition may be corrected by different methods. The notable ones will be mentioned in the following.

Resistance Quenching

If a high-value resistor is inserted in series with the counter and high voltage, a potential drop will occur across that resistor when current begins to flow. Consequently the potential across the tube will be smaller by the same amount. In this case the positive ions will not have so much energy on reaching the wall as previously. If the value of the resistor is made very large, the ions will not have enough energy to eject the bothersome photons or electrons. Thus the discharge ceases.

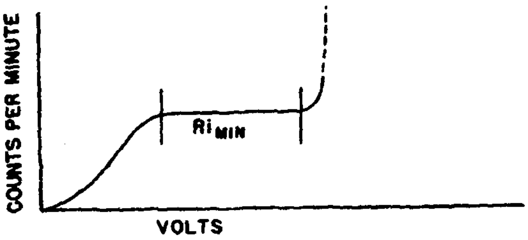


Fig. 4.2—Counts per minute

It is shown in Fig. 4.2 that the plateau (that is, the region in which the counting rate is independent of the voltage) is a linear function of the value of this quenching resistor. In practice it is very desirable to have a large plateau, but this cannot be done with a resistor since the presence of the high resistor presents a serious limitation on the counter from the viewpoint that time constants would become too long for effecting a system that is capable of recording high counting rates.

Neher-Harper Quenching

Various electronic circuits have been designed which will momentarily remove the potential across the counter when current begins to flow. These circuits are very simple and are much more satisfactory than resistance quenches. Famous among these is the Neher-Harper circuit.

Self-Quenching Counters

Today the self-quenching counter is almost universally used because of its simplicity. This counter is constructed in a manner similar to the non-self-quenching counter. The only difference is that a polyatomic molecule is added to the inert gas. A common example is alcohol added to argon (usually 10 parts of argon to 1 part of alcohol, filled to a total pressure of 8 to 10 cm Hg). The function of the polyatomic vapor is to absorb the photons that are emitted during the discharge. This absorption eliminates the need for external quenching.

However, the very generous participation on the part of the polyatomic molecule presents a useful lifetime limitation to the counter. During each discharge a large number of these organic molecules are decomposed. This means that after a certain number of counts the counter will no longer be useful. The counter characteristics change gradually owing to an increased pressure brought about by the broken-down organic molecule.

Indicating Devices

The pulse produced by G-M tubes may be recorded in various ways. The most commonly used methods are the following:

1. For indication only, a G-M tube may be used to fire a thyratron, which in turn operates a set of earphones.
2. For a fairly quantitative measure, a rate meter may be substituted for the earphones.
3. Very accurate records may be obtained by using a scaler in conjunction with a mechanical recorder.

4.4 THE PROPORTIONAL COUNTER
By R. A. Dandl

In any discussion of a gas counter it is desirable to introduce the gas characteristic, and then stress the region one is interested in.

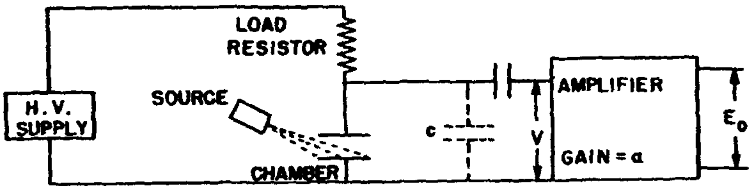


Fig. 4.3

Figure 4.3 is a block diagram of a typical counter circuit. Q_1 is the charge produced in the gas by one particle and is independent of the high voltage, Q_2 is the total charge on the chamber capacity as a result of a particle, c is the sum of the chamber capacity and the input capacity of the amplifier, v is the voltage change across the chamber due to collection of Q_2, a is the gain of the amplifier, and E_0 is the pulse voltage out of the amplifier.

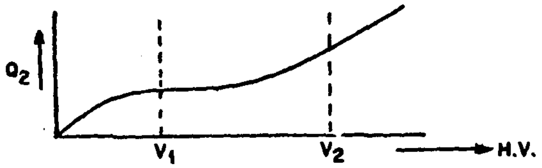


Fig. 4.4

Figure 4.4* is a gas characteristic curve, where Q_2 is a function of the high voltage across the chamber.

When the chamber is operated at voltage V_1, the charge collected, Q_2, is relatively independent of the chamber voltage, and all the ions, Q_1, produced by a particle are collected ($Q_1 = Q_2$). The pulse voltage output

$$V = \frac{Q_2}{c} = \frac{Q_1}{c}$$

Typical values of V range around 0.001 volt. The operation of a chamber in the region of V_1 requires a high-gain amplifier (approximately 300,000) which introduces the probability of stray counts due to mechanical shock and electrical disturbances.

Raising the voltage to V_2 causes the chamber to reach the proportional operating region. This is usually a high voltage for a parallel-plate counter, but the proportional region can be reached. The counter is in the proportional region when the field intensity is so high that the electrons produced by radiation receive enough energy to produce ion pairs.

To obtain a high field intensity requires a lot of voltage on a parallel-plate chamber, but a cylinder with a coaxial fine wire requires much less. This is the usual type.

Proportional Process. A particle forms ion pairs in the gas. Electrons are attracted toward the center wire, and as they get close to the center wire the attraction increases, and the velocity between collisions increases until the electrons acquire enough energy to produce ion pairs.

Q_2, the collected charge, is now different from Q_1, the charge induced by a particle. The ratio Q_2/Q_1 is called the gas amplification factor "A." Now

$$V = \frac{Q_2}{c} \quad \text{but } Q_2 = AQ_1$$

$$V = \frac{AQ_1}{c}$$

Practical gas amplification factors may be 500 or more. The gain of an amplifier is defined as the ratio of the output voltage to the input voltage or

$$\frac{E_0}{v} = a$$

but

$$v = A\frac{Q_1}{c}$$

---

*Reducing chamber pressure moves curve to left.

now

$$a = \frac{E_0c}{AQ_1}$$

It is obvious from the last expression that if one increases the proportional amplification, one reduces the amplifier gain needed and thereby avoids the attendant difficulties of high-gain amplifiers. It should be noticed that a reduction of input capacity also increases the pulse voltage output. Practical amplifiers are usually followed by discriminator circuits which reject low-level disturbances and produce uniform pulses for feeding rate meters or scalers.

Proportional counters have the advantage of speed over Geiger counters and are practically limited by the speed of the amplifiers one can construct.

Some gases used for proportional counters and reasons for their use follow:

Air is used in proportional alpha survey counters to avoid the inconvenience of a sealed tube. These counters can detect alpha path lengths of 1 mm. However, air has a steep proportional characteristic, and air counters require well-regulated power supplies. Since beta ion densities may be of the order of 20 per centimeter as compared to 30,000 per centimeter for alphas, beta proportional counters require considerably more gas amplification than alpha counters. This makes air unfeasible for use in a beta proportional counter. Methane and argon–carbon dioxide (98 per cent A, 2 per cent CO_2) are used for beta proportional counters and quantitative alpha proportional counters* because of their long proportional gas characteristic (long with respect to voltage).

Counter construction and operation hints: Precautions must be taken to minimize spurious discharges in the counter and its associated circuit. These precautions involve the use of good cables and connectors, well-insulated counters, filtered and regulated high-voltage supplies, and smooth, clean counter-electrode surfaces.

The use of some thin membranes as windows on alpha air proportional counters (nylon or rubber) has now become standard practice. The window is of considerable value in avoiding discharges due to acid fumes and moisture in the air.

Rate meters may sometimes be of value to a surveyor, but since there is no way of knowing how the contamination is embedded in a surface, a quantitative reading is usually of little value.

Colloquium

Question: How about the counting rate as a function of voltage?

Answer: With the laboratory-type counter, high counting accuracy requires precautions in the amplifier discriminator circuit to minimize multiple pulses and background pulses. If the gas amplification is too high for a given amplifier, the following bad effects result from overdriving the amplifier: multiple pulsing and differentiation difficulties. There is a plateau associated with the proportional counter, and it occurs when the gas amplification is sufficient to allow one to count all but the smallest pulses produced in the tube. The proportional beta counter circuit is much more difficult than the alpha counter because the spread of pulse heights is considerably greater. The spread of pulses for alpha is 25 to 50, and it is of the order of 500 to 1000 for betas.

Question: Is a plateau similar to that of the plateau in the Geiger region somewhat independent of the voltage?

Answer: Yes if the amplifier has sufficient gain and does not produce multiple pulses due to differentation difficulties or dead spots due to some tube being driven positive.

Question: What about acid in the presence of the open window? And how about beta tolerance?

Answer: You can get a spurious count from it. Beta tolerance of an alpha proportional counter is of the order of 10^7

Question: What about probing too fast?

---

*Methane is not used in a gas-flushed counter where inflammability is a problem, such as in some types of survey work.

Answer: Response time is not a consideration when using an alpha proportional survey counter as it is when using an ionization meter, but if one is looking for small amounts of activity, it is necessary to scan the surface slowly.

UNIT 5. RADIATION-DETECTION TECHNIQUES

5.1 COUNTING TECHNIQUES

By P. E. Brown

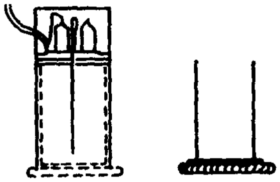
The scope of this talk will necessarily be limited to counting done in the Health Physics counting room and the application of this technique to field conditions.

Several types of samples are submitted to the counting room for counting:

A. Air samples
  1. Alpha counting
    a. Precipitron samples (Al cylinder)
    b. Filtron samples (filter paper)
  2. Beta counting
    a. Precipitron sample (Al cylinder)
    b. Filtron sample (filter paper)
B. Flat disk samples less than 2 in. in diameter, activity one side
  1. Isotope-carrier smears
  2. Floor smears
  3. Miscellaneous
C. Mud, water, biological tissues

The equipment used in this counting is probably familiar to most of you, and some have used all these types of counters. For the benefit of those who have not used them, the following description will be given:

For precipitron samples and filtron samples, both alpha and beta counting are done. The alpha-counting equipment consists of a California scaler (scale of 8) unit connected to an ionization chamber made so as to hold the cylindrical sample foil or filter paper upright, with the electrode running the length of the cylinder and carrying a high voltage (approximately 1250 volts). This diagram shows the sample in place between the two electrodes and single stage of amplification that carries the weak pulses from the chamber through the relatively long shielded cable. The chamber is suspended by rubber supports to prevent jarring. The geometry of this chamber is 50 per cent.

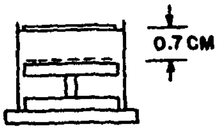


Auxiliary equipment consists of oscilloscope and automatic timer wired to count rate switch and recorder.

The counting of air samples for beta-gamma activity is done with a Higginbotham scaler and a long thin-walled tube (7 in. sensitive area). The tube is mounted in a horizontal lead pig to reduce the background radiation. Auxiliary equipment includes recorder and timer. The geometry of this type is 21 per cent.

Flat disk samples are counted for both alpha and beta activity if desired.

For alpha counting, a (scale of 8) scaler and parallel-plate ionization chamber are used. Auxiliary equipment includes recorder, oscilloscope, and timer.



In the above diagram the plates of the chamber are 0.7 cm apart and at difference of potential 1250 volts. The chamber is suspended by rubber supports to prevent jarring.

For beta counting, a counter, with a mica window of approximately 3 mg/cm^2 thickness, and a Higginbotham scaler are used. A recorder and a timer are auxiliary equipment. The tube is mounted in a vertical lead pig with a series of four shelves below the tube for supporting the sample. A sample counted on the top shelf has a geometry of approximately 30 per cent and on the bottom shelf approximately 3 per cent. The diagram (Fig. 5.1) shows the position of tube in relation to the four shelves.

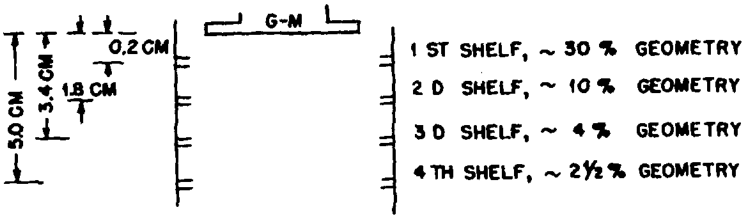


Fig. 5.1

The tube and sample holder are all enclosed by a vertical lead pig for shielding from background radiation. The geometry is as indicated in the diagram and is posted on the door of the shield. Conversion factors are also supplied for converting from other geometrics to 10 per cent. (Smear samples should not have a large amount of loose activity since this will contaminate the counter.)

Mud, water, fish, etc., samples are counted for beta activity only.

Reliability of Counts

Probable Error. In addition to the instrumental and other errors, causes of statistical variations, the accuracy of a count depends on the number of counts taken. The greater the number of counts, the less will be the probable error.

The probable error in the determination of the number of counts per minute from a sample is given by the expression

$$\frac{\sqrt{(\text{PE of BG})^2 + (\text{PE total counts})^2}}{\text{ct/min of sample}}$$

As the ratio of the total counting rate to the background counting rate becomes higher, the contribution of the probable error in determination of the background rate contributes less to the probable error in the determination of the net counting rate. For routine counting where the ratio of the total counting rate to the background rate is greater than 5, it is our practice to approximate the determination of the probable error of counting by use of the expression

$$\frac{0.6745 \sqrt{\text{sum of counts}}}{\text{sum of counts}}$$

$$\frac{9.55}{200} = 0.0477 \text{ or } 5\%$$

$$0.05 \times 200 = 10 \text{ ct/min}$$

Therefore, counting rate = 200 ± 10 ct/min.

If all counts are made for the same length of time when plotting a decay curve or absorption curve, a correction would need to be applied for each count, and the error would be increasing as the curve slopes downward (see Fig. 5.2).

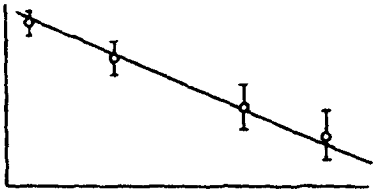


Fig. 5.2

If the per cent of error is specified when the counting requisition is made, all counts will have the same accuracy. This is done by the counting technician by referring to tables which have the ratio on horizontal axis and total number of counts needed to get the required accuracy on the vertical axis. Note on the charts that to get low probable error, long counts are necessary unless the ratio of counts to background is great. This shows why a low background is necessary for accurate counting (Fig. 5.3 is a rough copy of actual charts used).

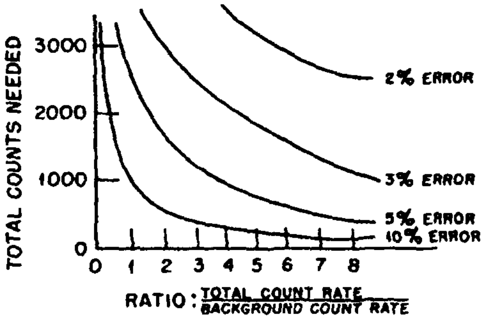


Fig. 5.3

Coincidence Loss. Where probable error is a loss of accuracy due to a very low count, coincidence loss is due to a high count or directly proportional to the counting rate. This is due to the fact that the resolving time of the counter is such that one count might be counted with another, as one. The amount of correction used is 1%/1000 ct/min for all counting in this counting room.

Special Counting Services

Long-lived Alpha and Beta Activity. In the determination of long-lived activity in radioactive dusts and mists, a correction must be made for the presence of radioactive contamination from natural radon and thoron sources. Four hours after collection of an air sample, the radon decay products will have decayed to a negligible quantity (27-min half-life, Ra B, is the controlling half-life). The 10.6-hr thorium B controls the thoron chain decay so a correction for this activity must be made to get the long-lived activity. This correction is applied in the following equation for long-lived activity

$$C_{LL} = \frac{C_2 - C_1\,(e^{-\lambda\Delta t})}{1 - (e^{-\lambda\Delta t})}$$

where C_2 = count taken after 24 hr (t_2)
C_1 = count taken after 4 hr (t_1)
Δt = time $t_2 - t_1$, in hours
λ = decay constant of thorium B; $\lambda = \frac{0.693}{10.6}$
T = ($e^{-\lambda\Delta t}$) (values of T for different values of Δt may be obtained from tables used in counting room)

Therefore

$$C_{LL} = \frac{C_2 - C_1T}{1 - T}$$

Example: $C_1 = 400$ ct/min, $C_2 = 200$ ct/min, $\Delta t = 20$ hr $= 0.271$ (from chart)

$$C_{LL} = \frac{200 - (400 \times 0.271)}{0.729} = \frac{200 - 109}{0.729} = 120 \text{ ct/min}$$

If tables are not available, use table of exponentials in handbook. These counts, C_{LL}, can be then converted to microcuries per sample by the following formula:

$$\mu c/\text{sample} = \frac{\text{ct/min}}{\text{geo.} \times (3.7 \times 10^{10} \times 60) \times 10^{-6}} = \frac{\text{ct/min} \times 10^{-6}}{\text{geo.} \times 2.2}$$

or for beta

$$\text{beta} = 2.17 \times \text{ct/min} \times 10^{-6}\ \mu c/\text{sample}$$

or for alpha

$$\text{alpha} = 0.91 \times \text{ct/min} \times 10^{-6}\ \mu c/\text{sample}$$

The surveyor then converts microcuries per sample to microcuries per cubic centimeter of air.

B. Absorption curves for beta activity are drawn for samples sent in thus enabling one with the aid of appropriate tables to determine the kind of activity present. This is done for alpha and beta activity.

It is conceivable that an emergency might arise when all counting facilities were destroyed or unusable. With only a Victoreen 263 and a few lead bricks an emergency facility for simple counting can be devised. With the tube removed from the stainless-steel shield and placed inside a lead cave, counts may be made on hands, shoes, smears, and even air samples. For the permissible counting rates given below, the G-M tube should be located 3 1/4 in. from the article checked.

Hands or gloves, 700 ct/min
Shoes outside, 10,000 ct/min
Smears or exposed surfaces, 200 ct/min

If the Victoreen 263 is not calibrated in counts per minute, a conversion of 3100 × mr/hr will give an approximate counting rate.

Air samples may be counted for beta activity by inserting the tube inside the cylindrical sample and counting. The tube must, of course, be protected from contamination by a thin paper covering.

The over-all geometry will depend upon the assembly used, but may be in the order of 20 per cent.

5.2 PROBABLE ERROR AND CONFIDENCE INTERVALS

By W. S. Snyder

The basic problem is that of selecting a sample from a population and from a study of the sample inferring some information concerning the parent population.

A sample, even when selected in a manner as free from bias as we can determine, may still be unrepresentative of the parent population. If a ball is drawn at random from an urn containing 99 red balls and 1 black ball, we may get the black ball, and if the experiment is repeated 10 times, we may get the black ball on every draw. The theory of probability only asserts that the eventuality is exceedingly unlikely. However, the same theory also asserts that if the experiment of making the 10 draws is repeated often enough, the eventuality of 10 blacks will certainly occur, and even in a certain percentage of the cases.

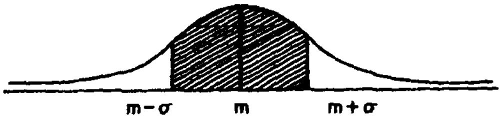
Probability and statistics are principally concerned with making precise statements about the results of a succession of trials and generally cannot assert anything very precise about a single trial. In the above example we assumed the population known. Generally we have to make more or less plausible assumptions about our population, and then we study our sample to substantiate or invalidate our assumption. We might begin by assuming our urn to contain red and black balls in equal proportions. But after 10 draws all yielding black we would certainly have serious doubts concerning this hypothesis. The hypothesis is not vigorously disproved; it is rather exceedingly unlikely.

One of the commonest assumptions concerning a population is that it is normally distributed. This means that the population when classified by a numerical measure of the attribute being studied follows the pattern of the Gaussian or normal error curve. This curve is defined by an equation

$$Y = \frac{1}{\sqrt{2\pi}\sigma} \frac{e^{-(x-m)^2}}{2\sigma^2}$$

containing two parameters, m (the mean) and σ (the standard deviation). To say the population is distributed in accordance with this law is to assert that the fractional part of the population lying between any two limits a and b is equal to the area under the curve between the same limits. The following tabulation lists some of the intervals most often used [see diagram (the shaded area is 68.3% of total area)].

	Area
$m \pm 0.675\sigma$	0.50
$m \pm \sigma$	0.683
$m \pm 2\sigma$	0.954
$m \pm 3\sigma$	0.997



Suppose an individual X is selected from the population in a random manner, i.e., so as not to favor any one individual of the population over another. We say the probability that X would lie between the values m $\pm\sigma$ is 0.683, and we abbreviate this statement thus

$$\text{Probability } (m - \sigma < X < m + \sigma) = 0.683$$

Similarly

$$\text{Probability } (m - 2\sigma < X < m + 2\sigma) = 0.95$$

Nevertheless, if an individual is drawn, he may not lie in the interval m $\pm 2\sigma$, and the above statement does not assert that he will. It simply asserts that this latter event is relatively unlikely, that in a large number of such trials in only about 5 per cent of the cases should this happen. If the percentage of such results continued to deviate considerably from this, we could only conclude that our procedure of selection is really biased or that our population is, perhaps, not distributed in accordance with the above law.

Now consider a population distributed by the above law, and suppose N individuals are drawn and the average taken

$$M = \frac{X_1 + X_2 \cdots\cdots X_N}{N}$$

Statistical theory shows that the values of M obtained when this procedure is repeated indefinitely again follows a normal distribution law with the same mean as for the parent population, but with a standard deviation given by $\sigma/\sqrt{N}$.

If N is large enough, $\sigma/\sqrt{N}$ is very small and since

$$\text{Probability}\left(m - \frac{\sigma}{N} < M < m + \frac{\sigma}{N}\right) = 0.683$$

we can accept M (note that M is determined from the sample) as an approximate value for m and we shall not be in error by more than the small quantity $\sigma/\sqrt{N}$ (except in 31.7 per cent of the cases when we make such an estimate).

If we would like to be correct more than about a third of the time, we can use the fact that

$$\text{Probability}\left(m - \frac{2\sigma}{\sqrt{N}} < M < m + \frac{2\sigma}{\sqrt{N}}\right) = 0.954$$

and now assert that M approximates m with error less than $2\sigma/\sqrt{N}$. We can expect this statement to be wrong in about 5 per cent of the cases. If we use the probable error, we can assert that

$$\text{Probability}\left(m - \frac{0.6745\sigma}{\sqrt{N}} < M < m + \frac{0.6745\sigma}{\sqrt{N}}\right) = 0.5$$

Accordingly we should expect that the average M of the sample will deviate from the true mean m by more than $0.6745\sigma/\sqrt{N}$ as often as not.

The above discussion tells us nothing very precise, since not only the true value of m but also the value of σ would, in practice, be generally unknown. There is, however, good reason to believe that statistical phenomena due to long-lived sources of radiation follow a somewhat different law, the Poisson distribution. The Poisson distribution depends only on a single parameter m which is again the mean. If m is not too small, the Poisson distribution is well approximated by a normal distribution having the same mean m and with standard deviation $\sigma = \sqrt{m}$. This fact enables us to regard M as, in a sense, giving us the approximate value of the mean and the standard deviation needed in the earlier statements.

More precisely, suppose a radiation field has an average rate of m particles per unit of time across the region of the counter. We ignore all deficiencies of the counter and deal here only with statistical fluctuations due to the randomness of the radiation field itself. The number of counts of our "perfect" counter per unit time then constitute a population which is Poisson distributed with mean m. However, we immediately replace this by a normal distribution with mean m and standard deviation $\sigma = \sqrt{m}$. If m is not assumed to be too small, the agreement will be good in general. Then if a count is taken for N units of time (N repetitions of the sampling process) we can assert that for the average rate M computed by M = total count/N

$$\text{Probability}\left(m - \sqrt{\frac{m}{N}} < M < m + \sqrt{\frac{m}{N}}\right) = 0.683$$

$$\text{Probability}\left(m - 2\sqrt{\frac{m}{N}} < M < m + 2\sqrt{\frac{m}{N}}\right) = 0.954, \text{ etc.}$$

Obviously if N is large, the per cent of error

$$\frac{M - m}{m} < \frac{1}{\sqrt{mN}}$$

in 68.3 per cent of the cases and

$$\frac{M - m}{m} < \frac{2}{\sqrt{Nm}}$$

in 95 per cent of such cases, etc.

For example, if the count is taken for 100 sec, with a total count of 22,500 counts, then

$$M = \frac{22500}{100} = 225$$

and we can assert that m differs from 225 by only

$$2\sqrt{\frac{m}{100}}$$

with 95 per cent confidence. Taking M as the approximate value of m, we get an error of only

$$2\sqrt{\frac{225}{100}} = 3$$

and can feel reasonably sure that m lies between 222 and 228 ct/sec, i.e., m will not lie between these limits in only about 5 per cent of the cases where such tests are made. We can use instead

$$3\sqrt{\frac{m}{N}} = 4.5$$

and assert that m lies between 220.5 and 229.5, with some assurance that this statement will be wrong in only 0.3 per cent of such cases.

The probability for various other intervals about the mean can be immediately derived from tables available in most texts on statistics. What degree of accuracy should one demand? Should one demand 0.3 per cent confidence or use the probable error and be satisfied to be right as often as wrong? Requirements will obviously vary with the risk incurred by being in error or by the difficulty encountered in taking a large value for N. Moreover, other factors are usually involved. There is obviously no point in prolonging the sampling procedure to get an answer which appears to be more exact than the limits of accuracy of the counter or of lower order than the decay of the source during the procedure. It would be quite hazardous for a mathematician not familiar with the type of counter to be used or with the field situation where the count is taken to attempt to set down any precise rules for practical work.

A much more detailed study of counting techniques is given by Jarrett in Statistical Methods Used in the Measurement of Radioactivity, MonP-126. For further reading on the fundamentals of the statistics involved, I recommend Hurd, Statistical Methods (Lecture Notes); Hoel, Introduction to Mathematical Statistics, Wiley, 1947; and Rider, Statistical Methods, Wiley, 1939; all of which are in the Health Physics Library or in the Training School Library.

5.3 THE POCKET METER AND PROJECTION MINOMETER
By H. H. Abee

The pocket meter used at Oak Ridge National Laboratory is a pencil-type ionization chamber. This meter is produced by the Victoreen Instrument Company. It consists of a tenite barrel with aluminum fittings at each end, an aluminum rod for a center electrode, polystyrene insulators, and an aluminum cap. The tenite barrel is molded to the aluminum fittings. A plug fitting closes one end of the barrel. The other end of the barrel has a tube fitting which leaves it open for insertion of the center electrode assembly. The barrel contains a graphite-impregnated paper liner to make it conducting and has a wall thickness of approximately 2.5 mm. The center electrode is an aluminum rod 1/16 in. in diameter. The electrode is insulated from the barrel by polystyrene insulators. One insulator, slightly larger in diameter than the electrode and molded to one end, insulates the electrode from the closed end of the barrel. The second insulator, whose diameter is such that it fits the inside of the barrel, is molded to the electrode about 1/4 in. from the other end This leaves a portion of the electrode protruding for making contact during charging and reading. The third insulator is a small bead molded around the center of the electrode to prevent accidental contact of the electrode with the wall of the barrel. The aluminum cap plugs into the open end of the barrel. Friction between a hard-rubber ring around the cap and the inner wall of the open aluminum fitting holds the cap in place.

The pocket meter was designed to measure gamma radiation. The walls of the meter are made of a substance which has the same effective atomic number as air, consequently the chamber may be any size. If the wall substance did not approach the effective atomic number of air, the calibration of the meters would be energy dependent. The wall thickness of the meter is sufficient to stop beta particles of less than 1 Mev energy. However, beta radiation capable of penetrating the wall will be recorded. As far as the meter is concerned, gamma rays and penetrating beta rays are not distinguishable in any way.

The pocket meter is charged to 150 volts by means of a projection minometer which will be discussed later. When exposed to gamma radiation, the air in the chamber ionizes and allows part of the charge to leak off. The amount of discharge or voltage drop is proportional to the radiation exposure. Exposure to 100 mr of gamma radiation will give a voltage drop of approximately 40 volts. Exposures between 0 and 300 mr are measured accurately by the pocket meter.

Pocket meters are calibrated with radium gamma rays, which consist of mixed energies ranging from about 0.3 to 2.2 Mev, with an average quantum energy of 0.8 Mev. They measure equally well gamma rays of all energies above 0.2 Mev. Between 0.08 and 0.2 Mev, their readings are about 10 per cent higher, which causes the wearer's record to show an exposure slightly larger than he actually received.

Each employee who works in the restricted area wears two pocket meters daily. The two meters should be worn side by side so they will have approximately the same geometry with respect to any incident radiation. The purpose of wearing two meters instead of one is to decrease the probability of getting an insignificant exposure record. The meters are collected and read daily. After reading, the lower of the two pocket meter readings is recorded as the significant reading, since any factor which tends to give a faulty reading would cause a meter to discharge more than the amount caused by the radiation exposure.

Faulty readings may be caused by any of the following reasons: (1) insulator leakage caused by dust particles or moisture on the insulator, a scored insulator, or a cracked insulator, (2) dropping the meter, or (3) misuse of the meter such as beating it on the edge of a table or taking the cap off and touching the electrode with the finger or some other object.

If a meter in routine use indicates an exposure of 30 mr or more, it is considered to be a leaking meter until proved otherwise by simple testing procedures. Should the meter fail to meet the requirements of the testing procedures, it is laid aside for repair.

In the case of insulator leakage where the insulator is not damaged, there are several things that can be done to repair the meter. First, the insulator may be blown with a stream of dry air to remove dust particles and surface moisture. Should this fail to stop the leakage, the insulator may then be washed with absolute alcohol to clean the surface of any remaining foreign matter and moisture. If this does not stop the leakage, the insulator may then be dipped in a solution of carbon tetrachloride and polystyrene which will give the insulator a coating of polystyrene.

Where the insulator of a meter is scored or cracked or when the above procedures fail to repair the meter, a new center electrode assembly should be installed.

This present meter is a vast improvement over the original pocket meter used here. The original meter had a bakelite barrel which was coated on the inside with graphite. Flaking off of the graphite in many cases caused the meters to give faulty readings. Instead of a plug-type cap, a threaded cap was used. Continual screwing of the cap into the meter wore off fine aluminum particles which collected on the insulators and caused leakage. This cap was also more cumbersome to use operationally. The center electrode assembly was different. The large insulator was not molded to the electrode, but was held in place by screwing it up against a ledge on the electrode with a small nut. There was no center-bead insulator on the electrode; consequently the meter was more frequently discharged by accidental contact of the electrode with the wall of the barrel. As a comparison, here are some operational figures showing the performance of the two meters. In December, 1945, 19,812 meters were read. Of these, 603 meters were completely discharged. In December, 1947, 44,141 meters were read. Of these, 70 meters were completely discharged. If it is assumed that none of the readings are due to radiation, in the first case the expected discharged or off-scale pairs would be

$$\left(\frac{603}{19{,}812}\right)^2 \times 9{,}906 = 8.9 \text{ pairs}$$

In the second case the expected off-scale pairs would be

$$\left(\frac{70}{44{,}141}\right)^2 \times 22{,}118 = 0.05 \text{ pair}$$

The Projection Minometer

The projection minometer is a quartz-fiber string electrometer. It contains a power supply for charging and a lens system for projecting the string image on a screen. The lens system has a magnification of 40×. The screen is a scale calibrated from 0 to 300 mr, and full-scale deflection of the image is 76 mm. When the minometer is discharged, the image of the fiber should be at the extreme right of the scale on a position marked Z. Upon charging, the deflection in millimeters of the fiber to the left is determined by the charging voltage and the tension on the fiber. The tension on the quartz fiber is adjusted so the 150 volts of charge on the instrument will give a full-scale deflection of 76 mm to the left. This position is the zero point of the scale. Should two or more minometers be required in an operation, it is imperative that the tension on the fiber of the instruments be adjusted so that a given charging voltage will give full-scale deflection on each instrument.

When a pocket meter is inserted into the minometer to be read, the two act as condensers in parallel. Following the principles of condensers in parallel, these two instruments share their charge in proportion to their respective capacities.

Summary

The pocket meter is a very good instrument for getting a daily dosage record of personnel working around radiation. The minometer is a part of the pocket meter, even though it is separated in usage.

5.4 SURVEY INSTRUMENT USAGE AND SELECTION (I)

By E. J. Kuna

This lecture, primarily, will be a résumé of my experiences with instruments, and as such, it will deal with a practical application of an instrument's characteristics to the surveyor's job. It must, of necessity, be a biased and prejudiced viewpoint, with theory subordinated for actual working practices.

The various and sundry instruments manufactured and developed at Oak Ridge National Laboratory are used to discover, differentiate, and measure the various types of radiation. Thus, for qualitative work the Walkie Poppie or the Walkie Talkie serves a useful purpose. For quantitative work the Zeus, Zeuto, or the Cutie Pie fills the requirements. In simplier terms, "first you must find it and then you must measure it."

The degree of radiation encountered limits one's choice of an instrument. Obviously an instrument such as the Cutie Pie with a maximum range of 10 r/hr could not be used in cell work where radiations up to 100 r/hr are encountered. This example would require a Fish Pole Probe which has an additional advantage of a 6-ft handle to cut down the exposure possibilities of the surveyor (inverse square law).

The radioactive tolerance of the area limits the choice of an instrument. Obviously a Walkie Talkie or Victoreen 263 would not be used in a high-background area. These would do well in a low-background area such as cold chemistry laboratories.

Finally, the degree of accuracy required of low-radiation readings would also limit the choice of an instrument. Such a requirement is required of isotope shipments where the electroscope is best suited to fill the need.

All instruments require practically the same technique of operation. First, of course, is the warm-up period, followed by the zero adjustment or setting. A few instruments also require the correct voltage setting, such as, for example, the Walkie Poppy. In use, the factors of distance and type of radiation encountered are noted along with the degree of radiation. At this point your own experience, knowledge, and observation guide you in making a judgment.

Factors of environment, light, heat, wind, and dampness affect all instruments to a degree. The Cutie Pie stands up well under this test, while the Zeus and Zeuto are very poor actors. Maneuverability and weight of the Cutie Pie puts it first again in comparison with the Zeus and Victoreen 263. The accuracy of these four instruments, the Cutie Pie, the Zeus, the Zeuto, and the Victoreen 263, is approximately the same. The calibration is made to within an error of ±10 per cent.

The Cutie Pie, Zeus, and Zeuto all have ion chambers that pick up the ionization current and send it through a selected resistance. The voltage drop across the resistance is then measured in microamperes. The Victoreen 263 is different in that it uses a G-M tube instead of an ionization chamber.

Summary

1. The Cutie Pie, a gamma-calibrated beta-indicating instrument with a 5 mr/hr to 10 r/hr range, an accuracy of 10 per cent, highly maneuverable, highly resistant to environmental factors. This has a window for differentiating between beta and gamma.
2. The Zeuto, an alpha-measuring instrument with a reported accuracy of 10 per cent, highly affected by environmental factors.
3. The Zeus, an alpha- and gamma-calibrated beta-indicating instrument between the ranges of 5 mr/hr to 5 r/hr. The accuracy is reported to be 10 per cent. It is highly affected by environmental factors and low maneuverability.
4. Victoreen 263, a gamma-calibrated beta-indicating G-M-tube instrument with three ranges from 12 to 20 mr/hr. It is best used as a qualitative instrument. It is slow reacting and heavy, and it has low maneuverability.

Remember that accuracy is a relative term useful for laboratory conditions. Do not compare two different instruments since their degree of sensitivity varies. Educate the survey requester to instrument limitations, but be diplomatic. Do a selling job of your services by the correct usage and interpretation of your instrument.

5.5 SURVEY INSTRUMENT USAGE AND SELECTION (II)

By G. R. Patterson

I am supposed to have given you a list of any references I use in this lecture, along with the abstract which you got belatedly. I shall give you those references now in case you might want to check my mistakes and really learn something about the instruments.

The most important general reference is the Instrument Manual compiled by D. M. Davis. I was lucky in that I received a copy of it, by coincidence, the same day I was notified that I was to speak to you.

The other references are: CP-3316, a paper on the Poppy, a pulse-type detector for alpha and other radiations; CP-3317, a paper on the Walkie Poppy, which is a portable model of the same type of detector; and MonH-448, a paper dealing with the conversion of the Poppy for detecting the soft beta radiation such as from C^{14}.

The instruments we are considering today are the Lauritsen electroscope, the Poppy Probe as used for alpha and C^{14} beta detection, and the Fish pole probe.

In the beginning let me say that, though I was exposed to physics, including some radio theory, in college, I know practically nothing about electronics and instruments. I cannot tell you why they work or how they work. Through my experience here at the laboratory I can tell you what they are intended to do, whether they work, and how they are used.

The Lauritsen electroscope is our most accurate portable instrument for measuring hard gamma radiation. In a purely gamma field it is as high as 99 per cent accurate. It is not calibrated to interpret beta ionization in rep or mrep, but it does respond to this ionization. Therefore, in field usage where radiation is usually a composite of gamma and beta radiation with a variety of energies, the electroscope is not quite so accurate. It is more accurate, however, than our other portable instruments, and it is much less subject to fluctuations in accuracy.

Its range of calibration is from 1 to 1000 mr/hr hard gamma. It will indicate intensities of less than 1 mr/hr, but it will not tell how much less it is.

It is a simple instrument, and it is easy to use. One simply charges the chamber with a push-button switch and times the fiber drift with a stop watch. The instrument requires no warm-up period.

Its main survey use is checking isotope shipments for conformance with maximum permissible radiation levels for shipping. A model of the Lauritsen electroscope using a boron-coated chamber is used in conjunction with a standard Lauritsen for computing thermal neutron flux.

It does have limitations. Since the operator must look directly into the microscope eyepiece at very close range, it cannot be used in small places where, for instance, the chamber of a Cutie Pie or a Fish pole probe could be thrust at arm's length.

The time-lag feature when making a reading makes it unsatisfactory for routine building or lab surveys, where perhaps many readings may be taken in a short time, or in scanning an area for hot spots or general background. This same time-lag feature, together with the fact that the surveyor is of necessity in the same radiation as the instrument, makes it undesirable for use in high radiation fields. Even though the fiber drift time is very short at the high limit of calibration and the maximum limit of the instrument is only 1000 mr/hr, it is much more satisfactory and desirable to use an instrument which can be read directly or instantaneously from a distance and which can be extended into the radiation.

The fact that the fiber drift time as the upper limit of calibration is approached is very short makes accurate timing with the stop watch very difficult at these higher levels.

The main cause of inoperation with this instrument is burned-out or very weak batteries. Other technical faults show up, also, such as dirty insulators or damaged insulators or fibers or flaking off of the conductive coating of the fiber, etc., but these are far less frequent.

When its limitations are realized and it is used as intended, the Lauritsen electroscope is a very good instrument.

The Poppy probe and amplifier is a pulse detector intended purely for qualitative detection or indication. It gives an audible signal indicating each pulse, but it cannot be used for quantitative measurement. Some of the Poppy amplifiers I have seen include a rate meter ostensibly for quantitative measurement of the radiation. However, none of those I have seen were calibrated or of any value.

When used for alpha detection the Poppy probe does not respond to beta or gamma radiation, and thus, unlike the Zeus and Zeuto, it can be used in the presence of beta-gamma radiation.

There are at least two models of the Poppy amplifier: one is a portable model which is battery-powered, and another uses an a-c power supply and amplifier.

Unfortunately the Poppy has many limitations. One serious limitation is already evident. The radiation is not measured.

The battery-powered amplifier is subject to the failings of all battery-operated instruments and to quite a few additional ones. It is extremely sensitive to weather and humidity changes. It is very hard to adjust to proper voltage, etc. In short, those I have seen never work. Several modifications have been tried, but to date I have not found one that is satisfactory.

The a-c–powered poppy is much more reliable and is a reasonably good detector. However, it does have serious limitations. The a-c Poppy is not very portable. It, too, is frequently very difficult to adjust to a proper or satisfactory operating voltage without background noise. It is subject to the line interferences and background noises of the a-c power supply—noises caused by motors, fluorescent lights, etc. Frequent loose connections in the cable and probe also give rise to considerable noise and flash indications. These noises, which for the most part cannot be told apart from true indications of alpha, make detection of small quantities of contamination or radiation difficult. The lower limit of detection of the instrument makes it practically impossible to detect concentrations of less than 1000 dis/min. In view of the emphasis placed on the need for freedom from alpha contamination, this seems to be a very serious limitation.

My experience with the Poppy for detecting the soft beta radiation of C^{14} is nil, and my knowledge of this application is very limited. I cannot give you any idea from practical experience how good or bad the instrument is. I hope I shall be able to give you some useful knowledge of its basis and its possibilities.

The range in air of beta particles from C^{14} is only on the order of 1 ft. Because of their very low energy, these particles, or the large majority of them, are stopped by even what are considered "thin-window" detectors. The very thin nylon window of the Poppy probe, which is only about 2/10,000 in. thick, makes it suitable for detecting these low-energy betas.

Some modifications are needed, however. The amplification-factor characteristics of air which is normally used in the Poppy probe are such that the voltage setting necessary for beta detection is too near the point of breakdown or the voltage setting for constant discharge.

The probe is slightly modified to permit a slight flow through the probe of a gas such as an argon-CO_2 mixture. Other gases also have amplification factors with a rate of change slow enough to be satisfactory, but the argon-CO_2 mixture was selected here for various reasons. Since the voltage for C^{14} beta detection is more critical, the power supply of the amplifier is also slightly modified to give a more stable voltage and decrease the lag in the change of voltage when the knob is turned.

The operation of the instrument is very similar to that of the alpha Poppy. However, when the operating voltage is properly set for the alpha Poppy there is very little background noise. Since the C^{14} Poppy is sensitive to beta and secondary electrons from gamma, there will be background noise comparable to that in a G-M tube detector. This makes it difficult to find the proper operating voltage, but a small beta source will help. This background also makes detection of low levels of activity rather difficult.

The Fish pole probe is quite similar to the Cutie Pie. Since it is calibrated for and used for the measurement of higher intensities of radiation, the ionization chamber is on the end of a pole. The surveyor can, from a safe distance out of the beam or field of radiation, hold the chamber in the beam or field and read the indication on the meter supported by a shoulder strap.

These probes are calibrated for hard gamma from 5 r/hr to as high as 100 r/hr. They also indicate soft gamma and beta, but are not calibrated for these. They are reasonably accurate and handy to use.

They are somewhat limited in their use by intention of design. They are not accurate below 5 r/hr, and each small dial division is equal to quite a number of roentgens or milliroentgens, as the case may be. This makes accurate reading difficult, especially since it is a little hard to hold the chamber steady at a distance and read the dial at the same time. Usually, however, at the intensity of radiation for which these probes are used, extreme accuracy is not vital. Persons are not likely to be entering fields of such high-intensity radiation.

All in all, where high-intensity radiations are likely to be encountered, Fish pole probes are very useful and necessary instruments for survey work. My own experience has been that they are good instruments for their intended use and not so subject to failings as are many of the other instruments on which we must now rely.

5.6 SURVEY INSTRUMENT USAGE AND SELECTION (III)

By R. L. Clark, E. L. Sharp, and P. E. Brown*

Instruments covered in the past two lectures were: Cutie Pie, Victoreen 263, Zeus, Zeuto, electroscope, Poppy, and Fish pole probe.

The past two lectures have given you descriptive material and key points in the selection and usage of the instruments listed above in making radiation surveys in the field. Unfortunately I was unable to attend these lectures, so a portion of the material presented today is likely to be repetitious. In addition, most of you have been making surveys with these same instruments for some time now. We hope, however, to be able to present some new ideas, or at least re-emphasize those old ideas which will be beneficial to you and the people you work with in your career as a health physics surveyor.

In selecting an instrument with which a survey is to be made, the two most important considerations which should govern your choice are: (1) type of radiation to be surveyed and (2) type of survey desired. The answer to the second question will generally be supplied by the person requesting the survey. The two types of surveys, of course, are qualitative or quantitative. Generally speaking, however, it will save you a trip back to the office if you take some quantitative instrument along, even when a qualitative survey is requested. About the only time you will be able to make a purely qualitative survey for beta-gamma activity is when the results are negative—that is, your survey with a scanning instrument fails to detect any activity above the natural background. If you do detect activity on the object or in the area surveyed, the next question put to you will most likely be, "How much?" Realizing this contingency and being prepared to take care of it will, in addition to saving steps for you, cast a favorable reflection both upon you as an individual and on the whole group. Once you realize that our official function here is to serve in an advisory capacity, you will understand why each surveyor is, in addition, a public-relations man for the division. For this reason I should like to emphasize the necessity for accuracy in making surveys.

It has been pointed out in previous lectures that the accuracy of the instruments you are called upon to use is decidedly limited. Accepting that fact, let that be the only limitation placed on the accuracy of your surveys. A little practice with a yardstick will help you in making accurate estimates of the distances involved in making surveys. You already know that none of the instruments now available for survey work will permit you to determine the radiation level at the surface of a container, on a floor, or, in other words, give the information most generally desired, namely, a "contact" reading. You can do the next best thing and give a reading at a distance, provided you are able to accurately estimate distances.

This ability is not only valuable to the person for whom you survey, but it is an aid in obtaining what is called "reproducibility of results." This factor of reproducibility is much to be desired if

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*Text by R. L. Clark.

you happen to be called upon to resurvey the same object or area—or if another surveyor, not knowing the results of your survey, is called on to survey it.

Another important factor in obtaining accuracy is to allow sufficient warm-up time before attempting to make a survey with an instrument. The recommended warm-up time for most of the instruments ranges from 30 sec up. In the Health Physics Instrument Manual compiled by D. M. Davis, his calibration procedure calls for a 5-min warm-up time before the calibration procedure is started. Obviously you cannot postpone every survey you are required to make for 5 min while you allow the instrument to warm up. You can allow as much warm-up time as possible, however, and remember to always allow at least 30 sec for those instruments which require a warm-up period. A procedure which I have found generally satisfactory is to take the 30-sec time in the office and make a test setting to see if the instrument appears to be working. Then, before leaving to make the survey, I put the sensitivity switch back on the "zero" position and run the microamperes up to around 5 while walking to the place where the survey is to be made. This allows me to add time to the warm-up period without delaying the actual survey too much.

Let us go back now and consider for a few minutes the other factor governing your choice of an instrument for making a survey, the type of radiation to be surveyed. By type of radiation I am referring to whether the material to be surveyed is an alpha, beta and/or gamma, or neutron emitter. This information may come either from your own knowledge of operations and materials in your areas or from the person who requests the survey. If this information is not available, it will be necessary for you to make a scanning survey first to determine the type of instrument needed to actually measure the intensity of the radiation. Once you have identified the radiation which you wish to survey, your choice of an instrument to use should be governed bv the points given you by Kuna and Patterson. To briefly reconsider these points:

1. For low-level beta or gamma radiation the Victoreen or Lauritsen electroscope serves best, with the electroscope being the more accurate and the Victoreen being the more sensitive of the two.
2. For intermediate-level beta or gamma radiation in the range of 5 mr/hr to 10 r/hr the Cutie Pie is the most commonly chosen. I should like to point out that in the range up to 1 r/hr the electroscope is still usable and more accurate than the Cutie Pie.
3. For high-intensity beta or gamma in the range from 10 r/hr up to 100 r/hr the Fish pole probe is the only choice available.
4. For surveying of alpha emitters, either the Zeus or the Zeuto may be used. The Zeus is preferable because it permits you to determine whether the activity is purely alpha or a mixture, while the Zeuto does not, even though it is sensitive to the beta-gamma radiation.
5. For surveying thermal neutron emitters it is necessary to use two electroscopes. One scope is the ordinary Lauritsen type, and the other is of the same type, but it has a thin coating of boron on the inside of the chamber. This boron, under neutron bombardment, emits alpha particles. These particles produce ionization in the chamber in addition to that produced by gamma rays from the source or beam being surveyed. The difference in the readings on these two scopes is due to the additional ionization produced by the alpha particles and, through calibration, can be used to measure the thermal neutron flux.
6. For fast neutron surveys we have only the Chang and Eng, which has so many limitations that for general survey purposes the instrument is nearly useless.*

5.7 RADIATION MEASUREMENTS WITH FILM

By John H. Roberson†

The Photographic Plate

Photographic film is composed of two essential parts, (1) emulsion and (2) the base or support for the emulsion. The active ingredient in a photographic emulsion is pretty well standardized, con-

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*More satisfactory instruments for fast-neutron surveys, using a proportional counter, have been developed in this division during the fall of 1949.

†Notes taken by C. Scott; edited by J. H. Roberson.

sisting of a chemical compound of silver and bromine. The AgBr has two characteristics which make it desirable for photographic emulsion. One is that when struck by light, the nature of the crystal changes very slightly. Subsequently the AgBr at the place where the light or radiation struck the emulsion can be changed chemically and retained in permanent form. Second, AgBr is insoluble in water, which means that it can be washed and treated with various chemicals without danger of losing it.

The AgBr is mixed in a gelatin carrier, which has very convenient properties for photographic use. It is transparent, and when immersed in water it will swell to several times the original size, permitting chemical solutions to go throughout the gelatin and get to the AgBr.

The base gives support to the gelatin, holding it in shape when the plate is immersed in water. The gelatin swells in one direction only and to 8 or 10 times the original thickness. When dried it comes back to the original position.

It is hard to describe the effect of radiation on photographic emulsion. Historically we find that early experimentors discovered that some colors of light would activate a silver halide crystal. Other colors would not affect it. Ordinary AgBr emulsion will respond to blue and green light and also to radiation in the near ultraviolet. Toward the red end of the spectrum the photographic effect falls rapidly to zero.

When light strikes AgBr it produces a change in the energy of electrons in the crystal. The thing that is important is that this changed crystal can be reduced by certain chemical agents so that the Ag in the AgBr crystal becomes metallic. This is the process known as developing.

After developing the image, the next thing to do is clean out from the gelatin that part of the AgBr not struck by light. The AgBr is soluble in certain chemical solutions, one of which is sodium thiosulphate, commonly known as "hypo." After development and "fixing" with hypo, the photographic image remains in the gelatin in the form of finely divided metallic silver, which appears black in color.

The speed of an emulsion as used in ordinary photography depends on the amount of light required to produce a given blackening in the developed plate. Speed is achieved in emulsion by increasing the size of the AgBr grain. It takes a certain number of light quanta to activate a grain. Apparently the number of quanta required is almost the same for big crystals as for little crystals. In order to get the maximum amount of silver in the developed image, the largest permissible grain size is used.

Dosage Measurement with Film

The health physicist tries to make dosage measurements in terms of roentgens. The roentgen is based on the ionization produced in air. In order to measure roentgens with photographic film, it is necessary to interpret the effect of radiation which was absorbed in the film into ionization that the radiation would produce in air. Since film and air do not absorb radiations of all wavelengths in the same proportion, the film is said to be "energy dependent" (i.e., wavelength dependent) when used as a dosimeter. Therefore the response of films depends on ratio of absorption of or gamma rays in film to the absorption that the same rays would have had in air. The absorption coefficient in air is almost a constant in the whole X-ray region. Ag and Br are heavy elements, and their photoelectric effect is large in the soft X-ray region of the spectrum. Because of this photoelectric effect, the absorption in the film emulsion is not the same as it is in air, which is composed of light elements.

Dr. Roberson then went on to explain some of the work Joe Deal had done in cooperation with the National Bureau of Standards,* and discussed the personnel monitoring film meters to some extent.

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*Deal, Roberson, and Day, Roentgen Ray Calibration of Photographic Film Exposure Meters, Am. J. Roentgenol. Radium Therapy, 59: 5 (1948).

5.8 CALIBRATION TECHNIQUE

By D. M. Davis*

Sources Used in Calibration

Gamma sources	Alpha sources
0.96 mg Ra	4,000 dis/min
23 mg Ra	13,000 dis/min
27 mg Ra	20,000 dis/min
48 mg Ra	42,000 dis/min
100 mg Ra	82,000 dis/min
1 curie Co	169,000 dis/min
	200,000 dis/min
	300,000 dis/min

The 0.96-mg Ra source is used in the calibration of monitrons. The 27-mg Ra source is used in the calibration of electroscopes and other fiber instruments of that nature. The 23- and 48-mg Ra sources are used in the calibration of film. The Co source is used in the calibration of the Cutie Pie, Zeus, and other ionization-type instruments. The 100-mg Ra source is often used in checking our calibration setup. The alpha sources are used for calibration of instruments which are designed for measuring alpha.

Calibration Cycle

Electroscopes, Lauritsen and L & W, are calibrated at 4-week intervals. Ionization-type instruments are calibrated at 2-week intervals. For various reasons the calibration of the instruments may vary from time to time.

With a calibration cycle as listed above, the instruments should not vary more than 10 per cent from the calibration chart. Fiber instruments vary less from the calibration than do ionization-type instruments.

Calibration Theory

The calibration of an instrument is to set up a standard to be used for measuring radiation intensities in an unknown field. By use of an instrument that is properly calibrated, we are simply saying that the radiation measured in the field is the same as that measured at a known distance from a known source. In other words, if by using a 50-mg Ra source we determine that at a given distance we are getting 100 mr/hr and the meter reads 10 μamp at this position, then in the field when the meter reads 10 μamp we can assume that we are in a field of 100 mr/hr.

Beta Calibration

We are often confronted with the question, "How do you calibrate for beta?" We do not at the present time calibrate for beta. There are many problems involved in the calibration of beta, such as various energy ranges, etc., which make the problem much more complicated than gamma calibration. It is recognized that beta calibration is desirable, and at the present time people in the Research Section are working on a method for the calibration of beta. We do have, on most of the instruments, what we call a "window." This window, however, does not give us an accurate measurement of beta in mrep, but only the amount of ionization. The efficiency of the instrument for beta radiation varies with the energy of the beta radiation.

One statement might be made as to the effect of contamination of instruments. Contamination on an instrument will always result in an erroneous reading. The fiber instruments will show a

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*Notes taken by C. Scott; edited by D. M. Davis.

greater error than the ionization type in which it is possible to zero out the reading due to contamination, but contaminated instruments should not be used in survey work.

Precalibration Checks

The following checks are made prior to the calibration of an instrument: (1) contamination check—this contamination check is a secondary precaution; all instruments should be checked for contamination before leaving the field, (2) batteries, (3) dial function, and (4) general physical condition.

Appendix A: Data on Sources and Calibration Standards*

(a) Radium Sources as Gamma Radiation Standards. Radium, in equilibrium with its short-lived decay products, is used as a standard for many source-strength or radiation-intensity measurements. For this purpose the radium is enclosed in a thin platinum filter, usually of either 1/2 or 1 mm thickness.

For small radium sources within platinum filters of thickness greater than 0.3 mm, the source strength is given by the equation

$$S = 8.98\ (1 - 0.13t)\ M$$

where S is the source strength, measured in roentgens per hour at 1 cm, t is the thickness of the platinum filter, measured in mm, and M is the mass of radium, measured in milligrams.

For a 1/2-mm platinum filter, $S = 8.4\ M$, and for a 1-mm platinum filter, $S = 7.8\ M$.

It is frequently convenient to use the foot as a unit of distance in estimating the radiation hazard from a radium source. If the source strength is expressed in roentgens per hour at 1 ft, the above equations become

$$R_1 = 0.0097\ (1 - 0.13t)\ M$$

for a 1/2-mm Pt filter,

$$R_1 = 0.0090\ M$$

and for a 1-mm Pt filter,

$$R_1 = 0.0084\ M$$

(b) Radium-Beryllium Neutron Sources. A common type of neutron source utilizes the alpha particles from radium and its short-lived equilibrium products to produce neutrons by their reaction with beryllium. Since three of the products of radium are alpha emitters, there are four alpha particles per disintegration of radium. Measurements on the neutron radiations from various Ra-Be sources give results varying from 8.7×10^6 to 14×10^6 neutrons/sec/g of radium.

Radium-beryllium sources must be shielded for both gamma and neutron radiation. Since the physiological effects of the two types of radiation are presumably additive, each type of radiation must be shielded more effectively than if it alone were present.

(c) Procedure for Determination of Source Intensity. (1) Sources Contained in a 0.5-mm Pt Tube. The radium-element content of the tube shall be obtained by adding to the "equivalent value" appearing on the face of the Bureau of Standards Certificate for that source the percentage of this value indicated in the note at the bottom of the certificate (for a 0.5-mm Pt tube this percentage is customarily 4.3). The source intensity will then be obtained in milliroentgens per hour at 1 cm by multiplying the radium element content, measured in milligrams, by the factor 8400 mr/hr/mg at 1 cm.

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*Supplementary study material compiled by F. Western.

(2) Sources Contained in a 1.0-mm Pt Tube. The radium-element content of the tube shall be obtained by adding to the "equivalent value" appearing on the face of the Bureau of Standards Certificate for that source the percentage of this value indicated in the note at the bottom of the certificate (for a 1.0-mm Pt tube this percentage is customarily 9.4). The source intensity will then be obtained in milliroentgens per hour at 1 cm by multiplying the radium-element content, measured in milligrams, by the factor 7800 mr/hr/mg at 1 cm.

(d) Gamma Radiation from a Radium Source. The gamma radiation from a radium source is almost entirely from its equilibrium products, RaB and RaC. The number of gamma quanta from these products, per radium disintegration, is approximately 2.3, and the effective average energy of the filtered radiation is approximately 0.8 Mev. The distribution of energy in the gamma radiation from a radium source is shown approximately in Table 5.1.

Table 5.1 — Gamma Radiation from Radium Source

Gamma energy, Mev	Number of quanta per radium disintegration	Origin of gamma
0.18	0.01	Ra
0.24	0.12	RaB
0.29	0.26	RaB
0.35	0.45	RaB
0.61	0.66	RaC
0.77	0.06	RaC
0.93	0.06	RaC
1.12	0.21	RaC
1.23	0.06	RaC
1.38	0.06	RaC
1.76	0.26	RaC
2.20	0.07	RaC

(e) Absorption by Lead of Gamma Radiation from a Radium Source. Since the radiation from a radium source consists of quanta of many different energies, the radiation transmitted through a lead shield becomes harder as the shield is made thicker. One cannot, therefore, give an absorption coefficient or half-value layer that can be used for all thicknesses of shielding.

Table 5.2 uses data from R. D. Evans, J. Ind. Hyg. and Toxicol., 25: 253 (1943), and G. Failla, Am. J. Roentgenol. Radium Therapy, 54: 574 (1945).

Table 5.2 — Absorption of Gamma Radiation from Radium

Lead, in.*	Fraction of radiation transmitted*	Lead, in.*	Fraction of radiation transmitted*
0	1	3	0.017
0.125	0.8	3.5	0.01
0.25	0.64	4	0.005
0.5	0.43	5	0.0017
1	0.21	5.5	0.001
1.6	0.1	6	0.00058
2	0.058	7.7	0.0001

*All values are intended to be significant to two figures. Final significant zeros are dropped for ease in reading.

Appendix B: Example of NBS Certificate

National Bureau of Standards Certificate for One Specimen of Radium Salt

NBS No. ________

Submitted By ________

Description. The material is contained in a metal tube____mm long and ___mm in external diameter. The specimen and container weight _____. The Bureau is informed that the material is a compound of radium. After the measurements had been completed, the specimen with a card bearing the number of this certificate was enclosed in a package suitably inscribed and secured by a seal of this Bureau.

THIS CERTIFIES that the specimen described in the preceding paragraph has been compared with the Radium Standard of this Bureau and has been found to have a gamma radiation equivalent to that from

xx.xx milligrams of radium

in radioactive equilibrium and contained in a tube of Thuringian glass 0.27 mm thick.* The uncertainty in this value does not exceed 0.7 per cent. Observations extending over 13 days indicate that the radium contained in the specimen is in radioactive equilibrium.

For the Director,

by ________________________
Chief, Radioactivity Section,
Division of Optics.

Test No.

Date:

*NOTE: The Bureau is informed that the tube is of platinum and has walls 0.5 mm thick. On this basis if the radioactive material were contained in a glass tube of the kind specified above, the gamma radiation from the specimen when in equilibrium would be approximately 4.3 per cent greater than the value named in the body of the certificate.

The observation upon which this certificate is based do not serve to distinguish between radium preparations and those containing mesothorium and radiothorium.

5.9 INSTRUMENT CARE AND REPAIR

By Davis, Gatner, Ryan, and Hamrick*

Cost of Instrument

The following list indicates the cost of some of our more commonly used instruments:

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*Notes taken by C. Scott; edited by D. M. Davis.

Cutie Pie	$200.00	a-c Poppy	$450.00
Zeus	300.00	Beckman	300.00
Electroscope	90.00	Victoreen 263	230.00
Zeuto	240.00	Monitron	500.00
Scaler	350.00 – 500.00	Dosimeter	45.00
		Landsverk	200.00
		G-M tube	10.00 – 50.00

The prices of the instruments may vary slightly between different companies, but the prices given are essentially correct.

Care of Instrument

The manuals accompanying most instruments state that they are light, portable, rugged, easy to operate, etc. The instruments are only rugged, however, when used in proper survey techniques. Abuse or maltreatment of any of our portable survey instruments will result in improper operation or faulty readings. Some of the ways in which the instruments may be abused or mistreated are as follows: (1) leave batteries on, (2) contamination, (3) jars or rapid movement of instrument, (4) dropping or knocking the instrument over, and (5) tampering with the instrument in the field.

We should like to stress that surveyors should not attempt repair work on the instruments in the field. When an instrument is in need of repair, the surveyor should call the Calibration Building and have the instrument picked up for repair. Where possible, a replacement will be issued at the time the instrument is picked up.

Repair of Instrument

There are several definite types and varieties of instruments, each having its own peculiarities as to malfunctions and methods of repair. We shall discuss only three instruments, which will be three definite types of instruments: the Cutie Pie, the Walkie Poppy, and the Victoreen 263.

Cutie Pie. The following is a list of some possible malfunctions of the Cutie Pie with probable causes:

1. Will not zero
   Causes
   a. 7 1/2-volt batteries do not have same voltage
   b. Resistors on both sides of zero potentiometer not balanced
   c. 22 1/2-volt battery down
   d. Filament battery down
2. Needle fluctuates
   Causes
   a. Insulator in back of ionization chamber dusty or dirty
   b. A bad connection anywhere
   c. A high-value resistor open
   d. Resistor dirty or tube dirty
   e. Burned wax
3. Reads off scale
   Causes
   a. Stud mounting ion chamber grounded to case
   b. Corroded or dirty connection
   c. One of resistors bad
   d. Bad switch on potentiometer
4. Will not zero on sensitivity scales
   Causes
   a. High-value resistors bad
   b. Tube base may be bad
   c. Dirty insulator at end of ion chamber

5. Needle goes below zero
   - Cause
     - a. Tube burned out
6. Indicates reading when no radiation is present
   - Causes
     - a. Contamination of instrument
     - b. Light getting into chamber
     - c. Mechanical balance of meter
7. Reading is less than known radiation
   - Causes
     - a. Instrument grounded at wrong place
     - b. Chamber and box giving signals of opposite polarities
     - c. Needle on meter sticks
8. Change of zero set in changing position of instrument from horizontal to vertical
   - Cause
     - a. Mechanical balance of meter
9. Needle drifting
   - Cause
     - a. Batteries getting low

<u>Walkie Poppy</u>. The following is a list of some possible malfunctions of the Walkie Poppy with probable causes:

1. Majority of troubles are due to batteries
2. High background noise
   - Causes
     - a. Leakage in high voltage
     - b. Bad bias cells of amplified section supply
3. No noise or pops
   - Causes
     - a. Tube burned out in amplifier section
     - b. High-voltage failure
4. Intermittent operation
   - Cause
     - a. Bad cable connection

NOTE: The filament batteries must be changed quite frequently. High voltage lags behind the adjustment. Operating point is very critical.

<u>Victoreen 263</u>. The following is a list of some possible malfunctions of the Victoreen 263 with probable causes:

1. Voltage is up but no signal or pops
   - Cause
     - a. Bad G-M tube
2. No background at all
   - Cause
     - a. Burned-out tube
3. High background outside radiation field
   - Causes
     - a. Low batteries
     - b. Bad tube
     - c. Contamination on instrument
4. Intermittent operations
   - Cause
     - a. Bad cable connections

5.10 THE PHOTOGRAPHIC EMULSION

By M. M. Shapiro*

In the past two years several physicists have applied the photographic-emulsion technique with remarkable results: Powell of England, Gardner of this country, and Lattes of Brazil. Powell discovered a new kind of meson (mass approximately $300m_e$). He further demonstrated that one of these heavy mesons can decay spontaneously into a lighter meson of mass about 200 times electron mass. Gardner plus Lattes succeeded in producing mesons with a cyclotron at Berkeley.

The photographic emulsion consists of little grains of silver bromide (AgBr) embedded in a matrix of gelatin. AgBr is the sensitive material.

Ag^+ Br^- ions

Action of light: When a quantum of light strikes an AgBr grain, some silver ions are changed to free silver. When the emulsion is developed, the entire grain is reduced to free silver (and, of course, Br_2). X rays also are capable of affecting photographic emulsion. Among the other particles which can do so are: group A consisting of gamma rays and beta particles (electrons), and group B, consisting of alpha particles, protons, deuterons (H^2), tritons (H^3), and fission fragments. All will blacken the emulsion, but there is a rather significant difference in the manner in which group B affects the emulsion from the way in which photons or electrons affect it.

If you have a source of alpha particles and expose them on a photographic plate and later develop the plate and then inspect it under a microscope, you find the alpha particles have left not just blackening, but tracks whose lengths are of the order of 20 to 40 μ. ($1\ \mu = 10^{-4}$ cm); 40×10^{-3} mm ≈ 0.04 mm. You must have a microscope to observe this.

Protons make the same kind of track, except that it does not ionize so densely. There are more gaps in the track. If you look at these with bright-field illumination you will get black spots, and if you have a dark field, then you will see bright spots on a black background.

Similar tracks are made by deuterons and tritons. Fission fragments produce exceedingly dense tracks. In almost any emulsion their ionization is much denser. It is possible to see both fission fragments in the same emulsion with suitable techniques.

A photographic emulsion is much simpler to operate than a cloud chamber, and one can study the details of tracks left behind by all these particles in a photographic emulsion.

Neutrons can be studied with these emulsions by the secondary effects which they produce. The blackening produced by X rays and gamma rays is due entirely to secondary effects. Electrons are ionizing particles because they are charged, and in passing through the emulsion they produce a blackening. The Eastman Company has succeeded in making film so sensitive that even electrons can leave a visible track. This is quite an achievement.

Principal Advantages of This Technique

1. Continuous sensitivity. (a) The cloud chamber is only sensitive a small fraction of the time; (b) rare events: for use with valuable equipment where time is at a premium.
2. Larger volume. (a) The larger volume is from great stopping power of dense emulsion versus air. Approximately 2000 times stopping power of air (1600 to 2000); (b) convenient, compact apparatus; (c) the typical thickness of emulsions is as follows: ordinary, approximately 10 μ; special, approximately 25 to 300 μ; and typical, 50 to 100 μ.
3. Permanent record
4. Inexpensive technique
5. Simple in principle (however, care is required to interpret results)

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*Notes taken by C. Scott; edited by M. M. Shapiro

Disadvantages

(These are less serious now since better emulsions have been obtained recently which are highly concentrated in AgBr.)

1. "Fog" grains, dense background in some experiments. The variety of agents to which emulsions are sensitive include (a) mechanical shock, (b) thermal change, and (c) cosmic rays. Often one gets other effects along with the desired ones.

2. Tedium involved in inspection (microscopy). Particles cannot be bent enough with magnetic fields once they are in emulsion (however, these can be bent outside plate before impingement). Get only ranges, directions, and number.

Applications in Health Physics

Monitor for neutrons: both slow and fast neutrons can produce secondaries to make tracks.

Slow Neutrons.

$$n + {}_7N^{14} \rightarrow p + C^{14} + Q$$

($Q = 1/2$ Mev) 6- to 7-μ length protons, 4 to 6 grains.

Number of tracks gives estimate of exposure.

Fast Neutrons. These give recoil proton, seen in the same way. Even 50 per cent accuracy is hard to get for one-third or one-fourth of tolerance.

Alternative Method for Slow Neutrons. (This method was proposed and demonstrated by M. Shapiro and J. Barnes.) The plates were impregnated with

$$Li^6 + n \rightarrow He^4 + H^3 + 5 \text{ Mev (alpha and triton, principal path } \sim 54\ \mu\text{, some denser)}$$

$$B^{10} + n \rightarrow Li^7 + He^4 + 2.5 \text{ Mev (alpha, principal path 7 to 8 } \mu\text{, much denser)}$$

With Li, for a given neutron exposure, it is possible to get 150 times as many tracks as from N interaction. With B it is possible to get 1500 tracks compared with N interaction.

UNIT 6. APPLIED HEALTH PHYSICS

6.1 PERSONNEL MONITORING PROCEDURES

By J. C. Hart*

The personnel-monitoring program is designed for the three main purposes of providing for (1) the prevention of unnecessary exposures to personnel, (2) the establishment of permanent legal and clinical records, and (3) the statistical aid to other methods of measuring radiation.

Personnel meters currently in use indicate the extent of exposure to radiation. Tissue damage may be approximated from the readings if certain facts are known regarding the type of radiation, the energy involved, and habits of the users. The health physics groups throughout the various projects have made records available to individuals at the request of the individual.

Effective personnel monitoring involves measurements of both external and internal radiation. External radiation is monitored at present by pocket meters (ionization chambers) and film meters. Internal radiation presents two problems in that we are faced with both radiation and toxic effects. Internal radiation is measured by a study of body fluids (this method is discussed in Lecture 6.3). Pocket meters used for measuring external radiation are currently worn in pairs. Their primary purpose is to give a quantitative measurement of radiation exposures on a daily basis. Film meters are not practical for measuring small doses and/or giving a quick check. Film meters do measure larger quantities of radiation and have the added feature of providing a permanent record. The effective pocket-meter range is from 0 to 300 mr. The effective film-meter range is from 0 to 20 r. A densitometer capable of measuring densities up to 4 or 5 would increase the range of the film meter by a factor of approximately 2 or 3. The film meter is impractical to process on a daily basis because of economics and the length of time required.

The film meter tends to show a threshold effect. The emulsion currently in use (Du Pont 552) picks up traces of radium gamma between 7 and 14 mr. A variation in densitometer readings introduces a slight error. Also, not all films give the same density over their surface. These three effects prevent us from interpreting accurate film readings below 30 mr on a corresponding radium-gamma calibration. Fortunately the pocket meter fills in this gap, providing readings to 5 mr.

It is desirable to integrate readings between pocket- and film-meter weekly totals. The problem which presents itself as the most difficult in integrating the film meters and pocket meters begins with the definition of the health physics week. This is increasingly difficult where the laboratory works shifts. The weekly pocket-meter total must be computed through Saturday, since all records culminate. It follows from this that the week must end with the beginning of the first shift following the initial film servicing. Thus the health physics week begins with the 4 to 12 shift on Saturday at 4:01 P.M.

Every available film meter is processed Saturday morning. This would include the meters for all persons who worked the 12 to 8 shift Saturday morning and those who will be working the 4 to 12 shift Saturday night. The only meters not processed on this initial servicing are those being used by the 8 to 4 shift. These meters will be processed on the 4 to 12 shift routine. From this it is seen easily that the 4 to 12 shift workers on Saturday use films which will not be processed until the following week. Thus we are correct in assuming that the health physics week begins at 4:01 P.M. on Saturday.

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*Notes taken by C. Scott; edited by J. C. Hart.

It is assumed that the lowest reading in a pair of pocket meters worn together is the most nearly correct reading (an understanding of the ionization chamber–minometer proves this to be true). A significant reading has been defined as the lowest reading in a pair, provided that it is not off scale. It follows then that an insignificant reading has no meaning. There are four types of insignificant readings. To simplify record keeping they have been symbolized as follows:

Number	Symbol	Meaning
I	OS	Off-scale meter
II	DM	Damaged meter
III	LM	Lost meter
IV	CM	Cap missing

When an insignificant reading is noted the meter involved is immediately checked for contamination leakage or damage. A reading is considered insignificant where both entries in a pair are insignificant. Similarly, we have significant and insignificant readings on film meters. Symbols have not been developed to describe film-meter insignificants, and when they occur the condition is noted by a statement.

6.2 EXPOSURE REPORTING AND PERSONNEL MONITORING RECORDS
(Demonstration of Personnel Monitoring Techniques)

By J. C. Hart*

Minometry

Reading. Two hands working in rhythm manipulating the controls on the face of the minometer have been known to cause incomplete charging of pocket meters. For this reason the technician is trained to use one hand at the control panel. The charging button being strategically located at an appropriate distance from the minometer well makes it impossible to effect the above undesirable conditions if one hand is used. This is usually the right hand. The left hand serves the purpose of holding the pocket meter cap, a desirable feature in that there is no lost motion such as would be encountered in searching for a cap deposited on the table. In addition, there is less likelihood of the cap picking up dust contamination which in turn would cause leakage across the electrodes of the pocket meter.

One recognizes incomplete charging of meters when, in a series of readings, there is a high frequency in the off-scale region between 110 and 130 and between 40 to 80. Readings in the off-scale region can be traced to (1) mechanical discharge, (2) loose connections in the minometer well, and/or (3) leakage. Readings in the other two regions when due to incomplete charging are due to (1) a loose minometer well and/or (2) the pocket meter being inserted in the minometer with removal from the well before the charging button is depressed. The latter is the most commonly encountered event when both hands are used at the control panel.

Testing. Although the vast majority of readings are legitimate (1 bogus reading in 500), all pocket meters which read 20 or more are treated as "suspects." Suspects fall into three classes: (1) the legitimate reading, (2) a "leaker," and (3) loose connections. All three classes are catalogued, and the good meters are put back into service after the following tests are performed:

1. Vibration Test. When a meter gets loose parts or the graphite conducting liner begins to slough off, leakage across the electrodes results. An ordinary commercial Boerner shaker has been remodeled for the purpose of giving the meters a vibration test. When they become "suspects" they are charged, put on a Boerner shaker for 1 min, and then read. Meters reading 20 mr or less are set aside for leakage tests. Meters reading 25 mr or more are sent to repairs.

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*Notes taken by C. Scott; edited by J. C. Hart

2. Leak Test. After the vibration test, meters are recharged and placed on an 8- to 16-hr stand test. Meters reading 20 mr or less are declared good meters and returned to service. Meters reading 25 mr or more are sent to the repair shop.

NOTE: The passing 20 mr or less value varies from time to time, depending upon the needs of the laboratory. The value has never been higher than 30 mr nor less than 10.

A Simple Laboratory Method for Calibrating Film

A. Formula

$$\text{mr/hr} = \frac{K \times \text{mg Ra}}{d^2}$$

where K = 7.8 r/hr for 1 mm Pt = 8.4 r/hr for 1/2 mm Pt
mg Ra = milligram of Radium
d^2 = distance in cm
S = source

B. Method (assumed)
  1. 100 mg of radium in a 1-mm platinum capsule
  2. 24-hr exposure
  3. Table

Exposure, mr	Distance, cm
100	432.4
250	273.5
500	193.5
750	157.8
1000	136.8
2000	96.7

Film Marking Equipment

A. Brass Plate Shield
  1. Capacity, 54 film meters
  2. Dimension, 18 by 18 in.
  3. Thickness, 1/4 in.

B. X-ray Marker
  1. X-ray—General Electric dental X ray (price approximately $750)
  2. Settings
    a. kv, 35 to 40
    b. ma, 5 to 6
    c. Time of exposure, 3 sec
    d. Distance from tube to brass plate, 30 in.
  3. Box shield
    a. Inside dimension, 20 by 20 by 50 in.
    b. Wood, 3/4 to 1 3/4 in.
    c. Lead lining, 1/8 in. sealed at all joints

Film-changing Equipment

A. Film slide and opener

B. Contamination checking counter of "over" and "under" G-M tube beta-gamma Poppy

Dark Room Equipment

A. Developing Tank

Fixer	Running-water coolant and rinse	Developer
68°F	68°F	68°F
5 gal	10 gal	5 gal
10 min	1 min	3 min

(Room temperature, 68°F)

B. Water bath

Running water
54-gal capacity
68°F
(Controlled by hot and cold water mixer)

C. Film dryer
1. Commercial type
2. Provides moderate air flow at 85°F
3. Dries film in approximately 20 to 60 min, depending upon the humidity of the room in which it is located

Densitometry

A. Derivation of density

$$T = \frac{I}{I_0}$$

$$O = \frac{I_0}{I}$$

$$D = \log_{10} \frac{I_0}{I}$$

where T = transmission
O = opacity
I = intensity after filtering
I_0 = intensity at source
D = density

B. Definition of density

Den. 1 = $\frac{1}{10}$ transmission
Den. 2 = $\frac{1}{100}$ transmission
Den. 3 = $\frac{1}{1000}$ transmission
Den. 4 = $\frac{1}{10{,}000}$ transmission
Etc.

Filing Board

4th Digit

3d Digit	0	1	2	3	4	5	6	7	8	9
0	W									
1										
2				X						
3										
4									Z	
5										
6										
7						Y				
8										
9										

4 3 2 1
A. For four (4) digit Nos. 1 0 0 0
B. Place films in groups of 100
C. W = location for film in 0 to 100 range
X = location for film in 3200 to 3299 range
Y = location for film in 5700 to 5799 range
Z = location for film in 8400 to 8499 range

Appendix A: Plastite-film Ring Meter

A plastic-film ring meter has been developed at the Oak Ridge National Laboratory in collaboration with certain commercial plastic manufacturers. It is intended to be used as a beta and/or gamma monitoring device for measurement of radiation received by specific parts of the body such as hands.

This plastic-film ring meter is made of a special highly opaque plastic. It is thus highly adaptable for all use with film, resisting even out-of-door exposure to strong sunlight for reasonable lengths of time. This meter may be used either as a ring for the finger, pinned to various parts of the clothing near the radiation sources to be monitored, or used in any general manner where local radiation exposure is to be checked. Because of its small size, convenience, and compactness, one is able to obtain a more accurate radiation-field distribution of intensity readings.

The film ring meter utilizes a cadmium filter above and below the film. This cadmium filter consists of a 9/16-in. washer with a 3/16-in. hole, the metal being 1 mm thick. The hole in the cadmium washer acts as an open window for determining the entry of soft radiation onto the film and radiation quality comparative measurements.

Film disks 37/64 in. in diameter are cut out with a flat punch of our own design and inserted between the cadmium washers. The film used is the more sensitive film of the Du Pont type 552 pack. After the meter is loaded with film, the plastic cap which is 11/16 in. in diameter and has a thin top 0.02 in. thick is then sealed in place, using a solvent.

The ring composition is molded "tenite" with adequate pigmentation and flow specifications. The solvent which we have found most satisfactory consists of a mixture of 40 per cent "cellosolve" and 60 per cent acetone. The 0.02-in. plastic cap appears to have a stopping power equivalent to approximately 25 mg/cm^2 of aluminum by experimental tests and thus should pass all beta radiation above 0.34 Mev.

Rings are unloaded after use by prying off the cover. The ring itself is disposable. The cadmium filter may be reused.

Adaptability features of the ring include the possibility of resizing by means of heating the band which is open at the bottom and reforming while warm to the appropriate size desired. Likewise, this band may be removed from the body of the ring, and the ring may be used as a small button.

X-10 area laboratories desiring to obtain some of these rings for their use may order them through J. C. Ledbetter, Personnel Monitoring Section, Health Physics Division, Oak Ridge National Laboratory.

(J. C. Ledbetter)

6.3 MEASUREMENTS ON INTERNAL EXPOSURE

By L. B. Farabee

The chief purpose of the Health Physics Division is to protect personnel from bodily injury due to radioactive hazards. The many branches of Health Physics coordinate into a unit, with each branch taking care of its specific duties. Instrument research and development, survey monitoring of contaminated working areas and materials, and personnel monitoring are integral parts which contribute to the above objectives. Body-fluid analyses rightly becomes a part of personnel monitoring.

Health physics surveyors in the field are constantly alert to survey areas and give advice to chemists, technicians, and laborers as to what conditions of safety from radiation exist. It is their purpose to help, instruct, and give advice to workers who are less familiar with radiation dangers. The department which carries out body-fluid analysis is the final link in the personnel-protection program. The purpose of this group is to measure internal exposure by means of analyses of bodily excretions.

The analysis of the human body shows that about every element found in nature goes to make up the body. Some are harmless in small amounts, but hazardous in excessive amounts. These elements can be detected in excretions from the body provided that enough is present so that detection can be made possible. The fact that these elements can be detected provides the basis for organization of a group in Health Physics for making such measurements.

The body-fluid analysis group in Health Physics is primarily interested in determination of radioactive materials. Therefore this group must be closely linked with the medical department and biological research. Chemical isolation of the radioactive material must first be done; then extrapolation to internal exposure is made on the basis of biological research done on experimental animals.

Since the organization of a group for body-fluid analysis at Clinton Laboratories in July 1945, the main concern has been the development of procedures of chemical isolation of radioactive elements, followed by the perfecting of these procedures to a point where they were usable. The development of instruments which are reliable has been one of the major problems. The quantity of any radioactive material to be found in either urine or feces is exceedingly small. Therefore the counting instruments must be dependable over a long period of time.

6.4 WASTE-DISPOSAL THEORY AND PRACTICE

By R. J. Morton

There are two ways in which I might discuss this problem: (1) a detailed discussion of radioactive waste problems here or (2) a more general discussion of the disposal of all types of waste here and elsewhere. My purpose is to discuss this problem in a general way so as to present a broad picture of waste-disposal problems as a whole and then speak more particularly of radioactive wastes.

When we have to deal with new problems, the tendency is to think of them as separate and entirely different from previous ones. After a while we usually find that there are many more points of similarity than we had realized—similar in principle, the same people are concerned, the same basic interests prevail, and previous experience gives valuable guidance in studying and solving.

Waste disposal is a universal problem, for example, in communities, industries, etc. It is axiomatic that communities and industries must find methods of satisfactory disposal for health, growth, and existence. Cities spend a large part of their income for sewerage, refuse disposal, and control of atmospheric pollution. Some industries have been forced to close or move. Lack of proper sewerage can often cause great health and economic damage, e.g., polluted water supplies (epidemics), streams ruined, property depreciated, odor nuisances, etc.

There is a great variety of wastes, from sewage of a city to shavings and sawdust in a carpenter shop. Typical and usual properties are listed below:

1. Domestic sewage: liquid, organic, disease germs, odors
2. Industrial wastes: liquid, organic or inorganic, no disease germs, odors, bad appearance, etc.
3. Refuse: solid or semisolid, organic and inorganic, few disease germs, odors, fires, insect breeding, etc.
4. Atmospheric pollutants: gases and dusts, organic or inorganic, no disease germs, odors, and sometimes toxic substances, etc.

This classification indicates the character of wastes to some extent, but it is based mainly upon the source and responsibility for disposal.

Wastes may be classified in other ways if more useful in treatment or disposal, e.g., organic versus inorganic; radioactive versus nonradioactive; infectious versus noninfectious, etc.

Who is concerned with problems of waste disposal ?

1. Individuals who may be damaged or annoyed
2. Owners, i.e., those who produce wastes and are responsible for damage, if any
3. Public health authorities, state and municipal health officials, health divisions of industry, and other regulatory agencies

Waste disposal does not mean just getting rid of and passing on to others; it implies satisfactory (especially safe) disposition of wastes. Waste disposal means the handling, processing, and ultimate disposal in such a manner as to prevent danger to health, objectionable nuisances, interference with the rights of others, or economic damage.

Listed below are a few examples of improper disposal:

1. Sewage containing disease germs (and all sewage may at times) discharged above a water supply or swimming place results in danger of disease transmission.
2. Excess organic matter (as from sewage, cannery wastes, milk wastes, etc.) into streams or lakes results in decay which causes odors, killing of fish, depreciation of property, interference with use of water, etc.
3. Excess of smoke, fumes, gases, chemical irritants, or dusts into atmosphere result in nuisance and damage and may cause danger to health and safety.
4. Burial of infectious, or offensive, material where it may pollute underground water and reach springs and wells, or where it may be uncovered as by washing out or tampering by people.

Available methods of waste disposal are few in number although infinite variety of conditions are involved. Some of these are listed below:

1. Mix with surface water, e.g., add industrial wastes to streams or lakes to take advantage of dilution, time for decay, die-away of germs, etc.
2. Deposit on surface of the ground within certain limits, e.g., lagoons, dumps, irrigation of land, etc., to take advantage of sunlight, drying, decay, etc.
3. Burial or absorption underground limits access of people, but may affect underground water. This takes advantage of segregation and time.
4. Discharge to the atmosphere, e.g., incineration or evaporation of solids or liquids and stacks for gas. This takes advantage of dilution: there is the disadvantage of wide dissemination and sometimes of concentration (meteorological conditions).
5. Recovery at the source for by-products, process use, or storage indefinitely. Certain constituents or the water may be valuable for use. Otherwise storage becomes more and more troublesome, expensive, and uncertain of results. Storage may be necessary for some strongly toxic chemicals and radioactive materials.

All these methods may be satisfactory, but only if the quantity of wastes is within permissible limits in order to prevent danger of damage or nuisance, for example, tolerable versus excessive amounts of sewage in water, too much smoke or gases in air, burial of more material than soil conditions can assimilate.

The guiding principle is that the quantity of the objectional constituents must not be too great for the capacity of the method of disposal employed under the most adverse conditions. The capacity of the method depends mainly upon harmful exposure of people, i.e., potential health hazard.

The purpose of waste treatment is to remove or neutralize the objectionable constituents to bring them within the capacity of the disposal method, for example, sewage bacteria in a stream near water supply or swimming place must be removed by settling, etc.; then, if necessary, kill remaining bacteria by chlorination of the sewage.

The same available methods and guiding principles apply to industrial waste disposal, although the constituents and their effects vary widely, depending upon the type of industry and its particular processes.

The criteria of proper disposal are:

1. Prevent danger to health
2. Prevent nuisance
3. Prevent interference with rights of others
4. Prevent economic damage

The present policy and practice tends toward disposal of industrial wastes together with municipal domestic sewage wherever feasible, with pretreatment, if necessary, to prevent damage to sewers or upsetting of sewage-treatment processes.

Certain radioactive wastes contain toxic chemicals which must be considered just as other toxic industrial wastes. The peculiar property is radioactivity, for example, radioactive toxicity.

Radioactive materials must not be disseminated promiscuously in considerable quantities into surface waters, underground waters, upon the soil or in the soil, or to the atmosphere unless reliable knowledge shows no present or future danger to man or other living things.

This policy sharply limits use of the natural methods of disposal so widely used for other wastes, namely, water, soil, and air, because knowledge is still inadequate, tolerance values are not well established (effects not certainly known), and factors of safety have to be ample.

Five methods of disposal of radioactive wastes mentioned by Forrest Western in a paper published in Nucleonics, August 1948, are as follows:

1. Dilution in water reduces amount of ingestion or exposure.
2. Dilution with stable isotopes limits reconcentration.
3. Retention for partial decay to reduce level of activity, as in tanks.
4. Permanent retention by various methods, including burial in earth (presumably will be retained) and concentration and retention in tanks.
5. Making into concrete (if highly concentrated activity) and disposal in oceans (really slow dilution).

All these include uncertainties and present more or less practical difficulties.

Health physicists agree that it is essential to adopt very conservative maximum permissible limits, to observe them strictly, with constant monitoring and supervision, and to carry on much more research and investigative survey work to improve our knowledge of radioactive waste-disposal practice.

General purpose and approach in waste monitoring and waste-disposal studies programs of Health Physics are outlined below:

A. Area monitoring group
  1. Purpose or general function: observations, records, and reports to see that
    a. Discharges of activity are not greater than prescribed limits,
    b. Accumulations or reconcentration of activity do not result in conditions that are actually or potentially hazardous
  2. Approach and activities
    a. Liquid wastes
      (1) Settling Basin effluent: daily sampling and continuous sampling for daily average. Limit for beta-emitting materials set at average of 5 curies/day. Recent study by Setter (January to November, 1948) shows that for 11 months the average was less than 3 curies/day

(2) White Oak Dam: daily sampling, e.g., floods; and more frequent series when needed. No exact limit in curies is prescribed. Permissible discharge complex and related to flows in Clinch River

(3) Special and periodic sampling: down Clinch River monthly for water, muds, algae, etc; White Oak Creek and Lake muds monthly with complete survey annually. Examinations of algae, fish, etc., from creek or lake as indicated (for reconcentration of activity)

b. Atmospheric, routine monitoring: continuous recording monitors (3 outside and some 10 to 12 inside various buildings). Number of outdoor stations to be increased (7 ± new). Routine observation of weather instruments. Reports include major factors, e.g., wind, inversions, rainfall, etc.

B. Waste-disposal studies program (Waste-disposal Section)

1. Program of research and fundamental studies

a. Survey studies concerning conditions that exist and behavior of radioactive materials in water, soil, air, plants, and animals (sampling and analysis activities)

b. Laboratory studies of treatment processes (sanitary engineering approach, i.e., health protection) include water-treatment processes, sewage and industrial-waste treatment processes, and other related methods of treatment or disposal.

6.5 DECONTAMINATION LAUNDRY

By W. H. Ray*

Laundering of contaminated clothing started during the war. Originally protective clothing was provided for maintenance men working on contaminated equipment. The clothing was then to be discarded. As a result of shortages during the war, clothing was difficult to replace; therefore laundering was attempted.

Paul Eisenacher, a trainee, began experimenting in a beaker with decontamination solutions. He found in the Physics-Chemistry Handbook that most of the acetates of the rare earths were soluble. Thus, we started using acetic acid for decontaminating clothing. The clothes were washed several times and then rinsed several times before being finally released.

The next problem was to set up a regular routine for inspection of garments with Geiger counters. We were concerned only with reducing the activity of garments to such an extent that they would not be harmful. It was later that we worried about the intake of alpha into the body.

Acetic acid proved to be irritating, and the odor was objectionable. It subsequently was found that citric acid was as good or better and would also work with Pu. In addition, the garments lasted longer. We are still using citric acid solutions.

The concentration of solution used and the number of washes are determined by the degree of contamination on the clothing.

If contaminated garments are just dumped together with relatively clean ones, there results a transfer of activity from one to another. Some materials have dyestuffs that have an affinity for contaminants. A very contaminated white laboratory coat will come out of a wash much cleaner than a virgin pair of coveralls. Cross contamination is also possible among garments of the same color. To minimize this transfer, it is desirable to segregate the clothes into categories—cool, warm, and hot.

It is not reasonable to launder garments that are not contaminated in the same laundry where contaminated clothes are washed. It is better to have separate laundries. Contaminated garments should not be sent to public laundries.

The life of a garment should also be taken into consideration in laundering. One washing does not necessarily decontaminate a garment. Strong solutions shrink the material and shorten the life of garments.

Is it worth while to launder garments? It is a question of economics, and it is quite possible that with careful study given to the way garments are used so that they would be only contaminated

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*Notes taken by C. Scott; edited by W. H. Ray.

when necessary, severe contamination could be kept to a minimum, and it might not pay to have the investment in laundry equipment. Cheap and effective expendable garments may be devised that are to be worn only once.

One way of salvaging garments is to put those garments with short-lived radioactive contamination in storage so that the activity may decay. It is debatable whether such a storage procedure is economical. It does not seem feasible to hold garments for more than 6 months. At one time shoes that were confiscated were kept for 6 months or more, but, invariably, when the shoes were safe to give back, they were not fit to wear or the employee had terminated.

A garment's life is about 15 to 25 washes if the cases of contamination are relatively mild.

There is the additional cost of checking, pickup and delivery, and segregation. Garments are now assigned to individuals. Originally they were assigned to operating areas. Protective garments should be written off the inventory when issued. Here at Oak Ridge where we have an "academic atmosphere," the clothes must be pressed. This adds to the cost.

Fifteen to 20 per cent of the clothes actually have a decontamination wash. Contaminated clothes are given three washes before casting them out if they fail to respond.

There is also a problem of laundry waste and laundry fumes. Garments which the laundry has been unable to decontaminate satisfactorily are buried. Storage and checking add to the cost considerably and may not be economical.

Monitoring Techniques. Hanford uses an adaptation of the fourfold counter. This is for gamma and hard betas. Alpha contamination is best checked by Poppy. Depending on the thoroughness of inspection, it requires from 4 to 30 min to check a pair of coveralls with Poppy.

6.6 HEALTH PHYSICS PUBLIC RELATIONS*

By W. H. Ray

I. Health Physics objective: reduce radiation injury to the individual and to the race to a minimum
II. Means of attaining objective
  A. Education
    1. Intelligent people when informed can do much to take care of themselves
    2. Many can be made to feel a moral responsibility for prevention of overexposure to others
    3. Propaganda can develop beneficial protective habits through mob psychology
  B. Guidance
    1. Surveyor surveillance
    2. Professional consultation
    3. Procedures and standards
  C. Coercion
    1. Rules and laws
    2. Line organization and directives
III. Techniques of applying means
  A. Salesmanship
    1. Character
      a. Absolute honesty
      b. Dependability
      c. Keep confidences
    2. Magnanimity
      a. Serve others
      b. Do more than is expected
      c. Do not "pass the buck." Feel responsible
    3. Personality
      a. Alertness

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*This is substantially the same lecture given in a previous Health Physics Survey Course by W. H. Ray, November 24, 1947.

        - b. Prompt action
        - c. Patient explanation to extent needed
        - d. Intelligence
    - 4. Cooperation
        - a. No backbiting
        - b. Mesh with efforts of others on your side
        - c. Be open minded enough to use the good ideas of others
- B. Dominance
    - 1. Maintain leadership in knowledge of field
    - 2. Report to top authority (in T.O.)
    - 3. Be practical as well as ideal

IV. Measuring success of Health Physics

- A. Any job can be done safely under any tolerance level that may be established if the will to do so, time, and funds are available
- B. Failures may not show up for generations or may not be recognizable. Also false evidence may be hard to disprove.

UNIT 7

(Since Unit 7 dealt primarily with the work of the laboratory, as mentioned in the Preface, it is not included.)

UNIT 8. SOME PRACTICAL PROBLEMS

8.1 GUARDING HAZARDOUS AREAS AND CONTAMINATION CONTROL

By G. C. Cain*

All hazardous areas should be promptly identified, marked, or guarded as soon as hazards occur to keep personnel from being over exposed and to keep other areas from becoming contaminated. That brings up the question of what a hazardous area is. From the point of view of radiation protection, any area is to be considered hazardous if the level of radiation or the degree of contamination is such that, over long periods of time, any person working in the area might be expected to receive radiation exposures above the established maximum permissible levels. A wide-awake survey will, in many cases, anticipate a hazardous area or contamination, even before it occurs, from experience and knowledge of the operations involved.

The maximum permissible levels for an occupied area are as follows: beta and gamma radiation† of 12.5 mrep/hr; alpha-emitting contamination of 3000 dis/min in an area of 150 cm^2, or the minimum amount of alpha material which can be detected by use of the "smear" technique; a thermal neutron flux of 1600 neutrons/cm^2 sec; a fast neutron flux of 200 neutrons/cm^2 sec; and atmospheric contamination of 10^{-8} $\mu c/cc$ for beta-emitting materials and 3×10^{-11} $\mu c/cc$ for alpha-emitting materials. If two or more types of radiation are present, the combination shall not produce an exposure to any part of the body biologically equivalent to more than 12.5 mrep/hr† of beta or gamma radiation.

Radioactive materials may present special hazards in case of fire or other accidents. For the protection of firemen, areas containing radioactive materials which may become hazardous in case of fire are marked by use of an 8-in. magneta plaque. Firefighting procedures require the use of masks in certain designated areas until it has been determined by authorized personnel that no significant inhalation hazard exists. As an emergency procedure, firemen may work in levels of radiation as high as 20 r/hr and may be permitted to receive single exposures as high as 5 r.

All the maximum permissible levels should be considered just so—as maximum permissible levels. They are not tolerable unless unavoidable. Our job is to try to see that people work in areas as free of radiation and radioactive contamination as possible.

The majority of radioactive materials originate in the pile building. They are usually sent from there to the other buildings. Some conditions under which guarding requirements arise are (1) active samples from the pile, (2) active products from other areas, i.e., isotopes, (3) transfer and storage of smaller active samples in the laboratories, (4) storage and use of sources, (5) transferring hot waste to the burial ground (may need patrol escort), and (6) shipments to X-10 from other sites of active wastes.

Ways and means of guarding or identifying these areas bring on a lot of discussion in determining just how areas should be marked off. There are small radiation-danger tags for individual small samples, contaminated tools, and equipment. We can use signs of a temporary or permanent nature. If it is temporary, then by all means describe the level at which the sign is placed and also the maximum level at the source of radiation. Those signs should be brought up to date each day. There are

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*Notes taken by C. Scott; edited by G. C. Cain.

†At the International Conference on Radiation Protection held at Chalk River, September 29-30, 1949, a maximum permissible rate of exposure of 0.3 rep/week was recommended.

many cases where signs are left for weeks and months, and personnel begin to pay little or no attention to them. If they are kept up to date, personnel will respect them. Rope barriers, i.e., rope off an area and put tags on it. In all cases you want to rope the area or barricade off at or under the maximum permissible level. In the case of a permanent hazard, put up a permanent barrier to keep people out of the area. You might lock existing doors and place signs on them and give supervision the responsibility for locking or unlocking the doors. In an extreme emergency you can call out a patrolman, and he will stand at the door and keep people away from it. You might in some cases paint lines on floors to distinguish between contaminated areas and areas free of contamination. This will aid materially in preventing the spread of contamination.

Contamination Control. This particular part of the surveyor's work in the field takes up more of his time than any other one thing, even radiation control. The best means to control contamination is to prevent it in the first place. If you prevent it, you have it licked right then. In many cases, and in the majority of cases, if you know your area, you will know, in general, in advance when to expect contamination and can guard against it before it occurs. In some cases contamination cannot be prevented, but it can be kept at a minimum. Try to arrange consultations with the person who is handling the area even before he attempts the job. In many cases he will come to you. The jobs that cause you the most trouble are the little ones. The big ones are carefully thought out, and, as a result, fewer accidents happen. On the small jobs that normally occur in that area, personnel usually will not come to you and ask. It is up to you to find them.

Other than preventing contamination before it occurs, perhaps the next best way to control it is to conduct thoroughly regular routine surveys of the building. It is a monotonous job, but usually you come upon something unexpected. Many hazards have been found in this way.

I believe it is safe to say that in most areas the protective clothing rules and regulations are good enough if they are enforced. In most areas protective clothing is available. Friskers, in many cases, have helped to control contamination. Instruments help control contamination. Monitrons and integrons, when showing a high level of activity, have quite often been the only indication of a spill. If the employees would take showers and clean up before leaving the area, it would help to keep contamination localized. Anything going to the shops, stores, or outside the area should be checked before leaving.

We have had cases where a job has been carried out without a single exposure. This is a result of the Operation's Superintendent calling meetings to plan the work and following through on all recommendations submitted by the Health Physics Division. This saves exposures, contamination, and cleanup.

In order to complete the job of guarding hazardous areas and controlling contamination, you must first know your particular area. Another requirement is to know the men you work with. The better you know the men, the better you can get your particular job done. Know the rules that apply to your area. Above all, do not undersell your job.

8.2 FULFILLING CLOTHING NEEDS

By A. D. Warden and J. T. Sutherland

E. L. Sharp gave a brief discussion of protective clothing in another lecture. I shall attempt to go into the protective clothing needs a little more in detail.

The words "protective" for clothing in this particular case is a misnomer. The words "expendable clothing" might be a better term to use. We may get a small amount of protection from soft betas, but we cannot expect clothing to stop very penetrating radiation.

Let us look at protective clothing in this light. "It is much easier to peel off clothing than it is to peel off the skin." Protective clothing may be easily replaced. Some types of contamination are extremely hard to remove from the skin. We have had a few cases of hand contamination where it was necessary to scrub the hands with various solutions until the hands were practically raw, and yet we could not remove all of the contamination.

In addition to the protective clothing normally worn by the personnel in the areas that have been discussed, extra items of wearing apparel such as overshoes, shoe covers, gloves, and caps are kept in limited numbers for messy emergency jobs.

1. The overshoes are not satisfactory at present.
2. Overshoes are used in liquid decontamination jobs.
3. Shoe covers are used in areas where walking surfaces are dry.
4. The canvas gloves are used normally where objects to be handled are dry.
5. Rubber gloves of all types are used in liquids.
6. Gloves and cloth caps are expendable and are seldom usable after being laundered.

Lab coats are frequently worn by supervisors in the various areas. A large number of supervisors feel that they do not work directly with the hot material enough to warrant the wearing of protective clothing. The lab dresses and slack-coveralls are worn by the female employees in the various laboratories.

Protective Clothing for Maintenance Men

The maintenance personnel are not furnished with free issue shoes or coveralls. It is up to the Health Physics Division to recommend on work orders the wearing of shoe covers or overshoes. When the maintenance man does get his personal shoes contaminated over the tolerance value, the company will issue him a pair of safety shoes, and we shall confiscate or store his own shoes. If a hot job is to be performed, the Area Supervisor in the area where the job is to be performed will issue the maintenance men coveralls, gloves, overshoes, caps, etc. The health physics men probe the maintenance men's personal clothing carefully because that is principally what they wear at work.

Protective Clothing: Spreader of Contamination

Although we use protective clothing to keep the radioactive materials from our skin, it is one of the greatest spreaders of contamination that we have. Shoes track the contamination all over the plant. The gloves we wear can contaminate hand rails, telephones, chairs, etc. Some areas have containers for grossly contaminated clothing and keep this separate from mildly contaminated clothing. This helps protect laundry personnel.

Protective Clothing Program at Present

Our protective clothing program at present is only about 33 1/3 per cent efficient for the following reasons:

1. There are not enough regulations concerning the necessity of wearing protective clothing
2. Enforcement of what rules we have is very lax
3. Some supervision professes only weak interest in the protective-clothing program

How the Protective-clothing Program Can be Remedied

The protective-clothing program can be remedied by zoning as follows: (1) cold areas, (2) cool areas, (3) warm areas, and (4) hot areas. The clothing requirements should be set up for these areas, and then supervision should be allowed to enforce the rules with common sense and a little psychology.

Laundry in Relation to Protective Clothing

Clothing Stores personnel mark all garments prior to issue with the station number followed by the badge number with nonwashable ink. This is the means of identifying each garment.

When protective clothing is picked up, it is delivered to the laundry receiving station. There the clothing is listed by station and badge number and passed on, in hampers, to the Health Physics Division checking station.

Health physics technicians make a prewash check—beta-gamma always and alpha in some instances. Eighty-five per cent of the clothes pass this prewash check, are sent to the laundry for a normal laundry wash, and are sent back to the appropriate laundry station. The 15 per cent that fails to pass is sent to the laundry for a decontamination wash and then sent back for another check.

If a garment has not passed after three decontamination washes, it is taken out of service. This is one way a person may expect to lose protective clothing. Less than 1 per cent of clothing handled is lost for this reason. When clothing is found to be free of a harmful amount of contamination, Mr. Taylor may decide that some garments are beyond repair and take them out of service. This is a second way a person may expect to lose his protective clothing.

A list is made up weekly of clothing taken out of service because of contamination, and a list is made up daily of clothing taken out of service because they are beyond repair. Copies of these lists are sent to Clothing Stores.

Clothing Stores personnel, treating each type of list in the same manner, do the following:

1. Credit the man's account with the garment listed
2. Check stock for a replacement garment of the right size
3. Debits the man's account with the replacement garment
4. Marks the garment with the appropriate numbers
5. Sends the garment to Mr. Taylor at the laundry for distribution

If the garment of correct size is not in stock, the clothing stores personnel sends notification to Mr. Taylor and to the employee concerned saying that a certain garment, or garments has been taken out of service because it is contaminated or because it is beyond repair, his account has been credited with same, and this notification slip will serve to draw this clothing when it is in stock.

Other ways an employee may lose his protective clothing are:

1. Another employee takes his clothing
2. Changing laundry stations and neglecting to have station numbers changed on clothing
3. Numbers misread by someone handling the clothing
4. Employee is deprived of clothing longer when more than one wash is necessary for decontamination

Trial storage is in effect. Because of the small number of garments involved and the low value of these garments, the expected saving is small and necessitates a very economical procedure. A hutment, 15 by 15 ft, is being used. Ten bins have been constructed at very little cost. There will be a 20-week storage with checks made at the end of 4, 10, and 20 weeks. The use of colored tags for marking garments individually will permit using a single bin for more than one weeks collection. Mildewing may present a problem. It will take at least several weeks to determine whether such a storage system will be economically worth while.

8.3 WEAR AND CARE OF MASKS

By H. J. McAlduff

For many years workers in the dusty or smoky trades attempted to relieve the discomfort of working under these conditions by wearing cloths or handkerchiefs over the nose and mouth, but it was not until 1919 that the systematic development of modern respiratory protective equipment began.

A respirator or mask is a device for rendering inhaled air respirable, and these may be classified as follows:

I. Supplied-air respirators
  A. Self-contained type
    1. Oxygen-breathing apparatus
  B. Hose type
    1. Air-line respirator
II. Air-purifying respirators
    1. Air is purified by drawing through chemicals which remove the contaminant by chemical reaction
III. Mechanical-filter respirator
    1. Removal of dust, fumes, or mists by suitable filters

IV. Chemical- and mechanical-filter respirator

1. Combination of chemical and mechanical removal of contaminant

Important considerations in the selection of any type respirator would include: low resistance to breathing, definite degree of protection under well-defined conditions, comfort in wearing, lightweight, and little interference with vision.

Masks should be intended primarily for emergency use, or for use in conditions not applicable to control by other means, but have definite limitations in their use which should be carefully observed.

Respirators at the laboratory have been chosen for use in those situations where the inhalation of radioactive materials above prescribed maximum permissible values is a possibility. These values at the present time are: 1.0×10^{-8} μc/cc of air (for long-lived β, γ emitters) and 3.0×10^{-11} μc/cc of air (for long-lived α emitters).

The Oak Ridge National Laboratory Fire Department has standardized on two masks: the Chemox, which is a self-contained oxygen unit, for use inside buildings and the Army assualt mask for use outside buildings.

The Health Physics and Safety Departments both function in an advisory capacity as to the type of respiratory equipment necessary for protection against the various hazards encountered about the plant area. The Health Physics Division recommends the protection necessary in those situations where the inhalation of radioactive materials is a possibility, while the Safety Department recommends the proper type of protection for those operations not involving the use of radioactive materials. Protective equipment chosen for use at the laboratory has been selected in most cases to furnish protection against both chemical contaminant and radioactive materials.

Situations in the field requiring respiratory protection are varied and depend on the particular area involved. In all instances where air contamination is above masking tolerance levels, masking or evacuation of personnel is mandatory. One disadvantage confronting the surveyor is the length of time necessary in running and counting an air sample, using the precipitron, to determine the activity level. It is dependent on the surveyor's good judgment and knowledge of operations in his area, therefore, as to whether to mask or evacuate personnel in the interim while the actual activity level is being determined. Discretion is necessary also in the use of protective equipment where its use might interfere with a man's safety. For example, a job requiring but 1 or 2 min to complete in a high-radiation zone where air contamination is a possibility might be better accomplished without the use of the mask, if its use would affect the workers efficiency to the extent of requiring a longer period of time in the high-radiation zone to accomplish the work. In like circumstances, if the use of a mask interferes with a man's movements, increasing the hazard of slipping or falling, its use might not be justified.

Respirators and masks should be stored in a clean dry location. Sterilization after use is usually done by the health physics surveyor, using 70 per cent ethyl alcohol. Contaminated masks are also decontaminated by the surveyor. Laundry decontamination facilities are available for masks extensively contaminated.

The number and types of masks and respirators should be held to the minimum necessary to furnish protection needed in the various areas. These needs are best determined by the senior surveyors in the respective areas. Army assault masks will furnish protection against most types of dusts, fumes, smokes, and mists. Where oxygen lack is suspected, or when absolute protection is needed, the Chemox, Scott Air Pack, or an airline respirator should be utilized. For chipping, grinding, or abrading solid materials, one of the lightweight respirators, approved for dusts or fumes, may be used.

8.4 SHIPMENT OF RADIOACTIVE MATERIAL*

By T. H. J. Burnett

The material covered was substantially identical to that contained in an article by Dr. Karl Z. Morgan on this subject, appearing in the Journal of Applied Physics, July 1948.† Copies of reprints of this article were distributed to the students.

In addition, the effects of bremsstrahlen were cited as means whereby the shipment of strong beta emitters could be accompanied by associated X or gamma rays in the energy range such as to have a k factor greater than 1 and possibly induce film damage. Bremsstrahlen was explained as a term of German origin meaning literally "braking rays" and constituting the radiation of photons accompanying acceleration changes of the beta electrons as they are deflected by interaction with matter.

An experimental evaluation of the effective k factor proper to use in shipping isotopes emitting gamma rays of more than one energy was described, and the results of these tests were given. No case was found where the k factor exceeded 2, this result being due to the effectiveness of the shielding in cutting out the low energies when sufficient to handle the higher energies also present.

A brief discussion of causes for intensification of film readings behind the badge shield was given, illustrating the difficulties inherent in interpreting the open-window reading in some cases. The conclusion emphasized again the importance of the professional investigation made by the health physics surveyor in all indicated cases of possible exposure.

8.5 SHIELDING

By R. S. Thackeray

The use of shielding in work involving radioactivity is required to limit personnel exposure to the established tolerances. Shielding may be broken down, in general, into the following categories:

1. Permanent shielding
2. Portable or flexible shielding
3. Close shielding

Gamma-ray shielding is done with heavy masses of material which have a high density. The radiation is never reduced to zero by such shielding, but it can be reduced to safe levels. Permanent shielding against gamma rays usually takes the form of thick walls of concrete. Sometimes space requirements dictate a special high-density concrete, but normally a large thickness of ordinary material will suffice.

Portable shielding is achieved through the use of specially designed containers of heavy metal such as our source pots.

Flexible shielding is erected around hot samples, using lead bricks. Gamma rays are, in general, too penetrating to allow the use of close shielding.

There is seldom any need to make permanent or portable shields for alpha, beta, and protons. Close shielding, that is, shielding built closely around the vessel containing the source, is most useful with these radiations. In this way all the radiation can be cut out through the use of a suitable thickness of plastic, because each of the radiations mentioned has a definite maximum range corresponding to its energy. Close shielding permits one to handle beakers, test tubes, etc., in a more or less normal fashion.

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*A report on the physical, biological, and administrative problems associated with the transportation of radioactive substances is under preparation by Robley D. Evans for the Subcommittee on Shipment of Radioactive Substances, Committee on Nuclear Science, Division of Mathematical and Physical Sciences, National Research Council. It is expected that this report will be issued in 1950. The report gives an excellent and comprehensive treatment of the materials indicated by the title and will be of interest to all persons handling radioactive materials.

†Some changes and extensions of shipping regulations have been made since publication of the article referred to above, and other modifications are in progress.

The problem of shielding against neutrons is more complicated. Fast neutrons are very poorly absorbed by most materials. It is therefore necessary to slow them down to speeds at which they are absorbed efficiently. For this reason a neutron shield usually consists of a mixture of hydrogenous material with a neutron-absorbing substance which gives off nonpenetrating secondary radiation. A typical example is paraffin containing borax. The hydrogen atoms in the paraffin slow down the neutrons, and the boron absorbs them while giving off an alpha particle. The alpha particle, of course, has a negligible range in the paraffin.

For small-scale laboratory work it is much more convenient to pile shielding around a source until the measured radiation is below tolerance limits than to try to calculate accurately the amount of shielding required. The measurement will account for all the leakage radiation and scattered radiation much more simply than any calculation can.

In erecting shielding it is essential to remember that shields cast shadows and that if you stand outside of the shadow you are receiving radiation exposure. It is necessary to remember that radiation can effectively bounce around corners, although such scattering reduces both the intensity and energy of the primary rays.

In agreement with the standard laboratory policy that health protection responsibility rests with supervision, the Health Physics Division has no formal responsibility for the adequacies of shielding unless it is specifically invited to make recommendations. However, any dangerous situation discovered should be reported immediately to supervision, and, if it is not corrected, it should be reported to Survey Headquarters.

Appendix A. Supplementary Study Material for Lectures 2.6 and 8.5—Preliminary Notes on Radiation and Shielding Problems

Dosage Rate from an Unshielded Point Source of Gamma Radiation. Within the range of energies 0.2 to 1.5 Mev, the dosage rate at 1 ft from an unshielded source is given approximately by

$$R_1 = 6\ C\ E$$

Where R_1 is the dosage rate in roentgens per hour at 1 ft, C is the number of curies of the radioisotope emitting the gamma radiation, and E is the gamma energy per disintegration.

For a wider range of energies the following table, giving the dosage rate per curie in roentgens per hour at 1 ft, may be used:

Table 8.1

E,* Mev	R_1/curie, r/hr at 1 ft	E,* Mev	R_1/curie, r/hr at 1 ft
0.1	0.5	1.5	8.2
0.2	1.1	2.0	10
0.3	1.8	2.5	11.6
0.5	3.1	3	13
0.8	5.0	4	15
1.0	6.0		

*One quantum of this energy per disintegration is assumed. If the number of quanta per disingegration is different from 1, or if more than 1 energy is present, the value of R_1 is adjusted accordingly. Dosages, not necessarily quantum energies, are additive. In the absence of more definite information, it is sometimes assumed that the radiation is equivalent to 1 quantum per disintegration of the most energetic gamma known to be present. This assumption may result in an estimate that is either too great or too small.

Dosage Rate from a Radium Source. For a small radium source, within a 1 mm platinum shield, the dosage rate, measured in roentgens per hour at 1 ft, is given by

$$R_1 = 0.0084M$$

where M is mass of radium measured in milligrams. (The dosage rate at 1 ft from an unshielded 1-mg source is about two-thirds of 8-hour tolerance.)

Dosage Rate as a Function of Distance from a Point Source. The above relations give the dosage rate at 1 ft from an unshielded point source. If shielding and scattering are negligible, the dosage rate at any other distance is found by dividing the value at 1 ft by the square distance measured in feet.

$$R = R_1/D^2$$

Shielding of a Point Source of Monoenergetic Gamma Radiation. For monoenergetic sources, shielding problems are simplified by the fact that, in most cases, the fractional reduction in the dosage rate produced by a layer of shielding material is independent of the amount of shielding that has already been used. To use this property to the best advantage, we define and make use of the concept of a tenth-value layer (TVL) of shielding material. A TVL is the thickness of material which will reduce the dosage rate of the radiation passing through it to one-tenth of its incident value. It follows that two tenth-value layers will reduce the dosage rate by a factor of 1/100; three TVL's by a factor of 1/1000; six TVL's by a factor of 10^{-6}, etc.

It is sometimes useful to know the thickness of a half-value layer; i.e., the thickness necessary to reduce the dosage rate of the radiation passing through it to one-half of its incident value. One half-value layer is equivalent to 0.30 tenth-value layer.

One of the parts of Table 8.2 gives the thicknesses of tenth-value layers of different materials as a function of the energy per quantum of gamma radiation. From the preceding discussion, it will be apparent that a TVL of one material will be equivalent to a TVL of any other material for the gamma energy for which they are given. The values given for concrete are for poured concrete with limestone aggregate. They are satisfactory for cement blocks of the same weight per unit volume, but for the many cement blocks of lesser density, proportionally greater thicknesses are required.

Table 8.2—Absorption of Gamma Radiation

Energy		Mass absorption coefficients, cm^2/g				Linear absorption coefficients, per cm			
Mev	A	Pb	Fe	Al	H_2O	Pb	Steel	Al	H_2O
0.2	0.062	0.90	0.17	0.126	0.14	10	1.30	0.34	0.14
0.5	0.025	0.15	0.080	0.080	0.090	1.7	0.61	0.22	0.090
1.0	0.0123	0.060	0.058	0.058	0.67	0.67	0.44	0.16	0.067
1.5	0.0082	0.050	0.052	0.052	0.057	0.56	0.40	0.14	0.057
2.0	0.0062	0.046	0.043	0.043	0.048	0.51	0.33	0.12	0.048
2.5	0.0049	0.042	0.040	0.040	0.042	0.47	0.30	0.11	0.042

Table 8.2, Continued

Comparative thickness to give equivalent absorption

Energy, Mev	Pb	Steel	Al	H_2O
0.2	1	7.7	29	71
0.5	1	2.8	7.7	19
1.0	1	1.5	4.2	10
1.5	1	1.4	4.0	10
2.0	1	1.5	4.3	11
2.5	1	1.6	4.3	11

Table of specific gravities

Aluminum	2.7
Limestone	2.7–3.0
Concrete	2.7
Iron (gray cast)	7.1
Steel	7.6
Lead	11.2

Tenth-value layers*

Energy, Mev	Lead		Steel		Concrete† or aluminum		Water	
	In.	Cm	In.	Cm	In.	Cm	In.	Cm
0.2	0.09	0.23	0.69	1.8	2.7	6.8	6.5	17
0.5	0.60	1.7	1.5	3.8	4.1	10.5	10	25
1.0	1.4	3.4	2.1	5.3	5.7	14	14	34
1.5	1.6	4.1	2.3	5.8	6.5	17	16	40
2.0	1.8	4.5	2.7	6.9	7.6	19	19	48
2.5	1.9	4.9	3.0	7.6	8.4	21	22	55

*One half-value layer = 0.30 tenth-value layer.

†See note in preceding paragraph on use of cement blocks.

Shape and Thickness of Shields. Example 1, below, illustrates the application of the above values to a spherical shield about a geometrically small source. If the shield is flat and relatively close to the worker or object being shielded, an additional thickness of perhaps 1 half-value layer will be required to reduce the radiation level to the desired value.

Examples.

1. It is desired to reduce the dosage rate at a certain position from 750 r/hr to less than tolerance (i.e., to less than 12.5 mr/hr). The energy of the radiation is thought to be 0.5 Mev. What thickness of spherical lead shield will be required?

Solution: Express the existing and maximum desired radiation in the same units of measurement; then determine by what power of 10 the first must be divided to reduce it to a value less than the second. In this case 750 r/hr = 750,000 mr/hr. Division by powers of 10 is most easily accomplished by shifting the decimal point. To reduce 750,000 to less than 12.5 is accomplished by shifting the decimal point 5 places to 7.50000. Then 5 TVL's will reduce the radiation to 7.5 mr/hr. From the table one finds that, for 0.5 Mev, 1 TVL of lead has a thickness of 0.60 in.; therefore the thickness of lead required is approximately 3 in.

2. Twenty curies of radioisotope emit one 1.5-Mev gamma and one 0.5-Mev gamma per disintegration. (a) What is the dosage rate at 15 ft? (b) What thickness of spherical lead shield will reduce the radiation to 5 mr/hr at 18 in.?

Solution: (a) At 1 ft the dosage rate due to the respective gamma energies is:

0.5-Mev gamma, 3.1 r/hr/curie × 20 curies = 62 r/hr

1.5-Mev gamma, 8.2 r/hr/curie × 20 curies = 164 r/hr

At 15 ft the dosage rates are 1/225 times as large. Total dosage rate at 15 ft, 0.276 r/hr plus 0.729 r/hr equals 1 r/hr.

(b) For a first trial, neglect the lower energy radiation which is absorbed much more rapidly than the other. At 1 ft the dosage rate due to the 1.5-Mev gamma is 164 r/hr, and at 1.5 ft it is 73 r/hr. Four TVL's will reduce this to 7.3 mr/hr, and an additional half-value layer will reduce it to 3.7 mr/hr, which is sufficiently close to 5 mr/hr for our purpose. For 1.5-Mev radiation through lead, this is equivalent to 4.3 TVL's times 1.6 in. per TVL, or 6.9 in.

In checking on the effect of the neglected 0.5-Mev gammas, one finds from the table that, for 0.5-Mev gammas, 6.9 in. of lead is approximately 11 TVL's, from which it follows that their contribution is practically zero. However, in many cases the contribution of the softer radiation will not be negligible, and the estimate of shield thickness must be adjusted to take it into account.

Limitations of the Exponential Law in Shielding from Gamma Radiation. In the above examples it was assumed that the number of quanta lost from the beam in traversing the shield was the number reacting with the material of the shield. This is true only for very narrow beams and for very thin shields. If the beam is wide, quanta from other parts of the beam are scattered toward the position of interest; and, if the shield is thick, significant numbers of quanta scattered out of the beam may reach the point of interest through repeated scattering. The magnitude of these effects depends upon the geometry involved and cannot be simply predicted, but for a thick shield the emergent radiation may be several times that predicted by the simple exponential law, using the absorption coefficient applicable to narrow beams and thin shields. In practice one may include the uncertainty involved in an appropriate safety factor. Measurements by J. H. Roberson, on the attenuation of a wide beam of a hard gamma radiation in lead, indicated that, for the geometry used

$$R/R_0 = e^{-\mu x}(1 + 0.2x)$$

where x is measured in centimeters. While this equation cannot be considered applicable to other conditions than those under which the measurements were made, it is given here to give some indication of the nature and magnitude of the error involved in the simple exponential law on which the use of tenth-value layers is based.

8.6 DECONTAMINATION OF PERSONNEL AND EQUIPMENT

By H. A. McClearen

Part I

I. Common contaminants
  A. Uranium
  B. Plutonium
  C. Fission products
    1. Ruthenium
    2. Cerium
    3. Strontium
    4. Columbium
    5. Zirconium
    6. Xenon
    7. Krypton
    8. Barium
    9. Iodine
    10. Tellurium
    11. Rare earths
  D. The degree of decontamination required depends upon two factors
    1. Permissible personal exposure after decontamination
    2. Probability of the spread of contamination
  E. The decontamination procedure depends upon
    1. Nature of the material to be decontaminated
    2. Chemical composition of the contaminant
II. Decontamination reagents
  A. Water and steam
    1. Cold and hot water
    2. Steam sparging (high temperatures and agitations)

- B. Acids and salts
  - 1. Increase solubility of the contaminants
  - 2. Complex the contaminants
  - 3. Remove contaminated surface by corrosive action
  - 4. HNO_3, solutions varying from a 30-min boil in a 70 per cent solution to dipping them in 10 per cent solution at room temperature
  - 5. H_2SO_4, HCl, and H_3PO_4 are usually poor agents because of corrosive actions
- C. Organic acids
  - 1. Citric, oxalic, and tartaric acids are considered good reagents on corrodible materials
- D. Bases
  - 1. NaOH and Na_2CO_3 used on corrodible surfaces
  - 2. NaOH and Na_2CO_3 used as follow up on HNO_3
- E. Soaps and detergents
  - 1. Zepco (commercial name), good foam action when used with citric acid
  - 2. Sterox (nonfoaming agent), used where air-borne contamination might be present, i.e., exhaust fans
- F. Organic solvents and oils
  - 1. Electronic instruments, acetone
  - 2. Waxed and painted surfaces, acetone, CCl_4
  - 3. Active dusts have been removed by using Spindle Oil, or by a soap solution of glycerine, followed with a damp cloth
- G. Common-ion solution
  - 1. A specific radioactive contaminant may be removed by using a solution of its nonradioactive isotope
    - a. KI removes I^{131}
    - b. Na_3PO_4 removes P^{32}
    - c. Barium salts may remove Ba contamination
- H. Short-lived products may be left to decay

III. List of materials and suggested reagents to be used in the decontamination of a particular material

IV. Use, storage, and disposal of materials that are unsuccessfully decontaminated

- A. Burial ground
  - 1. Stored above ground
    - a. Process tanks
    - b. Machinery
    - c. Lead
    - d. Concrete blocks
    - e. Red cans
  - 2. Buried material
    - a. Highly contaminated equipment
    - b. Contaminated obsolete equipment
- B. Salvage yard
  - 1. Valuable contaminated equipment
  - 2. Slightly contaminated scrap metal, less than 5 mr/hr
  - 3. All materials going to salvage yard should have green tag on them
- C. A large percentage of equipment stored in the burial ground is reused; after some decay and subsequent decontamination materials are safely used
- D. Clarification of contamination procedures
- E. Equipment to be excessed
  - 1. Other AEC contractors
  - 2. Other government agencies
  - 3. Roane-Anderson for disposal through commercial channels; limits—beta-gamma, twice background on V-263 at their scrap yard; alpha, none detectable with Poppy or Zeuto

- a. Scrap metal
- b. Machinery
- c. Clothing
- d. Furniture
- e. Mobile equipment

V. Contamination control, decontamination, of equipment in low activity level areas (area outside Restricted Area)
  - A. Frequent surveys in all maintenance shops
  - B. Cafeteria surveys
    - 1. Building smeared and probed
    - 2. People wearing protective clothing are forbidden to eat in the main cafeteria

Part II

I. Personnel decontamination, general
  - A. All personnel should be checked following all jobs that involve work with radioactive materials
  - B. Friskers in all buildings and at all area gates should be kept in good working order so that employees who are unknowingly contaminated may be detained
  - C. Probe personnel quickly to determine the extent of contamination
  - D. If the employee is generally contaminated have him remove his clothing and take an immediate shower; If hands are badly contaminated have gross contamination removed before the shower is taken

II. Hand contamination
  - A. Study the Hand Decontamination Procedure enclosed in the Health Physics Hand Decontamination Kit
  - B. Employees whose hands are unsuccessfully cleaned should be sent to Medical

III. Body contamination
  - A. Health Physics Hand Decontamination Kit contains the necessary reagents
  - B. Hair contamination is the main problem

IV. Medical correlation with Health Physics
  - A. Contaminated wounds
  - B. Hand contamination not reduced
  - C. Ingested active materials
    - 1. Feces
    - 2. Urine
  - D. Blood-count checks made on employee receiving high dosage rate

Appendix A: Hand-Out Material Suggested Reagents; Their Use for Alpha, Beta, Gamma Decontamination

Material	Treatment
Stainless steel	HNO_3 washes, varying concentration and temperatures, followed by Na_2CO_3
Mild steel	10% citric acid, 5% Zepco soap powder, using stiff scrub brushes, followed by rinse
Lead	HCl preferable to HNO_3, plenty of rinse water, hoods desirable to remove fumes
Brass	Acetone wipes with kleenex to emery cloth, ammonium citrate, trisodium phosphate, 6N citric acid
Aluminum	10% HNO_3 with vigorous brushing, synthetic detergents
Glass	Small glassware, immersed 60 to 70% HNO_3 and soaked for several days, followed by sulfuric–chromic acid cleaning solution, aqua regia, 70% HNO_3, with 1% HF

Material	Treatment
Plastics (lucite, tygon, neoprene, and rubber)	Some of the equipment is discarded, smooth surfaces wiped with acetone and emery cloth
Concrete	Stiff brush using Bon-Ami paste, soap and water Na_2CO_3, highly contaminated areas have to be chipped and removed, citric acid, trisodium phosphate
Wood	Sanding or planing, discarding wood materials
Tools and machine parts > 50 mr/hr	Boiled in 10% citric acid and 5% Zepco powder, scrubbed and rinsed to prevent corrosion and remove citrate film
Tools and machine parts < 50 mr/hr	Soaked in 10% citric acid and 5% Zepco, kept below the boiling point, scrubbed, and rinsed
Large machines and vehicles > 50 mr/hr	Smeared with a paste of 1 gal flake NaOH, 1 qt starch, 1 pt Zepco soap, allowed to soak, and removed with hot water under pressure
Large machines and vehicles < 50 mr/hr	Scrubbed with 2 parts citric acid, 1 part Zepco, made into a lather

Coated surfaces	Treatment
Ucilon paint	Decontaminated with water, trisodium phosphate, 5% ammonium citrate, 2% ammonium bifluoride, or remove Ucilon paint with organic solvent
Liquid envelope	Stripable coating sprayed on Ucilon paint; 10% nitric, 5% ammonium citrate. If necessary, the coating can be stripped from material
Prufcoat paint	2% ammonium citrate
Waxed linoleum	Apply organic solvent gently with cloth, avoiding the spread of contamination
Asphalt tile	Apply organic solvent; replace highly contaminated section of the tile

Appendix B: Hand Decontamination Procedure For Health Physics Surveyors

Procedure.

1. Obtain hand count on Hand and Foot Counter. If possible, locate specific hot spots with probe.
2. Wash thoroughly with soap and water, using a surgical scrub. Repeat Step 1. If tolerance is exceeded, proceed with Step 3.
3. Pour a liberal quantity of cleanser on the hands, brush, and follow with soap and water rinse. Repeat Step 1. Proceed with Step 4 if results remain unchanged.
4. Moisten hands with water, apply citric acid crystals and rub for several minutes, then rinse and follow with a soap and water wash. Check results, then proceed with Step 5, if necessary.
5. Apply $KMnO_4$ crystals on moistened hands. Rub into skin thoroughly, Rinse the hands. Shake citric acid crystals into the palm and mix well with the $KMnO_4$. If hands become warm, immediately rinse in tepid water followed by a soap scrub and water rinse. The $KMnO_4$ stain can be removed by $NaHSO_3$.
6. If all the above methods fail after a fair trial of each, send the employee to the Health Division.
7. Lanolin is to be spread upon the hands after decontamination to soften the hands and prevent chapping.

Suggestions.

1. Avoid prolonged use of any one method. Irritation of the skin might impede the success of more suitable procedures. No one chemical treatment is known to be specific for all the elements with which one may become contaminated.
2. Do not use organic solvents. These may make the skin more permeable to contamination.
3. Oxalic acid is poisonous. It is not to be used under any circumstances.

4. For wounds: rinse quickly under running water followed by a soap and brush wash if possible. Accompany employee to Medical where further hand decontamination will be attempted only under their advice and supervision.

<u>Decontamination Kits Contain.</u>

1. Cleanser
    a. Zephonite (powdered soap)
    b. Lanokleen (powdered soap)
    c. Liquid soap
    d. Old Dutch Cleanser
2. Four scrubbing brushes
3. Two 1-lb citric acid jars
4. Two 1-lb $KMnO_4$ jars
5. Two bars of soap
6. One bottle of $NaHSO_3$
7. Lanolin
8. One pair of surgical gloves

Keep kit up to full strength by replacing used articles from stores as soon as they are depleted.

8.7 THE DESIGN AND OPERATION OF RADIOCHEMICAL LABORATORIES*

By W. H. Ray

Responsible Operation Essential

In contrast to a chemical works, a laboratory cannot be made foolproof by its design or equipment. In addition to facilitating design and beneficent equipment, safety in a laboratory requires the elimination of fools and foolish operation. The prerequisite is education, but some persons must expect to be excluded from laboratory work if they exhibit inaptitude at protecting lives. Perhaps licensing may evolve, if legal activity is incited by misfortunes in laboratories dealing with especially hazardous materials and processes. This can be forestalled if appropriate selection and restriction is exercised by managements responsible for laboratory operation.

Safety in a radiochemical laboratory requires that each individual permitted to control any radioactive material shall be competent and responsible to prevent exposure to any person of radiation dosage in excess of the maximum permissible rate (0.1 rem/day). This responsibility extends beyond the radiation produced by sources assigned to the responsible person, and in some cases may mean that no additional exposure to radiation from these sources can be permitted. Control over all radioactive material is to be maintained until it is transferred to other competent and responsible individuals or until an approved preparation for disposal has been compounded.

Numerous rules and devices for assistance in keeping exposures to radiation less than the maximum permissible may be proferred, but they cannot make a laboratory safe if those entrusted with radioactive materials do not evidence responsibility for the associated hazards. The paramount rule is to prevent the exposure of any part of the body to irradiation in excess of 0.1 rem/24 hr. Intelligence exercised by the persons responsible will find many ways to achieve this, whether the hazard is exposure to penetrating radiations or the intake of radioactive contaminants that may be retained in the anatomy.

The following examples indicate some of the malpractices in a radiochemical laboratory which can be guarded against without special knowledge.

1. Use of food product containers to hold active materials
2. Use of refrigerators for foods and radioactive samples jointly
3. Eating in radiochamical laboratories
4. Smoking while handling radioactive materials, especially while wearing protective gloves

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*The following material, previously written by Mr. Ray for another purpose, was issued in lieu of lecture notes.

5. Handling telephones, reports, etc., while wearing protective gloves
6. Building of inadequate shields, i.e., a lead shield with one or more sides missing
7. Failure to wear coveralls, gloves, rubbers, respirators, and such protective chothing when job conditions warrant contamination protection
8. Continued wearing of coveralls, etc., that are highly contaminated
9. Failure to wear personnel-monitoring meters, especially for "hot" operations
10. Tampering with personnel monitoring meters to confuse the health physics records
11. Pipeting by mouth of radioactive solutions
12. Handling radioactive sources without proper protection by distance or shielding
13. Storage of radioactive material without use of proper danger signs
14. Leaving radioactive samples on benches when not in use
15. Storage of radioactive material in rooms assigned to other men without proper notification of person responsible for that room
16. Improper disposal of contaminated apparatus, e.g., "hot" material in an ordinary trash basket, or mixing of contaminated tools or equipment with clean items
17. Discharge of "hot" solutions via a "cold" sink
18. Failure to check for hand contamination on a routine basis
19. Assuming because an operation has been done before without observed ill effects that it is therefore safe
20. Continued acceptance of a region approved by a survey as safe without attention to changes caused by activity growth or transfer of radioactive materials from safe enclosures
21. Depending upon a statement of safety regarding a particular material without concern for the interpretation of safety relative to the working conditions

Contamination Must be Controlled

Restricting radioactivity to as small a zone as possible makes it easier to protect people from health hazards and simplifies material accounting. Immaculate housekeeping is almost synonymous with close control of contamination.

An ideal is approached when all radioactive materials are kept in a space sealed away from personnel. Hoods with an adequate air flow (in excess of 100 ft/min) cannot be expected to prevent tactile transfer of contamination. A cell to provide a contamination seal may take the form of a box equipped with windows in which the work with radioactive materials is performed manually through rubber gauntlets built into the walls of the enclosure. The colloquial term for this device, which permits work under a controlled atmosphere, is "dry box."

A more elegant dry box may be the size of a hood and be provided with built-in utilities. A service corridor should provide access to the rear, where removable panels permit changes of equipment without spreading contamination to the operating room. The service corridor can be arranged to accommodate any number of units. It terminates in a decontamination room and a decontamination change room. No occupancy in the service corridor is permitted without complete protective clothing and air-supplied hood. All items removed must pass through the decontamination room and be effectively cleaned or acceptably packaged.

The infiltration of contamination into walls and furred spaces must be controlled to minimize hazards to fire fighters and to facilitate razing and remodeling when they occur. Air-borne contamination has been known to travel through wall spaces along plumbing and to emerge in zones where no provision for contamination had been prepared. This phenomenon can be overcome by maintaining the atmosphere in spaces occupied by radioactive materials at pressures below ambient.

Smooth impervious simple surfaces facilitate the recovery of contamination that may escape experimental apparatus by accident. Stainless steel, glass, and lead can be subject to cleaning with strong acids. If linoleum surfaces fail to respond to decontamination, they can be economically replaced. Plastered surfaces should be sealed with a glossy paint. Seriously contaminated concrete requires physical removal. Obscuring of contamination, such as the covering of alpha emitters with paint, is unacceptable.

The wearing of protective clothing will not control the spread of contamination. However, the removal of protective clothing is an appropriate method for personnel to decontaminate where contamination control is to be effected.

Contamination zones of three types may be established to control the wearing of discardable clothing (protective clothing).

1. "Cold" zone in which contamination above background (natural) is impossible.
2. "Low-risk" zone in which immaculate conditions will normally be attained, but within which exists a possibility of accidental contamination. Such contamination is to be cleaned up immediately upon its occurrence.
3. "High-risk" zone into which entry without acquiring contamination is unlikely.

High-risk zones are exclusion areas to which casual access of personnel is prohibited. Protective covering shall be worn in high-risk zones which will prevent contact of the body with the contaminating materials to be encountered. All clothing worn in a high-risk zone shall be removed or passed by appropriate frisking with radiation-detection instruments before the wearer may leave the boundary of that zone.

The wearing of protective clothing in a low-risk zone is recommended. Contaminated garments and other contaminated items are subject to confiscation. Supervision is to maintain contamination levels below that which could be a health hazard for a continuous exposure of 8 hr per day throughout all low-risk zones.

Protective clothing worn in either high- or low-risk zones shall be excluded from any cold zones. Measures should be instituted to ensure that no contamination which might become a legal liability will be permitted to pass into a cold zone.

Cold zones may include counting rooms, instrument shops, urinalysis laboratories, off-plant areas, cafeteria, etc.

Details of zone boundaries are to be established for local areas by the supervision responsible for directing work in these areas. This supervision will also be responsible for adequately labeling the boundaries. A high-risk zone may suddenly develop within a low-risk area. A small high-risk zone may be established about the immediate vicinity of certain operations. Although too small to admit a body, any portion of a person entering a high-risk zone shall be governed by high-risk zone procedures.

Supervision of areas in which high-risk zones occur should provide protective clothing, change room, and contamination checking facilities to accommodate all workers on jobs for which they are responsible.

It is evident that a decontamination-type change room should be located at a zone boundary. Whether compulsory showers or merely slip-over garment procedures are required depends upon the nature of work to which personnel will be assigned. Arrangement should be such that correct procedure in the change facilities is easy to follow and difficult to avoid Crosstracking over areas where work shoes are worn by bare feet and shoes for the clean side should be avoided. If low-risk clothing will be worn more than once, lockers separate from those for street clothing should be provided for each person.

Contamination control is needed to preclude general air-borne radioactive contamination, as well as to prevent the accumulation of hazardous sources of penetrating radiation. The limitation based upon possible intake of radioactive material into the body is not always the most stringent. The need to prevent nuisance contamination which may foul instruments and thwart experiments provides an additional motive that favors the radiation safety program.

Devise Protection from Penetrating Radiation

When radioactive materials other than alpha emitters are to be cared for, protection from the radiation must be arranged in addition to the control of contamination.

Alpha radiation is excepted because its low penetrability makes any practical contamination-control barrier enough to shield live tissues completely. Energetic alpha rays will only penetrate a few centimeters of air and are likewise stopped completely by one or two sheets of bond paper.

Therefore rubber gloves are adequate protection. Even the dead outer layers of skin on the hands afford protection from external sources of alpha radiation.

Beta radiation may penetrate several meters of air, and for high energies 1 in. of wood may not be sufficient to stop it Beta radiation is attenuated by shielding, but since it takes but little to stop it completely, practice is to use enough for complete stoppage and thus eliminate concern for quantity measurements. The liquid in which a beta source is in solution provides an appreciable degree of self-absorption. If the source liquid is not exposed to the operator (mist, films, and drops included), the walls of the containing vessels may be made thick enough to provide adequate shielding without difficulty. For low-energy beta emitters, ordinary glassware may sometimes suffice. The thickness to be used is determined by the maximum energy of the radioactive beta emitter to be dealt with and not the quantity.

An exception may be considered when great intensities of beta radiation are involved. Beta radiation is composed of high-velocity electrons. When these strike an absorber, some deceleration is experienced, something like that in an X-ray-tube target. The resulting electromagnetic radiation, called "bremsstrahlung," is more penetrating than the beta radiation that produced it, and it can be observed emerging from the shielding where it is generated. Under the most favorable circumstances this can be only a few per cent of the incident beta, but in these days of high orders of radioactivity, bremsstrahlung can by itself be a serious health hazard.

This emphasizes the desirability of checking the effectiveness of shielding with radiation instruments for other reasons than the detection of faults.

Gamma rays, unlike beta and alpha, have no maximum range in an absorbing medium. They are attenuated exponentially. That means that no matter how thick a shield may be, some fraction of the penetrating electromagnetic radiation will always get through. Therefore the thickness of shielding depends not only on the energy of the emitted radiation, but also upon its incident intensity and the intensity that may be permitted to emerge.

The penetrating ability of gamma rays increases with their energy up to about 2 Mev. Above this energy per photon absorbers become more effective again, particularly those of high atomic number. For low-energy gamma radiation, absorbers are more effective as the atomic number increases. This is in addition to crowding atoms together to make a dense absorber. Lead is often used for economic and spatial reasons. Steel is structurally superior. Concrete also has structural and economic advantages, but it is not so conservative of space. To get the most absorption in a minimum thickness, uranium is suggested. Gold, too, is better than lead in this regard.

A convenient "rule of thumb" to remember is that for the penetrating radiation from radium a half-absorption thickness of lead is approximately 1/2 in. For example a radium source is stored in a lead pot 4 in. thick. The radiation coming through the wall of the office adjacent the store room is twice the desired magnitude. An additional 1/2 in. will produce the desired attenuation.

Tables are available to permit a computation of the shielding thickness required if it is known what the shielding material and density are to be, the intensity and energy spectrum of the incident radiation, and the intensity desired for the emergent radiation. The first two factors are determined by the materials to be used. The last factor is based on the protection of persons or the shielding of radiation-sensitive equipment.

To protect personnel we are guided by the requirement that exposure of any part of the body to beta or gamma irradiation must be limited to 1/10 roentgen equivalent physical (rep) per 24 hour day. This may be achieved with the assistance of shielding, but limiting time of exposure and controlling the distance of approach to a source are equally effective means of preventing overexposure. It is customary to measure radiation in dosage rate. Milliroentgens per hour has been used almost exclusively on the Plutonium Projects. If exposure is to a radiation dosage rate of 100 mr/hr, limiting the exposure time to 1 hr per 24-hr day will keep one under the maximum permissible exposure limit.

If 1 hr/day is not enough working time, doubling the distance from the source in the above example (providing it is essentially a point source) will increase the permissible working time to 4 hr/day. One must guard against applying the inverse square law to radiation close to extended sources. As a linear source, such as a pipe, is approached within a distance equivalent to its length, departures from the inverse square law merge into an inverse first-power relationship. If the source is an area that can be approximated by a circle, the intensity of radiation at a point along a normal to its center is the product of the intensity at the surface times ($1 - \cos^2 \theta$), where θ is the angle between the normal to the center of the circle and a line through the point to the circumference.

Combining these factors to attain protection from penetrating radiation involves integration of laboratory equipment and the manner in which the work to be done is performed.

If, at a point in an infinite volume of the most radioactive solution to be handled, a radiation dosage rate of 50 mrep/hr cannot be exceeded, solutions of this concentration may be manipulated up to 2 hr/day with only contamination control. This concentration would be 20 μc/liter if the average energy per disintegration were 1 Mev. (The concentration to yield the same irradiation is inversely proportional to the average energy.) For beta emitters the average energy can often be considered to be one-third of the maximum.

Sources that will not produce more than 10 mr/hr at 4 in. can be manipulated practically with tongs and forceps. Contact with radioactive materials should generally be avoided. The inverse square law intensity increases with amazing rapidity as distance approaches zero. In general, some form of remote control is necessary for manipulating emitters of penetrating radiation regardless of the extent of shielding. Once this has been mastered, there is little excuse to use shielding that is too thin.

Nelson Garden of Berkeley has discovered that three-fourths of an experimenter's time (in radiochemistry) is consumed in setting up apparatus and executing a cold-run dress rehearsal. He has devised standard jigs to fit in a shield or "cave" well equipped for remote-control manipulations. These jigs are equipped and tested in an ordinary chemical laboratory without radioactivity before being introduced into the cave for the actual experiment. At the conclusion of an experiment the jig emerges from the other end of the cave into a decontamination room for salvage or disposal.

Laboratories should be large enough to permit good housekeeping and make manipulations easy. Rooms should not accommodate more than one experiment or operation at a time. Only men working on the job should have access to each laboratory cubicle, and adequate shielding should be incorporated in walls and entry labyrinth to protect all neighbors. Multistory buildings are a disadvantage because of the need for additional shielding above and below. Shielding waste pipes and the need for elevators to handle shielded containers also deserve consideration if a multistory building is to be used for radioactive work.

Where it is necessary to wear gloves for an operation, it has been found helpful to have an assistant to do the bare-handed work in order to minimize glove removal and reduce the temptation to touch clean things with contaminated gloves. Not more than two men in a laboratory is advocated to eliminate unnecessary gratuitous irradiation.

Laboratory Design Tailored to Needs

In addition to applying the factors we have discussed to a specific situation, there are other considerations that should go into outfitting and operating a radiochemical laboratory. Packing units and formal rules can be correctly applied to specific installations only.

From the above, the desirability of radiation instrumentation in the laboratory is obvious. Radiation instruments are to be described in the next discussion.

The care required for solid, liquid, and gaseous-borne radioactive wastes is also to be discussed subsequently. Please bear in mind that this is also a part of radiochemistry laboratory planning.

8.8 SURVEY TECHNIQUES AND REFINEMENTS

By T. H. J. Burnett and Associates

The purpose of this discussion is the furtherance of a high professional standing, especially in the opinions of the groups associated with us at this laboratory.

It is valuable and desirable to exercise whatever reasonable additional care may be required so that those cases in which an apparent disagreement or contradiction is thought to be seen between the readings and/or recommendations of different individual health physicists will be reduced to a minimum.

The following six points are those frequently involved in such situations of potential criticism:

1. Continuity of effort involves situations from which might arise apparent conflicts of opinion on working times recommended for permissible exposure. On the same job, when it is continuing over several work periods, different health physics representatives may give different working times, and this is often noticed and wondered about, particularly by workmen with more meager scientific backgrounds.

This situation is best avoided by checking the log as soon as you come on duty to see the range of readings on such continuing jobs. Second, the prior safety work permits issued for specific jobs should always be checked. Thus, when a different working time seems more proper, it can be given with a casual comment on the probable reasons for its difference.

It is not hereby suggested that efforts should be strained to get the precise readings of your predecessor. However, knowing their values gives a basis for inquiry into the reasons, peculiar to each specific job, for different radiation levels and may indicate the need of more frequent survey coverage than had been anticipated.

A brief discussion of the features of the Safety Work Permit was given by A. D. Warden, followed by an outline of a proposed new procedure for individual exposure records for "hot" jobs by J. C. Hart.

Before resuming, it was stressed that the responsibility lies strictly with the supervisor to stipulate whether or not a radiation hazard should be anticipated. Health physics will exercise due professional care and concern lest any radiation hazards be overlooked.

2. Nonuniformity of permissible exposure is to be expected under certain circumstances. The maximum permissible daily exposure to an employee working routinely in a laboratory with a relatively high background may well be set lower than that of a maintenance employee called in for an occasional repair job. Similar discrepancies arise when part of the work crew has previously worked on a "hot" job, getting some fraction of their maximum permissible exposure for the day. These individuals, then, would have working times proportionately shortened in comparison to those of their fellows.

In such circumstances it is well for the health physics representative to consider the advisability of a casual word of simple explanation to the foreman in charge of the maintenance crew to allay any untoward misgivings.

3. Variation between instruments is difficult to forestall, since if one survey is made with an instrument having the allowable variation of approximately 10 per cent "high" and a survey is made of the identical conditions with another instrument having a variation of 10 per cent "low," a net difference in these two surveys of 22 per cent will result. In every case then, it is easily seen that the surveys made should be carried out with the maximum of care so as to utilize the inherent accuracy of these instruments to the full. A minor variation of 10 per cent can be casually cited as due to unavoidable instrument variation. When possible, it is desirable to designate on the safety work permit the number of the instrument used so a recheck with the same instrument can be made if desirable and also as an aid to discovering instrument malfunctioning.

4. Background change is the most frequent, important, and significant cause for variation in recommended working times. This may take the form of an increase in working time, such as would accompany successful progressive decontamination efforts, or the increase of shielding (in the case of covering a hot pipe line with dirt) due to the furtherance of the work or the use of temporary lead shielding.

Otherwise the progress of the job may result in lessening the working time, such as in the case of an excavation where shielding dirt removal from hot pipe lines produces an increasing radiation field (as well as introducing a serious contamination problem when such pipe lines have leaked) or in the case of equipment disassembly where interior parts exposed may be much higher in activity.

Under these conditions it is readily apparent that new working times proper to the stage of work are needed as the job progresses. In such types of jobs, the health physics representative should clearly establish the importance of his being currently informed of significant changes produced by the progress of the work so the job can be resurveyed. All work where such changes can be foreseen as possibilities should be watched with extra care by the Health Physics Division.

5. Failure to work within designated boundary lines is occasionally encountered, and obviously if one surveyor surveys location A and the men perform their work at location B or at A and B, then a survey of location B might very properly be indicative of quite different working times. This situation results most usually from an imperfect description of the area of action being furnished to the Health Physics Division.

If the health physics representative observes that a maintenance job is being carried out in an area or location different from that initially surveyed, he should certainly point out the need for a further check to the foreman of the job. It would not be improper under such circumstances to request that further work be halted, if possible, until the proper area affected can be surveyed to ascertain the correct working time.

6. Failure to follow recommendations made by the health physics representative is potentially serious in every case, and, if this is due to deliberate disregard of necessary precautions, it should be at once reported to appropriate supervision for whatever corrective or disciplinary action necessary.

Fortunately such disregard is most often the result of carelessness, forgetfulness, or thoughtlessness. This may show up in numerous ways, from the failure to wear proper protective clothing or observe given working-time limits to neglect of interposing necessary auxiliary shielding, or even complete failure on the part of operating supervision to notify maintenance crews of the provisions of the safety work permit.

Cognizance of these six points and the exercise of careful judgment, coupled with maximum reasonable accuracy in the use of the most appropriate instruments, will assist in the achievement of a professional standing and record of unexcelled caliber.

UNIT 9. QUALIFYING AS A HEALTH PHYSICIST (EVALUATION)

9.1 DISCUSSION OF ACHIEVEMENTS

By T. H. J. Burnett

Here, at the conclusion of this series of meetings, it should be pointed out that the work presented so far comprises but a survey or introduction to your field of health physics. Much yet remains to be learned by you, and only with further study and application can that necessary mastery be achieved of those more advanced topics requisite to your full stature as a health physicist.

You, as developing scientists, should have a fertile interest in all new progress in science as a whole, reading currently both in the science sections of the lay press and in the available scientific journals. The fertility of your interest will evidence itself by the suggested applications of what you read to our current endeavors. Typical of this might be new instrument developments, new medical discoveries such as the application of radioactive materials to treatment and/or research, new atomic-research centers, new methods of producing radioisotopes, catastrophe planning for atomic-weaponed attack, etc. For much of your further study, many of you will profit by home study, review, or night-class attendance in such subjects as mathematics, physics, biology, and chemistry. The latter subjects are locally available in evening courses most of the year.

Review and discussion by groups of your own of the material just covered in this course and outlined in these notes should be quite valuable to sustain your recently developed skills at their maximum usefulness.

Being alert to new problems that arise in your respective responsibility areas will enable you not only to be of greater service value to your associates, but will give you more interest in your work. Much valuable applied research to field problems can be done with the facilities at your disposal. Of these, to cite only one example, would be differentially shielded film studies of the exposure-energy distributions in typical work areas and operations, permitting better correlation of exposure monitoring results with prior instrument surveys.

As to the direction your further studies in health physics should take and what subjects ought to be included, no attempt is herewith made to present any thoroughly weighed program for an advanced course. However, it may be readily seen with even a little reflection that the following subjects might very appropriately be treated:

1. Additional work on radioactive decay and the growth of radioactive daughter products. This would include further study of the exponential equations involved, preferably without the handicap of avoiding calculus.
2. The above will integrate well with a more thorough and basic study of tolerances and tolerance concentrations, which involve consideration of both biological elimination and radioactive decay, considered as providing a combined exponential relationship of time and dosage.
3. More consideration given to instruments and instrumentation will be profitable since it will develop a further degree of skill in the choice and proper usage of the appropriate instruments, as well as the improvement of interpretation.
4. The above will profitably be associated with further and less naive study of ionization mechanism, nuclear structure and reactions, beta decay, and kindred topics. Knowledge of the interaction of radiation with matter, especially tissue, can here be given a better basis.
5. Emphasis can deservedly be given here to more detailed study of the biological aspects of radiation injury for the different types of radiation, as well as to a survey of the results to date of

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experimental research on radiation damage to various tissues, organs, and functions of the body as observed with laboratory animals, etc.

6. Correlation with industrial health and hygiene information, sanitary engineering as it ties in with radioactive waste disposal, more sophisticated and detailed shielding studies and planning, and many others are further topics for advanced study in the development of adequate qualifications as a health physicist.

While much more might have been included even in the scope of this introductory course, such as details of the characteristics of various specific isotopes, etc., it was felt that the program, as it has been given, represented the optimum coverage for the time available.

The full measure of your achievements in this course can only be gauged in the progress you make in the months to come and in the improvements you show as you fulfill our imposed obligation to reduce radiation-caused injury to the individual and to the race to a mimimum.

END OF DOCUMENT

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EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
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1.0 PURPOSE

Technical Information Bulletins (TIBs) are general working documents that provide guidance concerning the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained. TIBs may be used to assist the National Institute for Occupational Safety and Health in the completion of individual dose reconstructions.

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

There are instances of energy employees who, for a variety of reasons, were not monitored for internal exposure during the course of their employment at a U.S. Department of Energy (DOE) facility, or whose records of such monitoring are incomplete or unavailable. In such cases, data from coworkers may be used to approximate an individual's possible exposure. The purpose of this document is to provide the details of the calculation and assignment of intakes based on coworker data from Oak Ridge National Laboratory (ORNL) for the purpose of estimating unmonitored exposures or where records of monitoring are incomplete or unavailable, whether for discrete periods or for the entire period of employment.

2.0 OVERVIEW

Analysis of Coworker Bioassay Data for Internal Dose Assignment (ORAU 2004a) describes the general process used for analyzing bioassay data for assigning doses to individuals based on coworker results.

Bioassay results for ORNL were obtained from the Oak Ridge Institute for Science and Education (ORISE) Center for Epidemiologic Research (CER) Dosimetry Database, which contains urinalysis records from the ORNL site for the period 1951 to 1988. ORISE obtained this database from ORNL for the purpose of conducting an epidemiology study of site workers. The database results are in units of disintegrations per minute (dpm)/24 hours. Because of the varied operations at the different ORNL facilities over time with the potential for exposure from numerous different radionuclides, the database contains urinalysis data for numerous radionuclides. These data are stored using Electronic Data Processing (EDP) codes, as documented in Section 5.1.5.2 of the X-10 Technical Basis Document (TBD) (ORAU 2004b). In summary, data were stored under 64 different EDP codes and included measurements for five radioisotopes of uranium, four radioisotopes of plutonium, seven other transuranic radionuclides, numerous fission and activation products, and gross alpha and gross beta measurements. The majority of the EDP codes contained fewer than 100 data entries. In reviewing these data, it was determined that only a few of the EDP codes contained enough entries to allow statistical evaluation for purposes of dose reconstruction. These EDP codes were: SR and SR0 for ^{90}Sr (12,893 entries from 1951 to 1988), UR0 and UR for total uranium (11,434 entries from 1951 to 1988), PU0, PU9, PU, GA0, and GU0 for plutonium alpha-emitters (15,476 entries from 1951 to 1988), and TP0 and TP for transplutonium radionuclides, primarily ^{241}Am (5,670 entries from 1968 to 1987). Further review of these data indicated that, except for follow-up for accidental exposures, they were collected on an annual sampling basis; therefore, the analysis that follows considers chronic exposures to estimate annual intakes. A statistical analysis of these data was performed in accordance with ORAU (2004a). The resultant values were input to the Integrated Modules for Bioassay Analysis (IMBA) Expert OCAS-Edition computer program, and a fit to the data for each of the four radionuclides was performed to obtain intake rates for assigning dose distributions.

3.0 DATA

3.1 Selected Bioassay Data

Data for each of the EDP codes considered were extracted from a series of Microsoft® Access files that contain a version of the ORISE/CER Dosimetry Database. These files were titled "tblORNL_Urinalysis_rawData," for the periods 1951-1978, 1979-1985, and 1986-1988.

3.2 Analysis

A lognormal distribution for the annual data for each of the four radionuclides was assumed, and the 50th and 84th percentiles were calculated, using the method described in ORAU (2004a). Tables A-1 through A-4 in Attachment A show the statistical analysis results for strontium, uranium, plutonium, and transplutonium.

4.0 INTAKE MODELING

4.1 Assumptions

All results were assumed to be representative of a full day (24 hr) of urinary excretion. Each result used in the intake calculation was assumed to be normally distributed, and a uniform absolute error of 1 was applied to all results, thus weighting all results equally. A chronic exposure pattern was assumed; while this is unlikely for workers at ORNL, it will approximate a series of acute intakes with unknown intake dates. Intakes were assumed to be via inhalation using a default breathing rate of 1.2 m^3/hr and a 5-μm activity median aerodynamic diameter (AMAD) particle size distribution.

4.2 Bioassay Fitting

The IMBA Expert OCAS-Edition computer program was used to fit the bioassay results to a series of inhalation intakes. Data for each radionuclide were fit as a series of chronic intakes.

Because the Type S strontium, uranium, and plutonium isotopes and Type M plutonium and americium isotopes present at ORNL have very long half-lives and because the material is retained in the body for long periods, excretion results are not independent. For example, an intake in the early 1950s could contribute to urinary excretion in the 1980s and later. To avoid potential underestimation of intakes for people who worked at ORNL for relatively short periods, each intake was fit independently, using only the bioassay results from the single intake period. This will result in an overestimate of intakes, particularly for assumed Type S exposures extending through multiple assumed intake periods.

4.3 Radionuclides And Material Types

For each radionuclide considered, the bioassay results were entered into IMBA with assumed material types as discussed in more detail in the following sections. The assumed 50th-percentile intakes that result are displayed in the figures in Attachment A. In these figures, annual bioassay data used in the fits are shown as the dark dots (•), and data that are not used in the fits are shown as lighter dots (•). For certain radionuclides and material types, it is necessary to show figures for both the composite intake over the entire data period and the individual plots of subsets of selected groups of years during the data period that were used to develop the composite figure. The composite plot shows the merged fit for all of the intake periods to the bioassay data.

The IMBA output figures in Attachment A show the fit to the bioassay data for each annual intake period in terms of days after the date of initial data reporting. For example, for ^{90}Sr, day 1 equals June 1, 1951, and day 13,515 equals June 1, 1988.

4.3.1 Strontium-90

Strontium results were assumed to be Sr-90. Because of the presence of ^{90}Sr both as an effluent from reactor operations and in a highly insoluble titanite form, bioassay results for ^{90}Sr were fit using Type F and S material. Figure A-1 in Attachment A shows the individual fits to the 50th-percentile intake values for Type F ^{90}Sr. Table A-5 summarizes the intake periods and corresponding intake rates for the 50th- and 84th-percentile values, and the geometric standard deviations (GSDs) for Type F ^{90}Sr . The GSDs were determined by dividing the 84th-percentile intake rates by the 50th-percentile intake rates. Figures A-2 through A-6 show the fits corresponding to each of the subsets of years across the intake period used to develop the composite figure for Type S ^{90}Sr. Figure A-7 shows the composite of the subsets of selected groups of years used to fit the 50th-percentile intake values for Type S ^{90}Sr. Table A-6 summarizes the intake periods and corresponding intake rates for the 50th- and 84th-percentile values, and the calculated GSDs for Type S ^{90}Sr. The same intake periods were applied to the 84th-percentile values for both ^{90}Sr Type F and S material because the values followed a similar pattern; results of the individual fits are not shown here since they were largely in agreement. It is noted that the urinalysis data for 1964 were insufficient to support a complete analysis and was therefore was omitted from the sample set.

4.3.2 Uranium-234

Because a variety of uranium enrichments and exposure conditions are possible at the ORNL site, ^{234}U was assumed for the IMBA intake modeling even though the source data are for gross uranium in varying amounts, depending on the enrichment. This does not affect the fitting of the data for intake determination (i.e., the same total intakes would be obtained for any enrichment that was assumed) because all uranium isotopes behave the same biokinetically and the isotopes considered in this analysis have long half-lives relative to the assumed intake period. The ICRP 68 (1995) dose coefficients (also referred to as dose conversion factors) for ^{234}U are 7% to 31% larger than those for ^{235}U, ^{236}U, and ^{238}U. Because of the isotopic compositions of the source terms, the ^{234}U dose conversion factor will overestimate doses for any combination of the uranium radioisotopes, but the assumption of intake of 100% ^{234}U is made to ensure claimant-favorability of the doses.

Because uranium is found in many forms at ORNL, the bioassay data were fit using Type F, M, and S material. Figures A-8 and A-9 show the individual fits for the 50th-percentile values for Type F and M ^{234}U, respectively. Table A-7 summarizes the intake periods for the 50th- and 84th-percentile values, and the calculated GSDs for Type F ^{234}U, while Table A-8 shows similar information for Type M ^{234}U. Again, the GSDs were determined by dividing the 84th-percentile intake rates by the 50th-percentile intake rates. For Type S ^{234}U, Figures A-10 through A-16 show the fits corresponding to each of the subsets of years across the intake period used to develop the composite figure. Figure A-17 provides the composite of the subsets of selected groups of years used to fit the 50th-percentile intake values. Table A-9 summarizes the intake periods and corresponding intake rates for the 50th- and 84th-percentile values, and the calculated GSDs for Type S ^{234}U. Plots of the results of the individual fits to the 84th-percentile values for all three solubility types are not shown here since they were largely in agreement.

4.3.3 Plutonium-239

For this analysis, the analyzed material types for ^{239}Pu are assumed to be types M and S. Although the bioassay results are for all alpha-emitting isotopes of plutonium, the results were assumed to represent the concentration of ^{239}Pu alone,. Figures A-18 through A-21 show the fits for Type M material corresponding to each of the subsets of years across the intake period used to develop the composite figure. Figure A-22 shows the composite of the subsets of selected years used to fit the 50th-percentile intake values for type M ^{239}Pu. Table A-10 summarizes the intake periods and corresponding intake rates for the 50th- and 84th-percentile values, and the GSDs for type M ^{239}Pu. The GSDs were determined by dividing the 84th-percentile intake rates by the 50th-percentile intake rates. Again, the same intake periods were applied to the 84th-percentile values for Type M ^{239}Pu because the values followed a similar pattern; results of the individual fits are not shown here since they were largely in agreement.

Figure A-23 shows the fit for Type S material based on a chronic intake for the entire set of years fitted to the bioassay data for the last three years (1986 through 1988). Table A-11 summarizes the intake period and corresponding intake rate for the 50th- and 84th-percentile values, and the GSDs for type S ^{239}Pu. Again, the GSDs were determined by dividing the 84th-percentile intake rates by the 50th-percentile intake rates. The same intake periods were applied to the 84th-percentile values for Type S ^{239}Pu because the values followed a similar pattern; results of the fit are not shown here since they were largely in agreement.

4.3.4 Americium-241

ICRP 68 (1995) assigns all forms of americium to Type M. Figures A-24 and A-25 show the fits corresponding to each of the subsets of years across the intake period used to develop the composite figure. Figure A-26 shows the composite of the subsets of selected years used to fit the 50th-percentile intake values for Type M ^{241}Am. Table A-12 summarizes the intake periods and corresponding intake rates for the 50th- and 84th-percentile values, and the GSDs for Type M ^{241}Am. Again, the GSDs were determined by dividing the 84th-percentile intake rates by the 50th-percentile intake rates. The same intake periods were applied to the 84th-percentile values for Type M ^{241}Am because the values followed a similar pattern; results of the individual fits are not shown here since they were largely in agreement.

4.3.5 Additional Radionuclides

It is recognized that hundreds of different radionuclides were present at ORNL at some point during its operations. However, bioassay data for additional radionuclides beyond ^{90}Sr, ^{238}U, ^{239}Pu, and ^{241}Am were deemed to be of little use for coworker estimations, largely because there were too few measurements to be statistically reliable for intake estimation. For this analysis, three additional radionuclides, both important to internal dosimetry and present in the reported air monitoring data, were considered. These are ^{106}Ru, ^{137}Cs, and ^{144}Ce.

5.0 ASSIGNMENT OF INTAKES AND DOSES

The resulting calculated intake rate information, useful in dose reconstruction for the radionuclides identified for ORNL, is discussed in this section. For each radionuclide, the 50th- and 84th-percentile intake rates, and the GSDs, are provided in specific tables. The GSD values have been adjusted from the values given in the tables in Appendix A to allow for the addition of doses from different intake periods into a single input line for a given year in the IREP input file, and to ensure that none are less than 3, the value used when assigning intakes to individuals from person-specific bioassay results.

- Strontium-90. For ^{90}Sr, several intake periods were defined as shown in Table 5-1 for Type F material and Table 5-2 for Type S material. Note that there were five intake periods defined for ^{90}Sr Type F intakes and six intake periods defined for ^{90}Sr Type S intakes.

Table 5-1. Combined ^{90}Sr Type F intake periods and rates.

Date range		**^{90}Sr Type F intake rate, dpm/d**	
From	**To**	**50th percentile**	**GSD**
1/1/1951	12/31/1953	475.2	10.0
1/1/1954	12/31/1954	80.99	10.0
1/1/1955	12/31/1960	47.34	4.17
1/1/1961	12/31/1964	47.34	3.00
1/1/1965	12/31/1988	15.52	3.00

Table 5-2. Combined ^{90}Sr Type S intake periods and rates.

Date range		**^{90}Sr Type S intake rate, dpm/d**	
From	**To**	**50th percentile**	**GSD**
1/1/1951	12/31/1953	24646	9.53
1/1/1954	12/31/1954	7,232	9.53
1/1/1955	12/31/1960	2,379	5.73
1/1/1961	12/31/1964	2,379	3.00
1/1/1965	12/31/1983	795.0	3.00
1/1/1984	12/31/1988	425.5	4.51

- **Uranium-234.** Table 5-3 contains the calculated annual intake rates for the years 1951 through 1987, for Types F, M, and S material.

- Plutonium-239. Tables 5-4 and 5-5 contains the calculated annual intake rates for ^{239}Pu for Types M and S material respectively for the years 1951 through 1988.

- **Americium-241.** Table 5-6 contains the combined ^{241}Am Type M material intake periods and rates for 1968 through 1988. Note that there were two intake periods defined.

5.1 Contribution From Additional Radionuclides

To account for additional intakes, an evaluation of air monitoring data from the ORNL perimeter reporting stations for the period 1975 through 1984 was conducted. The approach was to develop the ratios of the isotopic concentration ratios of other radionuclides to the concentration ratio of ^{90}Sr reported in the air monitoring data. The results of this evaluation are summarized in Table 5-7, which shows the reported radionuclide concentrations, the ratio of the concentrations of the three selected radionuclides to ^{90}Sr, the peak concentration ratio, range of the concentration ratios, and the average concentration ratio.

5.2 Dose Assignment

In most cases, doses to be assigned to individuals potentially exposed on a routine basis are calculated from the 50th-percentile intake rates; the material type resulting in the largest probability of causation (which is determined by the Department of Labor) is selected. A comparison of the intake rates shows:

- For ^{90}Sr, the calculated intake rates for Type S material are one to two orders of magnitude higher than the intake rates of Type F material for all intake periods. However, because the Type S material remains in the lungs for an extended period while the Type F material is transferred to the systemic organs, it is necessary to compare the annual doses on a case-by-case basis to determine which will deliver the larger dose to the organ of interest.

Table 5-3. Annual ^{234}U Type F, M, and S intake periods and rates.

	^{234}U Type F intake rate, dpm/d		^{234}U Type M intake rate, dpm/d		^{234}U Type S intake rate, dpm/d	
Year	50th percentile	GSD	50th percentile	GSD	50th percentile	GSD
1951	11.19	3.08	47.03	3.33	850.7	3.0
1952	11.19	3.08	47.03	3.33	850.7	3.0
1953	11.19	3.08	47.03	3.33	850.7	3.00
1954	11.19	3.08	47.03	3.33	850.7	3.00
1955	11.19	3.08	47.03	3.33	850.7	3.00
1956	4.213	6.71	11.45	9.70	675.7	3.00
1957	1.942	9.25	7.49	9.70	247.7	7.74
1958	1.942	9.25	7.49	9.70	247.7	7.74
1959	5.171	3.48	21.81	3.33	509	4.17
1960	5.171	3.48	21.81	3.33	509	4.17
1961	5.171	3.48	21.81	3.33	509	4.17
1962	1.881	6.71	6.56	7.24	235.3	6.04
1963	1.881	6.71	6.56	7.24	235.3	6.04
1964	0.413	3.08	1.641	3.33	23.7	3.00
1965	0.413	3.08	1.641	3.33	23.7	3.00
1966	0.413	3.08	1.641	3.33	23.7	3.00
1967	0.413	3.08	1.641	3.33	23.7	3.00
1968	0.413	3.08	1.641	3.33	23.7	3.00
1969	0.413	3.08	1.641	3.33	23.7	3.00
1970	0.413	3.08	1.641	3.33	23.7	3.00
1971	0.413	3.08	1.641	3.33	23.7	3.00
1972	0.413	3.08	1.641	3.33	23.7	3.00
1973	0.413	3.08	1.641	3.33	23.7	3.00
1974	0.413	3.08	1.641	3.33	23.7	3.00
1975	0.413	3.08	1.641	3.33	23.7	3.00
1976	0.413	3.08	1.641	3.33	23.7	3.00
1977	0.413	3.08	1.641	3.33	23.7	3.00
1978	0.413	3.08	1.641	3.33	23.7	3.00
1979	0.413	3.08	1.641	3.33	23.7	3.00
1980	0.413	3.08	1.641	3.33	23.7	3.00
1981	0.413	3.08	1.641	3.33	23.7	3.00
1982	0.413	3.08	1.641	3.33	23.7	3.00
1983	0.413	3.08	1.641	3.33	15.25	4.67

1984	0.413	3.08	1.641	3.33	15.25	3.0
1985	0.0957	3.08	0.294	3.33	15.25	3.0
1986	0.0957	3.08	0.294	3.33	15.25	3.0
1987	0.0957	3.08	0.294	3.33	15.25	3.0

Table 5-4. Annual ^{239}Pu Type M intakes and rates.

Years	**^{239}Pu Type M intake rate, dpm/d**	
	50th percentile	**GSD**
1951-1952	40.75	3.0
1953-1959	10.98	7.9
1960-1968	10.98	3.0
1969-1984	7.35	3.0
1985-1988	1.614	4.2

Table 5-5. Annual ^{239}Pu Type S intakes and rates.

Year	**^{239}Pu Type S intake rate, dpm/d**	
	50th percentile	**GSD**
1951-1988	4.15	5.50

- For ^{234}U, the calculated intake rates for Type S material are one to two orders of magnitude higher than the intake rates of Type F and M material for all intake periods. Again, because Type S material remains in the lungs while Type F and M material is transferred to the systemic organs, it is necessary to compare the annual doses on a case-by-case basis to determine which will deliver the larger dose to the organ of interest.

- For ^{239}Pu, The calculated intake rates for type M material should be used for all systemic organs and intake periods. For non-systemic (respiratory and GI tracts) organs, the type S intake rate may only be used as an underestimate. If an overestimate or best estimate is needed for non-systemic organs, an individualized fit to the bioassay data, assuming type S material, for the specific work period of the Energy Employee being evaluated must be performed. Table A-3 provides the bioassay data to be used to perform the individualized fit. Both the 50th- and 84th-percentile values must be fit using the same intake dates and/or periods; the 50th-percentile intakes are used to assign the intake and the 84th-percentile is used to determine the GSD for each intake. The GSD for each intake will be the ratio of the 84th percentile intake rate to the 50th percentile intake rate. For input into IREP, the dose from each intake must be determined separately.

- For ^{241}Am, the calculated intake rates for Type M material should be used for all organs and intake periods as given in Table 5-6.

Table 5-6. Combined ^{241}Am Type M intake periods and rates.

Year	**^{241}Am Type M intake rate, dpm/d**		
	50th percentile	**84th percentile**	**GSD**
1968-1984	6.673	14.61	3.00
1985-1988	2.207	6.659	3.00

- When a ^{90}Sr intake is assigned, the worker should also be assigned intakes of ^{106}Ru, ^{137}Cs, and ^{144}Ce, consistent with the average isotopic ratios developed from the air monitoring data given in Table 5-7, using the ^{90}Sr intake as the basis. For example, if a worker were exposed in 1965 to Type F ^{90}Sr, the intake (from Table 5-1) would be 15.5 dpm/d. Intake of ^{106}Ru would also be assigned at a value of 103.8 dpm/d (i.e., 15.5 dpm/d of ^{90}Sr times 6.7, the average ratio of ^{106}Ru to ^{90}Sr from Table 5-7), along with a 34.1 dpm/d intake of ^{137}Cs and 170.5 dpm/d intake of ^{144}Ce. Although Table 5-7 is based on data from 1975 through 1984, the average ratios calculated should be used for all years.

  The lognormal distribution is selected in the Interactive RadioEpidemiological Program (IREP), with the calculated dose entered as Parameter 1 and the associated GSD as Parameter 2. The GSD is associated with the intake, so it is applied to all annual doses determined from the intake period. The GSD for ^{90}Sr should be assigned when intakes of ^{106}Ru, ^{137}Cs, and/or ^{144}Ce are included.

Table 5-7. Evaluation of the contribution of additional radionuclides.

Year	Reported ^{90}Sr concentration (10^{-15} µCi/mL)	Reported ^{106}Ru concentration (10^{-15} µCi/mL)	Ratio: ^{106}Ru to ^{90}Sr	Reported ^{137}Cs concentration (10^{-15} µCi/mL)	Ratio: ^{137}Cs to ^{90}Sr	Reported ^{144}Ce concentration (10^{-15} µCi/mL)	Ratio: ^{144}Ce to ^{90}Sr
1975[a]	0.78	6.7	8.6	1.4	1.8	11	14
1976[a]	0.77	0.88	1.1	0.55	0.7	2.4	3.1
1977[a]	1.30	11	8.5	1.60	1.2	21	16
1978[a]	0.81	9.7	12	2.14	2.6	16	19
1979[a]	0.15	1.56	10	0.67	4.5	1.8	12
1980[a]	0.08	0.49	6.1	0.37	4.6	0.98	12
1981[a]	0.46	4.40	9.6	1.2	2.6	11	24
1982[a]	0.15	0.40	2.7	0.24	1.6	0.03	0.2
1983[a]	0.08	0.12	1.5	0.10	1.2	N/A	N/A
1984[a]	0.07	N/A	N/A	0.07	1.0	N/A	N/A
Peak ratio	-	-	12	-	5	-	24
Ratio range	-	-	1.1 - 12	-	0.7 – 4.6	-	0.2 - 24
Average ratio	-	-	**6.7**	-	**2.2**	-	**11**

[a] References: 1975 – (Union Carbide 1975); 1976 – (Union Carbide 1976); 1977 – (Union Carbide 1977); 1978 – (Union Carbide 1978); 1979 – (Union Carbide 1979); 1980 – (Union Carbide 1980); 1981 – (Union Carbide 1981); 1982 – (Union Carbide 1982); 1983 – (Martin Marietta 1983); and 1984 – (Martin Marietta 1984).

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ATTACHMENT A

Table A-1. Statistical summary of annual ORNL strontium 24-hour urinary excretion rates, 1951-1988.

Effective bioassay date	GM (50th) dpm/day	GM*GSD (84th) dpm/day
7/1/1952	109.15	1,008.55
7/1/1954	24.68	241.98
7/1/1955	13.10	63.17
7/1/1956	13.54	82.88
7/1/1957	15.71	68.85
7/1/1958	17.01	75.45
7/1/1959	12.20	59.03
7/1/1960	15.67	65.07
7/1/1961	9.48	28.06
7/1/1962	12.04	29.49
7/1/1963	10.08	26.23
N/A*	N/A*	N/A*
7/1/1965	4.86	15.75
7/1/1966	3.62	17.78
7/1/1967	3.35	12.86
7/1/1968	1.02	7.77
7/1/1969	3.89	8.96
7/1/1970	6.07	13.96
7/1/1971	6.91	15.08
7/1/1972	7.54	22.75
7/1/1973	5.18	13.45
7/1/1974	7.66	24.76
7/1/1975	5.27	10.68
7/1/1976	5.14	11.59
7/1/1977	5.19	11.58
7/1/1978	5.04	9.61
7/1/1979	2.44	6.68
7/1/1980	5.69	23.05
7/1/1981	3.04	7.22
7/1/1982	5.23	12.06
7/1/1983	5.71	14.48
7/1/1984	4.38	14.14
7/1/1985	2.11	5.93
7/1/1986	0.67	2.88
7/1/1987	3.37	15.98
7/1/1988	0.56	1.96

* Insufficient data for analysis.

Table A-2. Statistical summary of annual ORNL uranium 24-hour urinary excretion rates, 1951-1987.

Effective bioassay date	GM (50th) dpm/day	GM*GSD (84th) dpm/day
7/1/1951	3.41	7.81
7/1/1952	2.88	8.01
7/1/1953	3.64	7.54
7/1/1954	2.27	5.46
7/1/1955	3.07	9.34
7/1/1956	1.20	8.16
7/1/1957	0.50	3.65
7/1/1958	0.64	4.74
7/1/1959	1.43	6.93
7/1/1960	1.36	3.97
7/1/1961	1.52	5.73
7/1/1962	0.52	3.57
7/1/1963	0.58	3.24
7/1/1964	0.25	1.29
7/1/1965	0.17	0.62
7/1/1966	0.18	0.52
7/1/1967	0.09	0.21
7/1/1968	0.13	0.34
7/1/1969	0.10	0.33
7/1/1970	0.16	0.65
7/1/1971	0.09	0.26
7/1/1972	0.13	0.39
7/1/1973	0.07	0.30
7/1/1974	0.06	0.22
7/1/1975	0.08	0.32
7/1/1976	0.10	0.46
7/1/1977	0.20	0.48
7/1/1978	0.14	0.31
7/1/1979	0.14	0.52
7/1/1980	0.11	0.39
7/1/1981	0.13	0.22
7/1/1982	0.17	0.31
7/1/1983	0.12	0.28
7/1/1984	0.10	0.19
7/1/1985	0.07	0.17
7/1/1986	0.04	0.08
7/1/1987	0.00	0.03

Table A-3. Statistical summary of annual ORNL plutonium 24-hour urinary excretion rates, 1951-1988.

Effective bioassay date	GM (50th) dpm/day	GM*GSD (84th) dpm/day
7/1/1951	0.13	0.30
7/1/1952	0.11	0.31
7/1/1953	0.04	0.14
7/1/1954	0.07	0.36
7/1/1955	0.07	0.26
7/1/1956	0.04	0.36
7/1/1957	0.14	0.73
7/1/1958	0.15	0.52
7/1/1959	0.12	0.62
7/1/1960	0.13	0.28
7/1/1961	0.07	0.21
7/1/1962	0.01	0.08
7/1/1963	0.01	0.09
7/1/1964	0.08	0.23
7/1/1965	0.13	0.26
7/1/1966	0.12	0.21
7/1/1967	0.10	0.19
7/1/1968	0.12	0.24
7/1/1969	0.04	0.17
7/1/1970	0.08	0.19
7/1/1971	0.06	0.23
7/1/1972	0.06	0.18
7/1/1973	0.03	0.11
7/1/1974	0.06	0.12
7/1/1975	0.04	0.12
7/1/1976	0.05	0.13
7/1/1977	0.07	0.15
7/1/1978	0.06	0.15
7/1/1979	0.08	0.14
7/1/1980	0.05	0.11
7/1/1981	0.09	0.16
7/1/1982	0.10	0.17
7/1/1983	0.07	0.13
7/1/1984	0.07	0.11
7/1/1985	0.01	0.05
7/1/1986	0.01	0.04
7/1/1987	0.01	0.04
7/1/1988	0.00	0.03

Table A-4. Statistical summary of annual ORNL transplutonium 24-hour urinary excretion rates, 1968-1987.

Effective bioassay date	GM (50th) dpm/day	GM*GSD (84th) dpm/day
7/1/1968	0.12	0.31
7/1/1969	0.10	0.42
7/1/1970	0.10	0.33
7/1/1971	0.09	0.24
7/1/1972	0.12	0.27
7/1/1973	0.09	0.16
7/1/1974	0.08	0.15
7/1/1975	0.12	0.30
7/1/1976	0.15	0.52
7/1/1977	0.11	0.18
7/1/1978	0.10	0.18
7/1/1979	0.10	0.17
7/1/1980	0.10	0.14
7/1/1981	0.07	0.17
1/1/1983	0.10	0.17
7/1/1984	0.10	0.15
7/1/1985	0.05	0.09
7/1/1986	0.01	0.05
7/1/1987	0.01	0.04

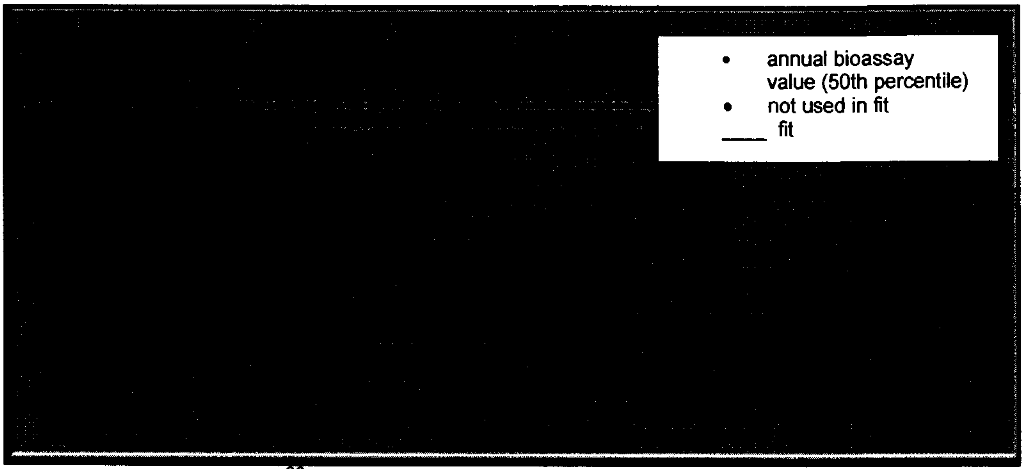

Figure A-1. Assumed ^{90}Sr intake, 1951 to 1988, 50th-percentile results, Type F.

Table A-5. Type F ^{90}Sr intake periods and rates.

Start date	End date	^{90}Sr intake rate (dpm/day) 50th percentile	^{90}Sr intake rate (dpm/day) 84th percentile	GSD
1/1/1951	12/31/1953	475.2	4389	9.24
1/1/1954	12/31/1954	80.99	810.4	10.04
1/1/1955	12/31/1960	47.34	197.3	4.17
1/1/1961	12/31/1964	47.34	62.43	1.32
1/1/1965	12/31/1988	15.52	36.88	2.38

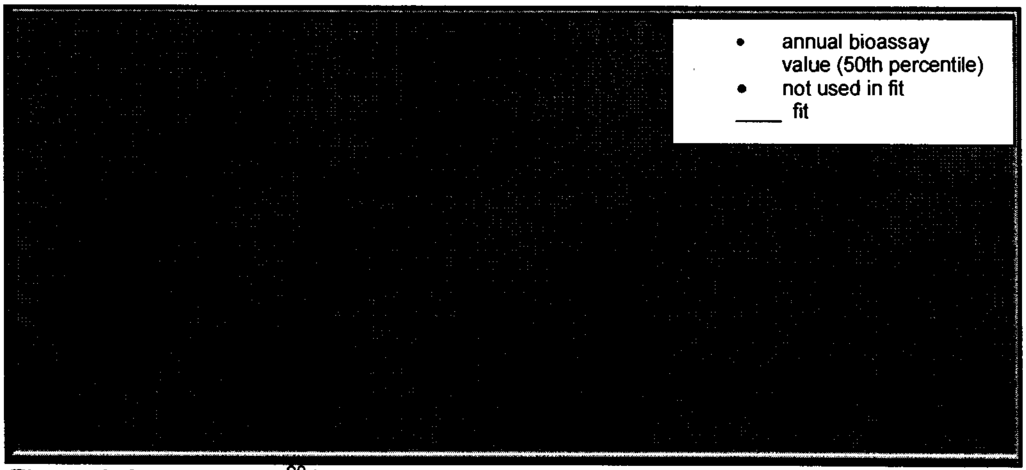

Figure A-2. Assumed ^{90}Sr intake, 1951 to 1953, 50th-percentile results, Type S.

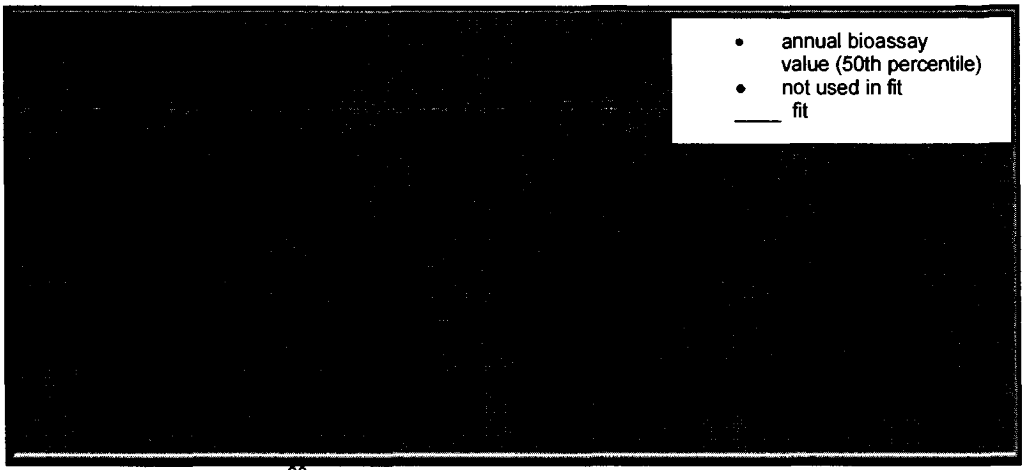

Figure A-3. Assumed ^{90}Sr intake, 1954, 50th-percentile results, Type S.

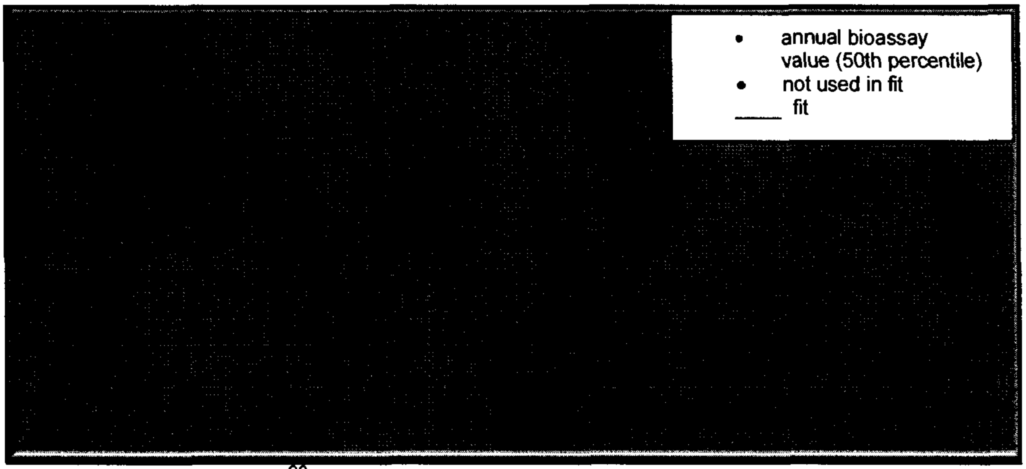

Figure A-4. Assumed ^{90}Sr intake, 1955 to 1964, 50th-percentile results, Type S.

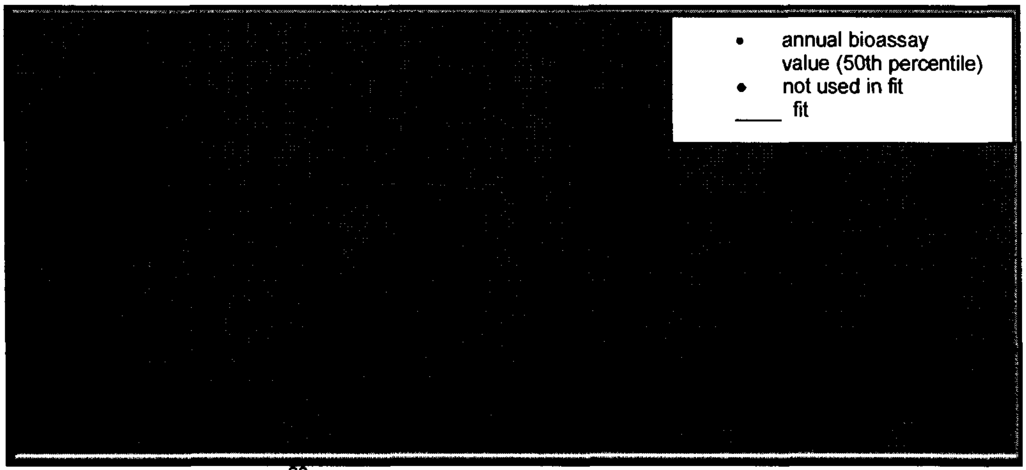

Figure A-5. Assumed ^{90}Sr intake, 1965 to 1983, 50th-percentile results, Type S.

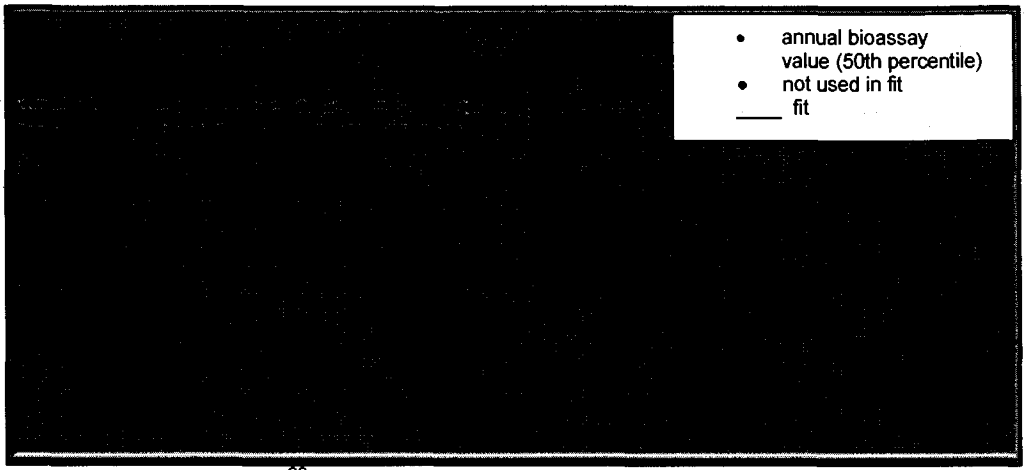

Figure A-6. Assumed ^{90}Sr intake, 1983 to 1988, 50th-percentile results, Type S.

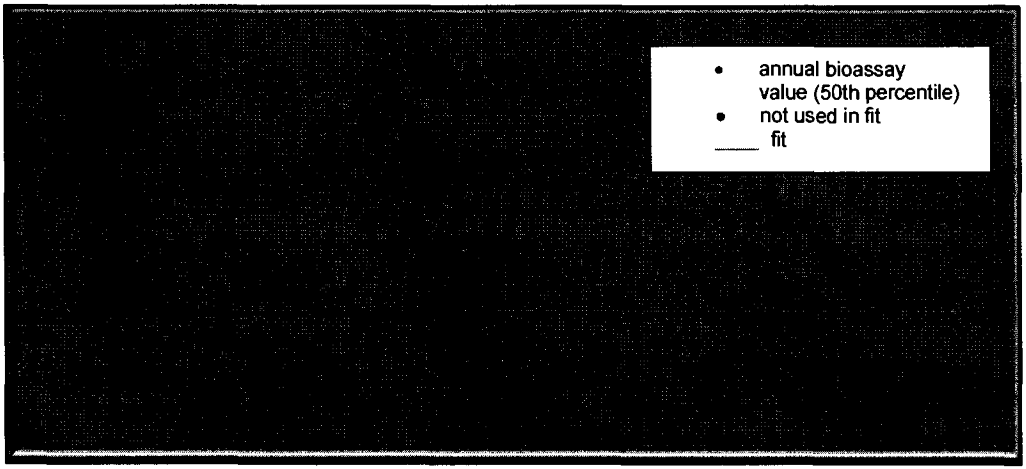

Figure A-7. Predicted strontium excretion rate from independently fit intakes, 1951 to 1988, 50th-percentile, Type S.

Table A-6. Type S ^{90}Sr intake periods and rates.

		^{90}Sr intake rate (dpm/day)		
Start date	**End date**	**50th percentile**	**84th percentile**	**GSD**
1/1/1951	12/31/1953	24646	227730	9.24
1/1/1954	12/31/1954	7,232	68924	9.53
1/1/1955	12/31/1960	2379	13628	5.73
1/1/1961	12/31/1964	2,389	6431	2.69
1/1/1965	12/31/1983	795.0	1,917	2.41
1/1/1984	12/31/1988	425.5	1,917	4.51

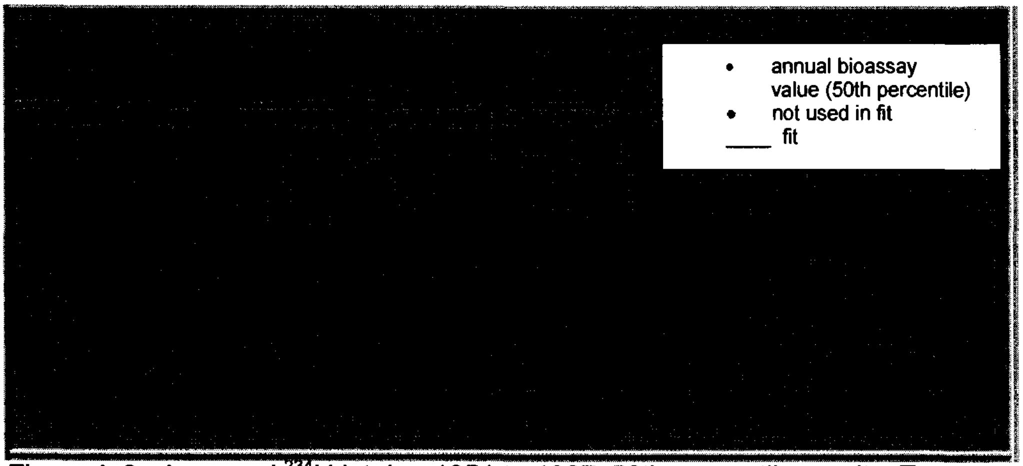

Figure A-8. Assumed ^{234}U intake, 1951 to 1987, 50th-percentile results, Type F.

Table A-7. Type F ^{234}U intake periods and rates.

Year	Type F ^{234}U intake rate (dpm/day)		GSD
	50th percentile	84th percentile	
1951	11.19	28.28	2.53
1952	11.19	28.28	2.53
1953	11.19	28.28	2.53
1954	11.19	28.28	2.53
1955	11.19	28.28	2.53
1956	4.213	28.28	6.71
1957	1.942	17.97	9.25
1958	1.942	17.97	9.25
1959	5.171	17.97	3.48
1960	5.171	17.97	3.48
1961	5.171	17.97	3.48
1962	1.881	12.05	6.41
1963	1.881	12.05	6.41
1964	0.413	1.272	3.08
1965	0.413	1.272	3.08
1966	0.413	1.272	3.08
1967	0.413	1.272	3.08
1968	0.413	1.272	3.08
1969	0.413	1.272	3.08
1970	0.413	1.272	3.08
1971	0.413	1.272	3.08
1972	0.413	1.272	3.08
1973	0.413	1.272	3.08
1974	0.413	1.272	3.08
1975	0.413	1.272	3.08
1976	0.413	1.272	3.08
1977	0.413	1.272	3.08
1978	0.413	1.272	3.08
1979	0.413	1.272	3.08
1980	0.413	1.272	3.08
1981	0.413	1.272	3.08
1982	0.413	1.272	3.08
1983	0.413	1.272	3.08
1984	0.413	1.272	3.08
1985	0.0957	0.227	2.37
1986	0.0957	0.227	2.37
1987	0.0957	0.227	2.37

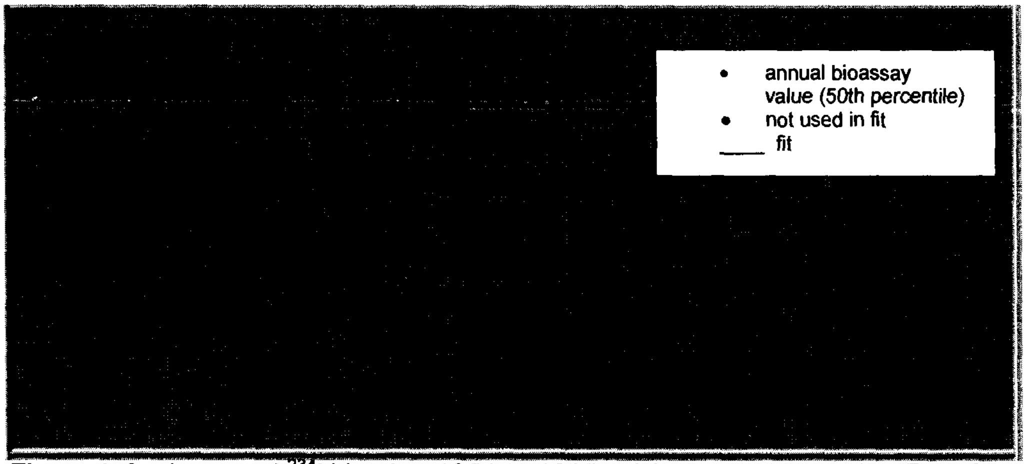

Figure A-9. Assumed ^{234}U intake, 1951 to 1987, 50th-percentile results, Type M.

Table A-8. Type M ^{234}U intake periods and rates.

Year	Type M ^{234}U intake rate (dpm/day)		GSD
	50th percentile	84th percentile	
1951	47.03	117.8	2.50
1952	47.03	117.8	2.50
1953	47.03	117.8	2.50
1954	47.03	117.8	2.50
1955	47.03	117.8	2.50
1956	11.45	117.8	10.29
1957	7.49	72.66	9.70
1958	7.49	72.66	9.70
1959	21.81	72.66	3.33
1960	21.81	72.66	3.33
1961	21.81	72.66	3.33
1962	6.56	47.49	7.24
1963	6.56	47.49	7.24
1964	1.641	4.809	2.93
1965	1.641	4.809	2.93
1966	1.641	4.809	2.93
1967	1.641	4.809	2.93
1968	1.641	4.809	2.93
1969	1.641	4.809	2.93
1970	1.641	4.809	2.93
1971	1.641	4.809	2.93
1972	1.641	4.809	2.93
1973	1.641	4.809	2.93
1974	1.641	4.809	2.93
1975	1.641	4.809	2.93
1976	1.641	4.809	2.93
1977	1.641	4.809	2.93
1978	1.641	4.809	2.93
1979	1.641	4.809	2.93
1980	1.641	4.809	2.93
1981	1.641	4.809	2.93
1982	1.641	4.809	2.93
1983	1.641	4.809	2.93
1984	1.641	4.809	2.93
1985	0.294	0.659	2.24
1986	0.294	0.659	2.24
1987	0.294	0.659	2.24

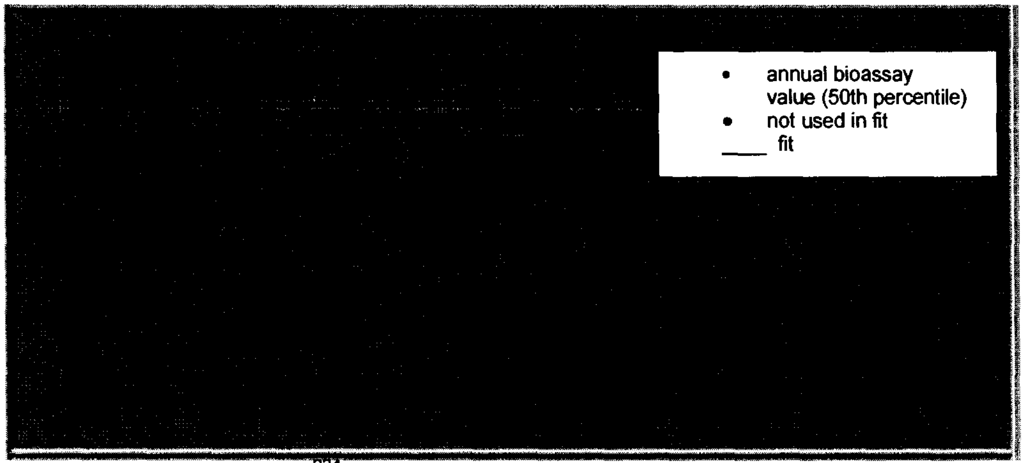

Figure A-10. Assumed ^{234}U intake, 1951 to 1955, 50th-percentile results, Type S.

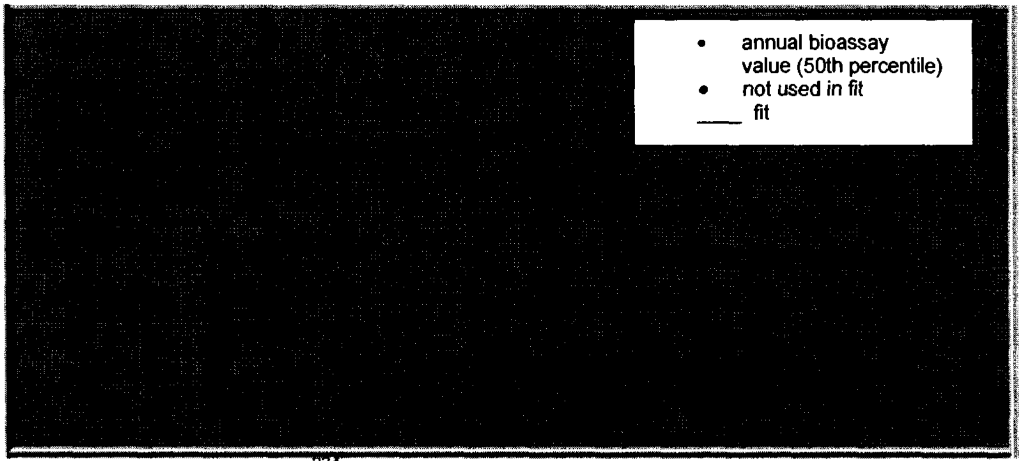

Figure A-11. Assumed ^{234}U intake, 1956, 50th-percentile results, Type S.

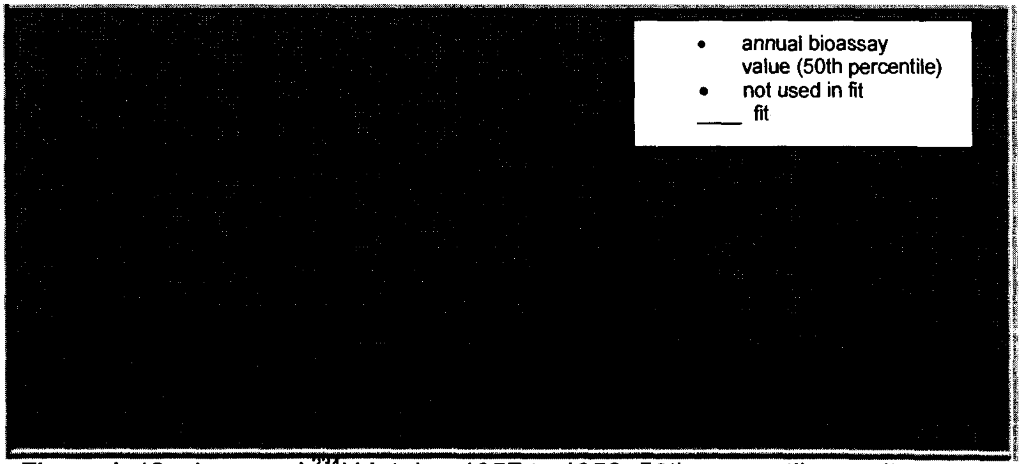

Figure A-12. Assumed ^{234}U intake, 1957 to 1958, 50th-percentile results, Type S.

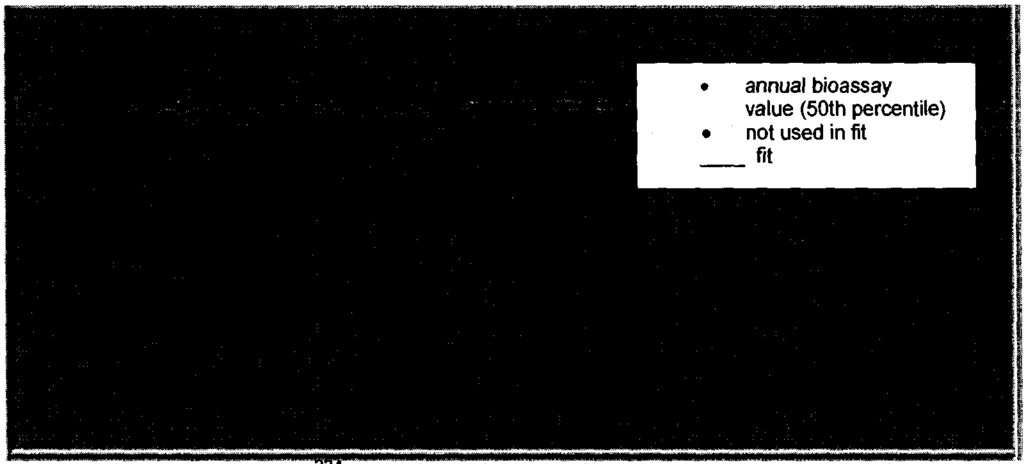

Figure A-13. Assumed ^{234}U intake, 1959 to 1961, 50th-percentile results, Type S.

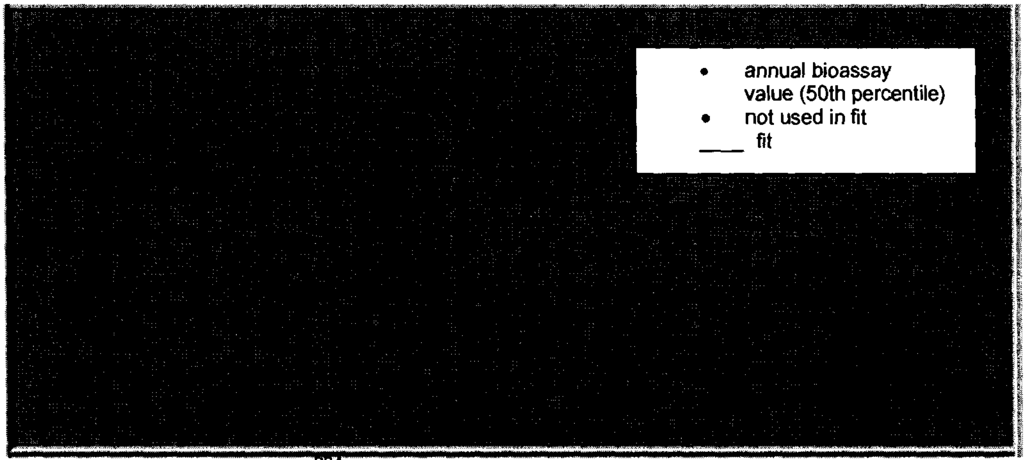

Figure A-14. Assumed ^{234}U intake, 1962 to 1963, 50th-percentile results, Type S.

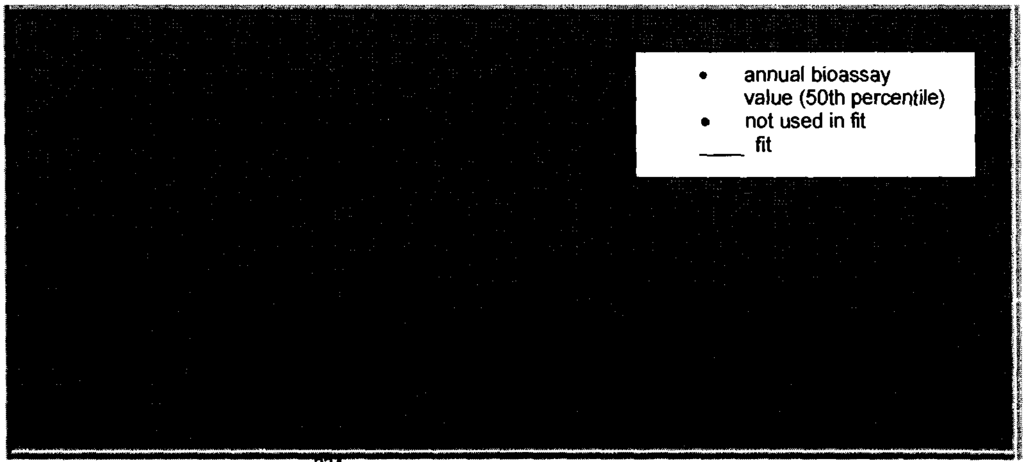

Figure A-15. Assumed ^{234}U intake, 1964 to 1982, 50th-percentile results, Type S.

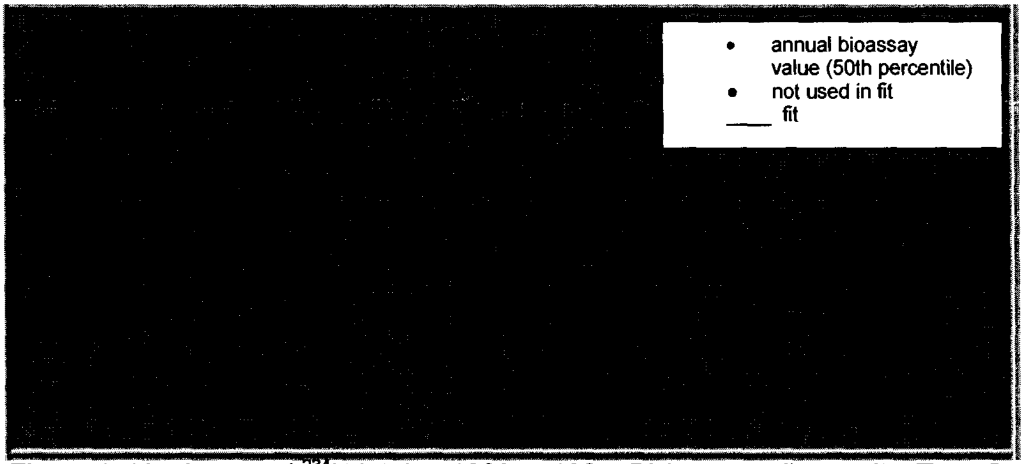

Figure A-16. Assumed ^{234}U intake, 1983 to 1987, 50th-percentile results, Type S.

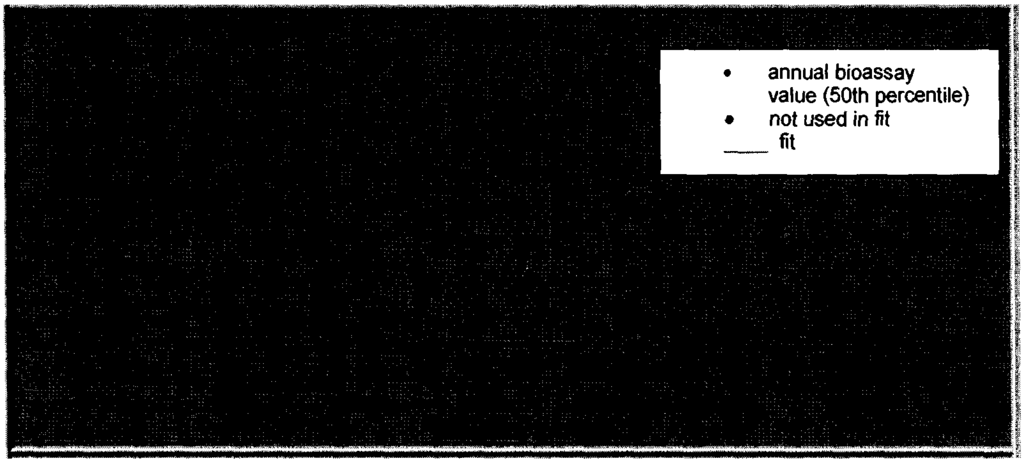

Figure A-17. Predicted uranium excretion rate from independently fit intakes, 1951 to 1987, 50th-percentile, Type S.

Table A-9. Type S ^{234}U intake periods and rates.

Year	Type S ^{234}U intake rate (dpm/day)		GSD
	50th percentile	84th percentile	
1951	850.7	2048	2.41
1952	850.7	2048	2.41
1953	850.7	2048	2.41
1954	850.7	2048	2.41
1955	850.7	2048	2.41
1956	675.7	2048	3.03
1957	247.7	1,917	7.74
1958	247.7	1,917	7.74
1959	509	1,917	3.77
1960	509	2,125	4.17
1961	509	2,125	4.17
1962	235.3	1,422	6.04
1963	235.3	1,422	6.04
1964	23.7	71.203	3.00
1965	23.7	71.203	3.00
1966	23.7	71.203	3.00
1967	23.7	71.203	3.00
1968	23.7	71.203	3.00
1969	23.7	71.203	3.00
1970	23.7	71.203	3.00
1971	23.7	71.203	3.00
1972	23.7	71.203	3.00
1973	23.7	71.203	3.00
1974	23.7	71.203	3.00
1975	23.7	71.203	3.00
1976	23.7	71.203	3.00
1977	23.7	71.203	3.00
1978	23.7	71.203	3.00
1979	23.7	71.203	3.00
1980	23.7	71.203	3.00
1981	23.7	71.203	3.00
1982	23.7	71.203	3.00
1983	15.25	71.203	4.67
1984	15.25	31.58	2.07
1985	15.25	31.58	2.07
1986	15.25	31.58	2.07
1987	15.25	31.58	2.07

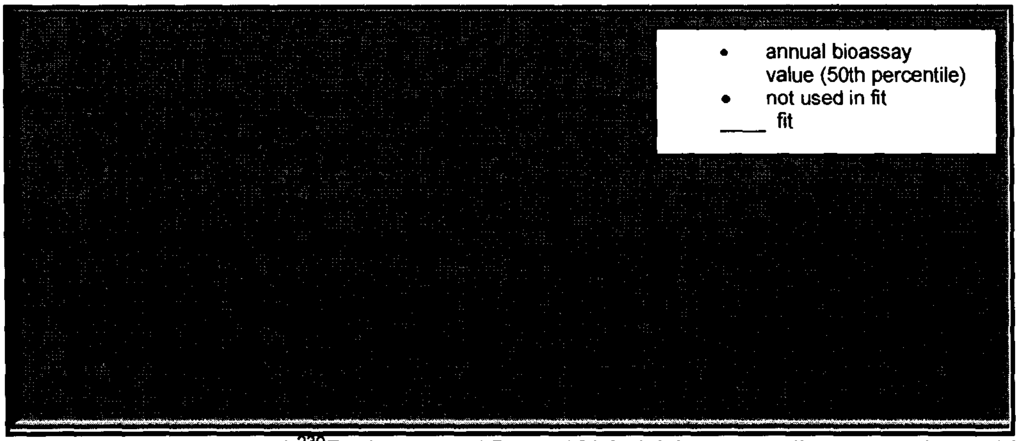

Figure A-18. Assumed ^{239}Pu intake, 1951 to 1952, 50th-percentile results, Type M.

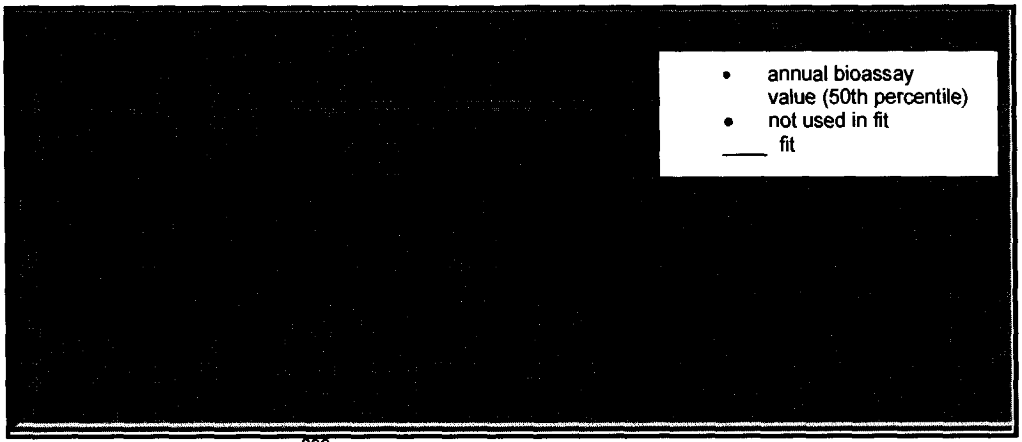

Figure A-19. Assumed ^{239}Pu intake, 1953 to 1968, 50th-percentile results, Type M.

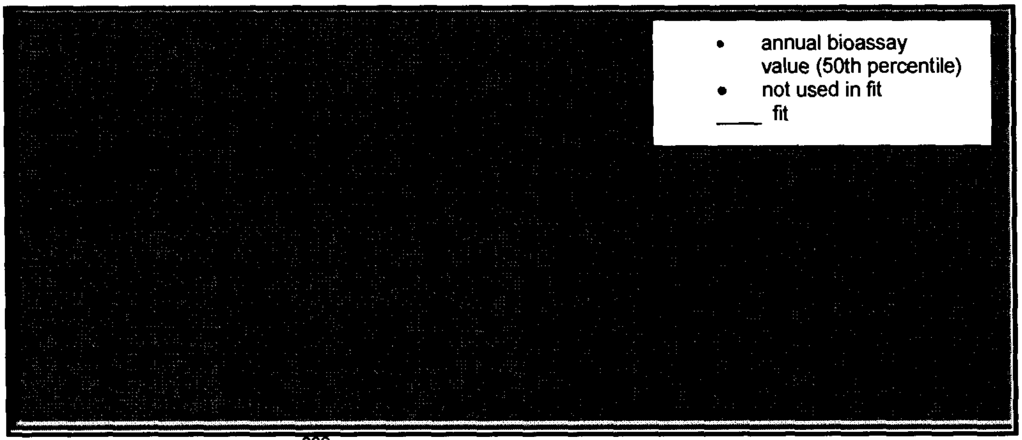

Figure A-20. Assumed ^{239}Pu intake, 1969 to 1984, 50th-percentile results, Type M.

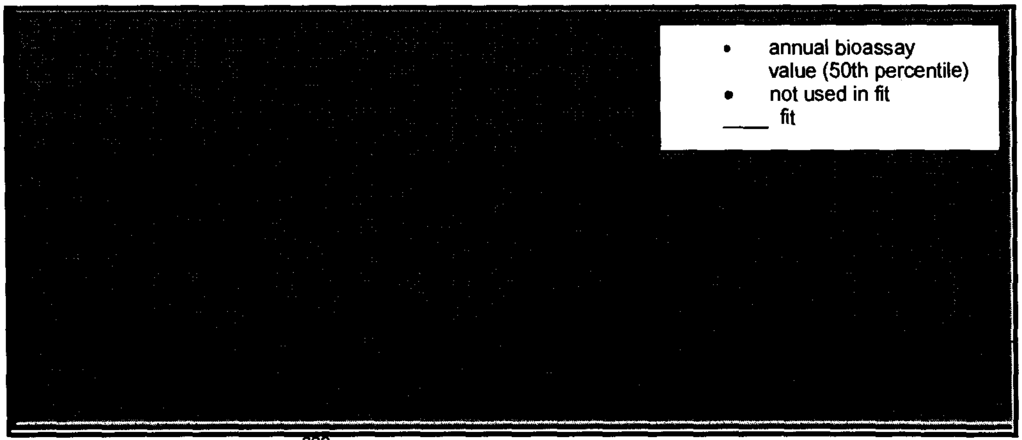

Figure A-21. Assumed ^{239}Pu intake, 1985 to 1988, 50th-percentile results, Type M.

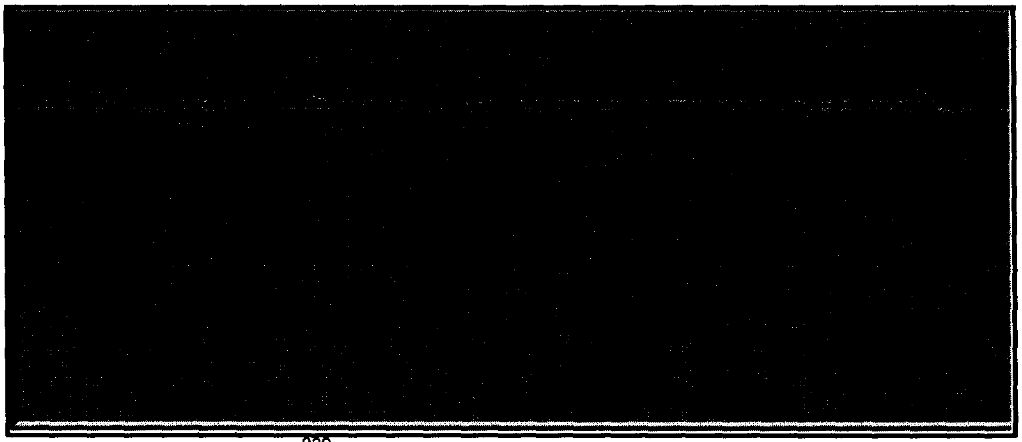

Figure A-22. Predicted ^{239}Pu intake, 1951 to 1988, 50th-percentile composite results, Type M.

Table A-10. Type M ^{239}Pu intake periods and rates.

	^{239}Pu intake rate (dpm/day)		
Year	**50th percentile**	**84th percentile**	**GSD**
1951	40.75	109.8	2.69
1952	40.75	109.8	2.69
1953	10.98	86.42	7.87
1954	10.98	86.42	7.87
1955	10.98	86.42	7.87
1956	10.98	86.42	7.87
1957	10.98	86.42	7.87
1958	10.98	86.42	7.87
1959	10.98	86.42	7.87
1960	10.98	22.45	2.04
1961	10.98	22.45	2.04
1962	10.98	22.45	2.04
1963	10.98	22.45	2.04
1964	10.98	22.45	2.04
1965	10.98	22.45	2.04
1966	10.98	22.45	2.04
1967	10.98	22.45	2.04
1968	10.98	22.45	2.04
1969	7.35	22.45	3.05
1970	7.35	22.45	3.05
1971	7.35	22.45	3.05
1972	7.35	22.45	3.05
1973	7.35	18.4	2.50
1974	7.35	18.4	2.50
1975	7.35	18.4	2.50
1976	7.35	18.4	2.50
1977	7.35	18.4	2.50
1978	7.35	18.4	2.50
1979	7.35	18.4	2.50
1980	7.35	18.4	2.50
1981	7.35	18.4	2.50
1982	7.35	18.4	2.50
1983	7.35	18.4	2.50
1984	7.35	18.4	2.50
1985	1.614	6.811	4.22
1986	1.614	6.811	4.22
1987	1.614	6.811	4.22
1988	1.614	6.811	4.22

Figure A-23. Assumed ^{239}Pu intake, 1951 to 1988, 50th-percentile composite results, Type S.

Table A-11. Type S ^{239}Pu intake period and rates.

Start date	Stop date	^{241}Am intake rate (dpm/day)		GSD
		50th percentile	84th percentile	
1951	1988	4.15	22.86	5.50

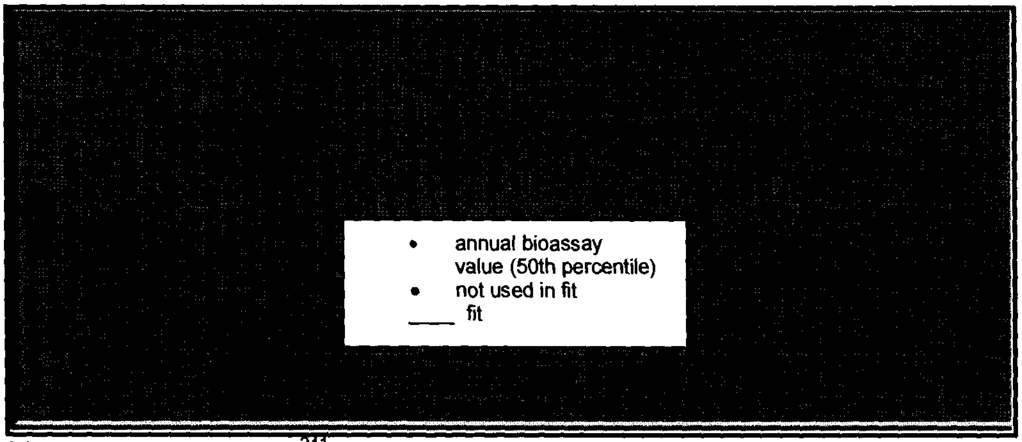

Figure A-24. Assumed ^{241}Am intake, 1968 to 1984, 50th-percentile results, Type M.

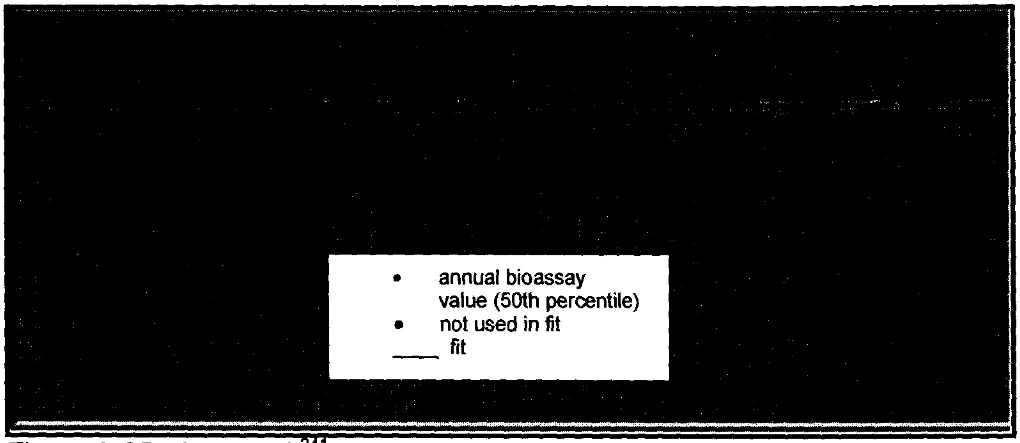

Figure A-25. Assumed ^{241}Am intake, 1985 to 1988, 50th-percentile results, Type M.

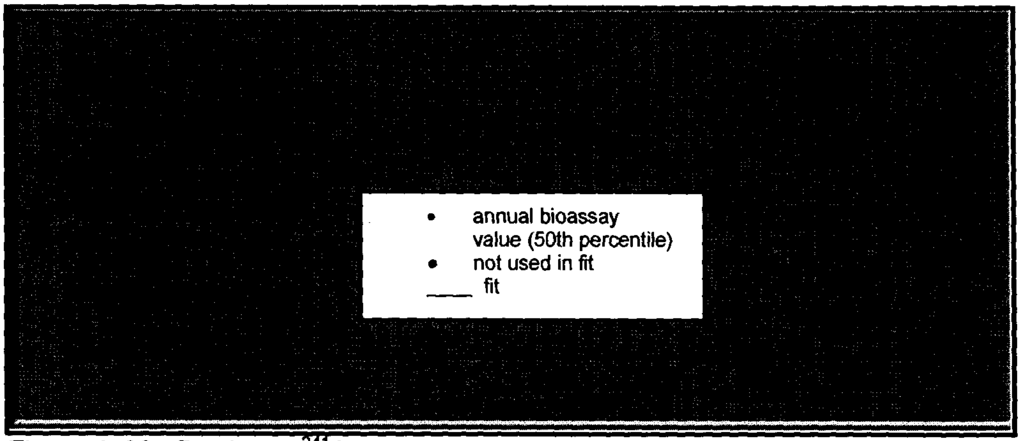

Figure A-26. Predicted ^{241}Am intake, 1968 to 1988, 50th-percentile composite results, Type M.

Table A-12. Type M ^{241}Am intake periods and rates.

		^{241}Am intake rate (dpm/day)		
Start date	**Stop date**	**50th percentile**	**84th percentile**	**GSD**
1968	1984	6.673	14.61	2.19
1985	1988	2.207	6.659	2.99

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☐ New ☐ Total Rewrite ☒ Revision ☐ Page Change

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
11/23/2004	00-A	New technical information bulletin to provide information for internal dose overestimates for facilities with air sampling programs. Initiated by Donald E. Bihl.
01/06/2005	00-B	Incorporates internal review comments. Initiated by Donald E. Bihl.
03/10/2005	00-C	Incorporates changes requested by NIOSH, including Attachments B and C. Updates RU contaminant levels. Eliminates discussion about ORAUT-OTIB-0002 related to ORAUT-OTIB-0018. Initiated by Donald E. Bihl.
03/18/2005	00	First approved issue. Initiated by Donald E. Bihl
05/05/2005	01-A	Draft revision to 1) incorporate into Table 4-3 Nb-95 type S and Tc-99 type F beta choice and replace RU with U-234 alpha choice, 2) add Table 7-2 which provides excretion values for uranium and radiostrontium below which the default intakes was still valid, 3) add special air concentration values for LANL into Table 4-1,4) correct error in the ingestion intake formula #4. Text was added or modified in Sections 1.0 through 5.0, 7.0, and Attachment A. No entire Sections were deleted. Initiated by Donald E. Bihl.
06/10/2005	01-B	No further changes occurred as a result of formal internal review. Initiated by Elizabeth M. Brackett and Donald E. Bihl.
07/15/2005	01-C	Incorporates NIOSH review comments. Initiated by Donald E. Bihl.
08/09/2005	01	Approved issue of Revision 01. Training required: As determined by the Task Manager. Initiated by Donald E. Bihl.

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ACRONYMS AND ABBREVIATIONS

1.0 PURPOSE

The purpose of this technical information bulletin (TIB) is to provide an alternative method to that discussed in Technical Information Bulletin 002, *Maximum Internal Dose Estimates for Certain DOE Complex Claims* (ORAU 2004), for estimating intakes and internal doses when overestimated doses are acceptable. The internal doses estimated using this TIB will be smaller than those obtained using ORAU (2004) under some situations but are still overestimates. See Section 7.0 for additional discussion about when to use this TIB rather than ORAU (2004).

Technical Information Bulletins (TIBs) are general working documents that provide guidance concerning the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained. TIBs may be used to assist the National Institute for Occupational Safety and Health in the completion of individual dose reconstructions.

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

2.0 BACKGROUND

ORAU (2004) provides default overestimated intakes based on the assumption "that an intake resulting in 10% of a MPBB [maximum permissible body burden] would not have likely occurred to an unmonitored worker or would have likely resulted in a readily noticeable bioassay result in a monitored worker, readily noticeable air sample, or other indicators of personnel contamination." ORAU (2004, Table 3.1.1-2) applies this assumption to each of a list of radionuclides to which workers at most U.S. Department of Energy (DOE)[1] sites might have been exposed. The list includes three beta/gamma-emitting radionuclides and nine alpha-emitting radionuclides for nonreactor sites and an additional 16 beta/gamma-emitting radionuclides for sites with reactors. However, the ORAU (2004) method resulted in unrealistically high intakes of long-lived, long-retained radionuclides when the exposure time was only a few years; that is, the air concentrations would have had to have been unrealistically high for a worker to have incurred unnoticed intakes resulting in 10% of the MPBB.

For efficiency in processing claims, another method was needed to estimate intakes that are

- Overestimates
- Derived from more realistic air concentrations

Most DOE sites had air sampling programs for particulate radioactive contamination and, with the exception of short-term, unplanned situations, controlled air concentrations to limiting values established by the National Council on Radiation Protection and Measurements (NCRP) or DOE. The first official national guidelines on air concentrations were produced by the NCRP and published in 1953 in National Bureau of Standards (NBS) Handbook 52 (NBS 1953). The NCRP revised its methodology and expanded the list of covered radionuclides in 1959, published as NBS Handbook 69 (NBS 1959). Both of these documents were based on and consistent with similar guidance from the International Commission on Radiological Protection (ICRP 1959). Subsequent to the publication of the NBS handbooks, DOE issued orders or regulations establishing air concentration limits applicable to its sites [for instance, in U.S. Atomic Energy Commission (AEC) Manual Chapter 0524 (AEC 1968)

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[1] In this document, reference to DOE includes its predecessor agencies.

and DOE Order 5480.11 (DOE 1988)]. Since December 1993, 10 CFR Part 835 has established limits on air concentrations. Table 2-1 lists the history of workplace air concentration limits applicable at DOE sites for a number of important particulate radionuclides.

Table 2-1. Limiting air concentrations for selected particulates from 1953 to present.

	Limiting air concentration (µCi/mL)				
Radionuclide[a]	**NBS 1953**	**NBS 1959**	**AEC Manual Chapter 0524[b]**	**DOE Order 5480.11[c]**	**10 CFR 835**
Alpha emitters					
Po-210	7E-11	2E-10	2E-10	3E-10	3E-10
Th-230	NL[d]	2E-12	2E-12	3E-12	3E-12
Th-232	NL	2E-12[e]	3E-11[f]	5E-13	5E-13
U natural sol	1.7E-11	7E-11	7E-11[f]	NL	NL
U natural insol	1.7E-11	6E-11	6E-11[f]	NL	NL
U-234 sol (D)	NL	6E-10	6E-10	5E-10	5E-10
U-234 insol (Y)	NL	1E-10	1E-10	2E-11	2E-11
U-238 sol (D)	NL	7E-11	7E-11	6E-10	6E-10
U-238 insol (Y)	NL	1E-10	1E-10	2E-11	2E-11
Np-237	NL	4E-12	4E-12	2E-12	2E-12
Pu-238	NL	2E-12	2E-12	3E-12	3E-12
Pu-239	2E-12	2E-12	2E-12	2E-12	2E-12
Am-241	3E-11	6E-12	6E-12	2E-12	2E-12
Cm-244	NL	9E-12	9E-12	4E-12	4E-12
Beta emitters					
Mn-54	NL	4E-8	4E-8	3E-7	3E-7
Co-58	NL	5E-8	5E-8	3E-7	3E-7
Co-60	1E-6	9E-9	9E-9	1E-8	1E-8
Fe-59	6E-7	5E-8	5E-9	1E-7	1E-7
Zn-65	2E-6	6E-8	6E-8	1E-7	1E-7
Sr-90	2E-10	3E-10	1E-9	2E-9	2E-9
Y-91	4E-8	3E-8	3E-8	5E-8	5E-8
Nb-95	4E-7	1E-7	1E-7	5E-7	5E-7
Zr-95	NL	3E-8	3E-8	6E-8	6E-8
Tc-99	NL	6E-8	6E-8	3E-7	3E-7
Ru-103	NL	8E-8	8E-8	3E-7	3E-7
Ru-106	3E-8	6E-9	8E-8	5E-9	5E-9
Cs-134	NL	1E-8	1E-8	4E-8	4E-8
Cs-137	2E-7	1E-8	1E-8	7E-8	7E-8
Ba-140	6E-8	4E-8	4E-8	6E-7	6E-7
La-140	6E-8	1E-7	1E-7	5E-7	5E-7
Ce-141	NL	2E-7	2E-7	3E-7	2E-7
Ce-144	7E-9	6E-9	6E-9	6E-9	6E-9
Pm-147	2E-7	1E-7	6E-8	6E-8	6E-8
Eu-152	NL	3E-7	2E-8	1E-8	1E-8
Eu-154	6E-9	4E-9	7E-9	8E-9	8E-9
Eu-155	NL	7E-8	7E-8	4E-8	4E-8

a. Most limiting form (e.g., soluble, insoluble, inhalation class D,W,Y) is listed, with the exception of uranium, for which the soluble form was based on chemical toxicity.

b. From 1968 version

c. From 1988 version (effective January 1989)

d. NL – not listed.

e. Printed as 2E-12 µCi/mL, but a footnote stated that continued use of 3E-11 µCi/mL was recommended until further investigation.

f. Based on special definitions for the curie that added disintegrations from selected progeny.

Most sites performed total alpha and total beta counting on air samples, using either a time delay or mathematical correction to account for radon progeny. Radionuclide identification was either not performed or was performed occasionally as a check on basic assumptions from knowledge of the facility source terms and processes. Trigger levels and prompt decisions for changing access to an area, usually by placing a "respiratory protection required" restriction for the area, were usually based on total alpha or total beta air concentrations. Because the radionuclide was not identified for every sample, the limit for the most restrictive, plausible radionuclide was often used to establish the trigger level. For facilities handling transuranics, the most restrictive air concentration for the alpha measurements was based on plutonium; for facilities with potential exposure to fission or activation products, the most restrictive air concentration for beta measurements was based on ^{90}Sr; at uranium facilities, the limits for uranium were generally used.

The validity of the assumption that workers' intakes, excepting accidental intakes, were controlled such that intakes at limiting air concentrations for a chronic period are reasonable median values is discussed in Appendix B. Appendix B uses worker bioassay results.

3.0 APPLICABILITY, LIMITATIONS, AND ASSUMPTIONS

3.1 APPLICABILITY

This TIB applies to:

- Sites or facilities that rigorously sampled particulate air concentrations in areas of risk and controlled exposure to intakes according to the measured concentrations.
- Employment between 1953 and the present. (See first bullet in Section 3.2 for an exception.)
- Claims for which it is likely that the covered employee had no significant intakes of particulate radioactive material. (See Section 7.0 for a discussion on insignificant intakes in the context of this TIB.)
- Intakes of particulate radioactive material only.
- All organs except respiratory tract organs for monitored workers and the thyroid; however, it can apply to the thyroid for uranium facilities or plutonium-only facilities (i.e., where there is no chance of exposure to radioiodines).

3.2 LIMITATIONS

This TIB has the following limitations:

- It does not apply to employment prior to 1953; however, if the limiting air concentrations for the site of exposure are known for years prior to 1953, those concentrations can be used in the manner discussed in Sections 4.0 and 5.0. In addition, see Attachment A. It does not apply to Nevada Test Site outdoor exposures prior to 1963.
- It applies only to particulate radioactive material. These particulate intakes are in addition to any intakes of ^{3}H, radioiodines, ^{14}C, or radon/thoron and their progeny, as applicable.

- Intakes of particulate material with a documented particle-size distribution other than 5-µm activity median aerodynamic diameter (AMAD) must be checked to ensure that the dose to the organ of concern from this TIB is greater.

- If site-specific limiting air concentrations were greater than those listed in Section 3.3, those concentrations must be used. If the site-specific concentration was applicable to contamination for a given radionuclide only and exposure was limited to that radionuclide, the dose reconstructor can apply the intake to that radionuclide only. For instance, Mound and Los Alamos National Laboratory had special limits for exposure to ^{210}Po and Los Alamos had special limits for general alpha-emitters and beta-emitters through 1966.

- It does not necessarily provide an overestimate to respiratory tract organs for monitored workers.

- It does not provide an overestimate to the thyroid (regardless of monitoring), unless the worker was employed only at a uranium facility or a plutonium-only facility (i.e., where there was no chance of exposure to radioiodines).

3.3 ASSUMPTIONS

This TIB applies the following assumptions:

- All significant intake exposure was covered by an air sampling program, or the radiation protection program had a valid method to ensure that air concentrations did not exceed limits. (For instance, airborne contamination might have been controlled by controlling the inventory of radioactive material in an area and/or the physical form of the material.)

- Chronic intakes were for 40 hours per week, or 2,000 hours per year, of particulate radioactive material with a 5-µm-AMAD particle-size distribution.

- The breathing rate was 9.6 m^3/workday (ICRP 1994a) or 1.2 m^3/hr averaged over an 8-hr day.

- For sites or facilities with an exposure to alpha-emitting radionuclides that was principally transuranics, chronic exposure was at 2×10^{-12} µCi/mL.

- For sites or facilities with an exposure to beta-emitting radionuclides, chronic exposure was at 2×10^{-9} µCi/mL.

- For sites or facilities with an exposure principally to uranium, chronic exposure was at 6×10^{-10} µCi/mL.

- Measured air sample concentrations were not always representative of the air concentration breathed. However, when applied to the long-term, 40-hour-per-week, chronic-intake assumption, it is assumed that the air sampling program was sufficient to prevent long-term intakes from exceeding the limiting air concentrations. See Section 6.0 for additional consideration of the possible nonrepresentativeness of air sampling.

4.0 INHALATION INTAKE

4.1 INTAKE QUANTITY

The inhalation intake is determined by

$$\text{Intake} = (\text{air conc.})(\text{breathing rate})(\text{exposure period}) \qquad (1)$$

From Equation 1, intakes in Bq/yr can be determined by

$$\text{Intake (Bq/yr)} = (\text{air conc. } \mu\text{Ci/mL})(1.2\ \text{m}^3/\text{hr})(2{,}000\ \text{hr/yr})(10^6\ \text{mL/m}^3)(3.7 \times 10^4\ \text{Bq}/\mu\text{Ci})$$

or

$$\text{Intake (Bq/yr)} = (8.9 \times 10^{13})(\text{air conc. } \mu\text{Ci/mL}) \qquad (2)$$

where the air concentration is obtained from Table 4-1 or from site-specific limiting air concentrations (in µCi/mL) when applicable.

Table 4-1. Default limiting air concentrations.

Exposure type	Air concentration (µCi/mL)	Associated daily intake (pCi)
Alpha-emitting radionuclides except for uranium facilities (includes Hanford from 1949, see Attachment A)	2E-12	13.2
Beta-emitting radionuclides except for uranium facilities (includes Hanford from 1949, see Attachment A)	2E-9	1.32E4
Uranium including recycled uranium (includes Hanford from 1949, see Attachment A, and Los Alamos from 1951)	6E-10	3.96E3
ORNL alpha-emitting radionuclides except for uranium facilities, 1944-1952	3E-11	1.98E2
ORNL beta-emitting radionuclides, 1944-1952	1E-7	6.60E5
Hanford alpha-emitting radionuclides except uranium facilities, 1946-1949	4E-11	2.64E2
Hanford beta-emitting radionuclides, 1946-1949.	1E-8	6.60E4
Los Alamos alpha-emitting radionuclides except uranium facilities or Po-210 facilities, 1948-52	3.2E-11	
Los Alamos alpha-emitting radionuclides except uranium facilities or Po-210 facilities, 1953-66	4E-12	
Los Alamos beta-emitting radionuclides, 1948-52	1E-7	
Los Alamos beta-emitting radionuclides, 1953-66	3E-9	
Los Alamos uranium facilities, 1948-50	1.8E-10	
Los Alamos Po-210 only[a], 1948-53 (exposure ended in1953)	6.3E-10	
Mound Po-210 only[a], 1952-71, (exposure ended in 1971)	NA	5.88E3
Rocky Flats alpha-emitting radionuclides in Pu chemistry operations or default for the site, 1953-1994 (2E-12 µCi/ml was used in the metal operations).	4 E-12	26.4

a. Los Alamos and Mound had special air concentration limits or urinalysis limits for Po-210. The doses associated with this intake will be larger than for general alpha-emitting radionuclides at 13.2 pCi/d for years of exposure but will be less for years after exposure ends. The dose reconstructor will have to run both cases and determine which option gives the largest probability of causation.

The intake per calendar day in pCi is

$$\text{Intake (pCi/cal. d)} = (\text{air conc. } \mu\text{Ci/mL})(1.2\ \text{m}^3/\text{hr})(2{,}000\ \text{hr/yr})(10^6\ \text{mL/m}^3)(10^6\ \text{pCi}/\mu\text{Ci})/365\ \text{cal. d/yr}$$

or

$$\text{Intake (pCi/cal. d)} = (6.6 \times 10^{12})(\text{air conc.}) \quad (3)$$

For most sites intakes will be assigned for alpha and beta intakes. Only the uranium intake should be assigned for facilities with exposure to only uranium (or natural thorium), with the exception of years when exposure was to recycled uranium. For recycled uranium, add intakes as listed in Table 4-2.

Table 4-2. Intakes of contaminants in recycled uranium as fraction of uranium intake.

Radionuclide	Activity fraction of contaminant (e.g. pCi X/pCi U)
Pu-238	0.06
Np-237	0.005
Tc-99	0.4
Th-232	0.02
Ru-106	0.04

4.2 CHOICE OF RADIONUCLIDE

Because total alpha and total beta activity were measured on the air filters, the measurements represented all the particulate activity being breathed (as opposed to bioassay measurements that usually measure only certain radionuclides such that unmeasured radionuclides have to be accounted for in addition to the measured radionuclides). To be most accurate, the fractions of the total activity (alpha or beta) would have to be assigned to each radionuclide in the mixture in the air breathed. Those fractions were generally not determined on a regular basis and would have varied among sites, facilities, processes, or even specific work tasks.

Rather than estimating radionuclide fractions, this TIB overestimates the internal dose by assigning 100% of the intake to the single radionuclide that produces the largest dose per unit intake to the organ of concern. In addition, organ dose depends on the absorption type of the radionuclides. The radionuclide and absorption type combination that produces this "largest dose" can differ by organ, time of exposure, and time after end of exposure. Because the annual organ dose is the dose of interest, and the annual dose varies from year to year for a given radionuclide, the largest dose contributor could change over the years. For example, for a 10-year chronic intake of beta emitters, ^{134}Cs type F delivers the largest dose to the urinary bladder during years 1 through 10. However, if the date of diagnosis is sometime after the intake period, the assumption of a ^{106}Ru type F intake during the 10 years of exposure yields the largest dose for years 11 through 14, as does ^{90}Sr type F for all subsequent years. Rather than determining which radionuclide and absorption type combination produces the largest dose, an additional claimant-favorable assumption was made. Each year between the start of exposure and date of diagnosis is evaluated individually and the largest dose is assigned. For the previous example, it would be assumed that the intake was comprised of 100% type F ^{134}Cs when assigning years 1 through 10. For years 11 through 14, it would be assumed that the intake had been entirely made up of type F ^{106}Ru, and that it was ^{90}Sr for years 15 through 65.

Table 4-3 lists the inventory of radionuclides and absorption types from which the combination producing the largest dose can be obtained. This is the default list at present.

Notes: Ac-227 is a progeny of ^{231}Pa in the ^{235}U decay chain. It was probably not present in significant quantities unless ^{231}Pa or ^{227}Ac was purposely concentrated; therefore, it was not included in the

Table 4-3. Inventory of radionuclides and absorption types.

Radionuclides/absorption types associated with the alpha choice

Am-241	M, S	Pu-240	M, S
Cf-252	M	Pu-242	M, S
Cm-244	M	Ra-226	M
Np-237	M	Th-228	M, S
Po-210	F, M	Th-230	M, S
Pu-238	M, S	Th-232	M, S
Pu-239	M, S	U-234	F, M, S

Radionuclides/absorption types associated with the beta choice

Ag-110	M, S	Fe-59	F, M	S-35	F, M
Ba-140	F	Hf-181	F, M	Sb-125	F, M
Ce-141	M, S	La-140	F, M	Sc-46	S
Ce-144	M, S	Mn-54	F, M	Sn-113	F, M
Co-58	M, S	Mo-99	F, S	Sr-89	F
Co-60	M, S	Na-24	F	Sr-90	F
Cr-51	F, M, S	Nb-95	M, S	Tb-160	M
Cs-134	F	Ni-63	F, M	Tc-99	F, M
Cs-137	F	P-32	F, M	Y-91	M, S
Eu-152	M	Pm-147	M, S	Zn-65	S
Eu-154	M	Ru-103	F, M, S	Zr-95	F,M,S
Eu-155	M	Ru-106	F, M, S		

Radionuclides/absorption types associated with the uranium choice

Th-232	M, S	RU	F, M, S

a. RU = U-234 plus the contaminants from Table 4-2 in the ratios provided in Table 4-2.

default list for the alpha choice. It should be added to the list if ^{231}Pa or ^{227}Ac was handled in a pure or concentrated form at the site in question.

Th-232 is included in the uranium list because many uranium facilities also processed thorium at some time. Th-232 can be removed from the list if it is known for sure that the facility did not handle thorium. Removal of ^{232}Th will reduce dose from 2 to 700 times depending on organ and length of exposure.

For strontium, type S is applicable to the titanate form only and is unlikely to be present at most sites. Type S was not included in the default list for the beta choice. If it is established that a site had strontium titanate, type S strontium will have to be added to the list. Check the Site Profile for applicability. Adding type S strontium will significantly increase the dose for some organs.

5.0 INGESTION INTAKE

The possibility of ingestion intakes has to be considered separately from inhalation intakes when intake is not based on bioassay measurements. *Estimation of Ingestion Intakes* (OCAS 2004) provides the method for estimating ingestion intakes based on air concentrations:

$$\text{Ingestion intake} = 0.021\ (\text{inhalation intake}) \qquad (4)$$

These intakes would be applied to the same years for which inhalation intakes occurred, using the same radionuclide. Use the f_1 associated with the absorption type from Annex F in ICRP Publication 68 (ICRP 1994b).

6.0 DOSE DISTRIBUTION

Assuming an intake at the limiting air concentration for every minute of employment is probably an overestimate. However, there are sources of uncertainty. For instance,

- The air sample results might not have been representative of the air breathed. If a worker was close to a localized source, the air concentration the worker breathed might have been higher than the air collected by a sampler several to tens of feet away. Offsetting this problem somewhat is the likelihood that the worker breathed larger particles than the air sampler collected. The time spent near a localized source would have been less than full time.

- Enforcement of the limiting air concentrations at some facilities for some periods might have been lax.

- Prior to widespread use of alarming continuous air monitors (CAMs), exposure to air concentrations above limiting air concentrations for periods of days to a few weeks was possible. For instance, a filter exchanged weekly with delayed counting for radon decay could easily result in a 2-week lag between the change in air concentration and change in access control. Even with the use of CAMs, radon progeny interference was a problem with alpha monitoring in many facilities such that immediate recognition of an air concentration slightly greater than the plutonium limit was difficult.

- For beta emitters, the counting efficiency for converting counts to total activity is dependent on the beta energy. If the beta energy of the radionuclide on the filter is less than the beta energy of the calibration source, the total activity would have been underestimated. This is offset somewhat by the generally lower dose per unit intake from the lower energy beta emitters. For alpha emitters, self-absorption of the alphas due to high dust loading might not have been compensated for.

- The volume of air sampled might not have been measured accurately; for instance, it was common to use flow rate meters and the average of the on/off flow rate to estimate the volume of air rather than using total air volume meters.

- Operational logistics might have temporarily compromised the air sampling in some areas, such as pump failures, leaking filters, filters destroyed by harsh chemicals, or filters obviously contaminated by a source other than the air.

These problems arguably could have enabled intakes exceeding the air concentration limits for some periods. Because of the sensitivity to these uncertainties of the method used in this TIB to determine dose, **the dose uncertainty distribution is lognormal with the geometric mean as determined by the equations in this document with a geometric standard deviation of 3.**

Remember that exposure to tritium, radioiodines, radon, or ^{14}C has to be accounted for separately from these intake/dose estimations.

7.0 WHEN TO USE THIS TIB

Dose reconstructors should use this TIB for workers who were not on a bioassay program or were monitored but had no indication of intakes in the bioassay data, except as discussed below and subject to the limitations in Section 3.0.

Some bioassay methods were capable of detecting small intakes, well below the intakes determined using this TIB. Whole-body counting for gamma emitters with high gamma abundance and energies greater than about 200 keV is a good example. Use of this TIB is still an overestimate even if there are bioassay indications of intakes (i.e., bioassay results exceeding the decision level, minimum detection level, or recording level, as appropriate), provided the results meet the conditions listed in Table 7-1 (whole body counts) or Table 7-2 (urinalyses). The values in Tables 7-1 and 7-2 are actually a factor of 10 less than that which would equal the dose produced by the assumed intake to account for uncertainties in counting, the possibility of several radionuclides in a single count, undetected radionuclides, and the different impacts between inhaled or ingested material.

Urinalysis results from the intakes listed in this TIB can be predicted using the current ICRP models. If an individual has urinalysis results that are less than 0.1 times that predicted by the intakes of this TIB using the most favorable absorption type, the TIB intake can be considered an overestimate.

Table 7-1 Whole-body count (WBC) results below which this TIB method is still considered an overestimate.

Radionuclide	Absorption type	Time since last WBC (d)[a]	Organ of concern	WBC result must be less than (nCi):
Cs-137	F	90	Adrenals	34
			Urinary bladder	30
			Brain	28
			Breast	27
			Gall bladder	58
			Heart wall	31
			Kidneys	31
			Liver	150
			Muscle	30
			Ovaries	32
			Pancreas	32
			Testes	29
			Thyroid	30
			Red bone marrow	115
			Bone surface	220
			Stomach	30
			SI	31
			ULI	52
			LLI	130
			Skin	26
			Spleen	31
			Thymus	30
			Uterus	32
			ET	69
			Lung	640
			Colon	87
			Esophagus	30
Cs-137	F	180	Adrenals	39
			Urinary bladder	34
			Brain	32
			Breast	30
			Gall bladder	66
			Heart wall	35
			Kidneys	35
			Liver	170
			Muscle	33
			Ovaries	36
			Pancreas	36
			Testes	33
			Thyroid	34
			Red bone marrow	130
			Bone surface	250
			Stomach	34
			SI	36
			ULI	58
			LLI	140
			Skin	29
			Spleen	35
			Thymus	34
			Uterus	36
			ET	78
			Lung	720
			Colon	98
			Esophagus	34

Table 7-1 (Continued). Whole-body count (WBC) results below which this TIB method is still considered an overestimate.

Radionuclide	Absorption type	Time since last WBC (d)[a]	Organ of concern	WBC result must be less than (nCi):
Cs-137	F	365	Adrenals	24
			Urinary bladder	21
			Brain	19
			Breast	19
			Gall bladder	16
			Heart wall	22
			Kidneys	21
			Liver	106
			Muscle	21
			Ovaries	22
			Pancreas	22
			Testes	20
			Thyroid	21
			Red bone marrow	80
			Bone surface	150
			Stomach	21
			SI	22
			ULI	36
			LLI	89
			Skin	18
			Spleen	21
			Thymus	21
			Uterus	22
			ET	48
			Lung	440
			Colon	60
			Esophagus	21
Cs-137	F	730	Adrenals	2
			Urinary bladder	2
			Brain	2
			Breast	2
			Gall bladder	4
			Heart wall	2
			Kidneys	2
			Liver	10
			Muscle	2
			Ovaries	2
			Pancreas	2
			Testes	2
			Thyroid	2
			Red bone marrow	8
			Bone surface	15
			Stomach	2
			SI	2
			ULI	4
			LLI	9
			Skin	2
			Spleen	2
			Thymus	2
			Uterus	2
			ET	5
			Lung	43
			Colon	6
			Esophagus	2

Table 7-1 (Continued). Whole-body count (WBC) results below which this TIB method is still considered an overestimate.

Radionuclide	Absorption type	Time since last WBC (d)[a]	Organ of concern	WBC result must be less than (nCi):
Co-60	M or S	90	Adrenals	10
			Urinary bladder	21
			Brain	25
			Breast	5
			Gall bladder	23
			Heart wall	4
			Kidneys	15
			Liver	36
			Muscle	13
			Ovaries	14
			Pancreas	11
			Testes	28
			Thyroid	15
			Red bone marrow	43
			Bone surface	108
			Stomach	11
			SI	13
			ULI	15
			LLI	25
			Skin	16
			Spleen	9
			Thymus	6
			Uterus	20
			ET	4
			Lung	11
			Colon	20
			Esophagus	6
Co-60	M or S	180	Adrenals	12
			Urinary bladder	27
			Brain	31
			Breast	8
			Gall bladder	30
			Heart wall	7
			Kidneys	19
			Liver	45
			Muscle	17
			Ovaries	18
			Pancreas	14
			Testes	35
			Thyroid	19
			Red bone marrow	51
			Bone surface	135
			Stomach	14
			SI	17
			ULI	18
			LLI	31
			Skin	29
			Spleen	14
			Thymus	9
			Uterus	25
			ET	6
			Lung	19
			Colon	25
			Esophagus	9

Table 7-1 (Continued). Whole-body count (WBC) results below which this TIB method is still considered an overestimate.

Radionuclide	Absorption type	Time since last WBC (d)[a]	Organ of concern	WBC result must be less than (nCi):
Co-60	M or S	360	Adrenals	13
			Urinary bladder	28
			Brain	32
			Breast	9
			Gall bladder	30
			Heart wall	8
			Kidneys	19
			Liver	46
			Muscle	17
			Ovaries	18
			Pancreas	15
			Testes	36
			Thyroid	19
			Red bone marrow	60
			Bone surface	139
			Stomach	15
			SI	17
			ULI	19
			LLI	32
			Skin	21
			Spleen	14
			Thymus	10
			Uterus	26
			ET	5
			Lung	20
			Colon	26
			Esophagus	10
Co-60	M or S	730	Adrenals	6
			Urinary bladder	13
			Brain	14
			Breast	4
			Gall bladder	14
			Heart wall	4
			Kidneys	9
			Liver	21
			Muscle	8
			Ovaries	8
			Pancreas	7
			Testes	17
			Thyroid	570
			Red bone marrow	27
			Bone surface	63
			Stomach	7
			SI	8
			ULI	9
			LLI	15
			Skin	10
			Spleen	7
			Thymus	4
			Uterus	12
			ET	2
			Lung	9
			Colon	12
			Esophagus	4

Table 7-1(Continued). Whole-body count (WBC) results below which this TIB method is still considered an overestimate.

Radionuclide	Absorption type	Time since last WBC (d)[a]	Organ of concern	WBC result must be less than (nCi):
Mn-54	F or M	90	Adrenals	24
			Urinary bladder	49
			Brain	42
			Breast	23
			Gall bladder	35
			Heart wall	20
			Kidneys	29
			Liver	47
			Muscle	40
			Ovaries	31
			Pancreas	30
			Testes	65
			Thyroid	51
			Red bone marrow	70
			Bone surface	113
			Stomach	39
			SI	42
			ULI	43
			LLI	100
			Skin	47
			Spleen	39
			Thymus	26
			Uterus	43
			ET	15
			Lung	150
			Colon	71
			Esophagus	26
Mn-54	F or M	180	Adrenals	8
			Urinary bladder	17
			Brain	14
			Breast	16
			Gall bladder	12
			Heart wall	13
			Kidneys	10
			Liver	16
			Muscle	14
			Ovaries	11
			Pancreas	10
			Testes	22
			Thyroid	18
			Red bone marrow	24
			Bone surface	39
			Stomach	14
			SI	11
			ULI	15
			LLI	35
			Skin	16
			Spleen	16
			Thymus	17
			Uterus	15
			ET	6
			Lung	117
			Colon	24
			Esophagus	17

Table 7-1 (Continued). Whole-body count (WBC) results below which this TIB method is still considered an overestimate.

Radionuclide	Absorption type	Time since last WBC (d)[a]	Organ of concern	WBC result must be less than (nCi):
Mn-54	F or M	365	Adrenals	0.5
			Urinary bladder	0.9
			Brain	0.8
			Breast	0.9
			Gall bladder	0.6
			Heart wall	0.7
			Kidneys	0.5
			Liver	0.9
			Muscle	0.7
			Ovaries	0.6
			Pancreas	0.6
			Testes	1.2
			Thyroid	1.0
			Red bone marrow	1.3
			Bone surface	2.1
			Stomach	0.7
			SI	0.6
			ULI	0.8
			LLI	1.9
			Skin	0.9
			Spleen	0.9
			Thymus	0.9
			Uterus	0.8
			ET	0.3
			Lung	14
			Colon	1.3
			Esophagus	0.9
Mn-54	F or M	730	Not applicable for all organs	
Nb-95	M or S	90	Adrenals	18
			Urinary bladder	34
			Brain	50
			Breast	13
			Gall bladder	35
			Heart wall	11
			Kidneys	16
			Liver	53
			Muscle	24
			Ovaries	14
			Pancreas	21
			Testes	65
			Thyroid	33
			Red bone marrow	57
			Bone surface	51
			Stomach	18
			SI	13
			ULI	11
			LLI	15
			Skin	34
			Spleen	22
			Thymus	14
			Uterus	27
			ET	3
			Lung	12
			Colon	13

Table 7-1 (Continued). Whole-body count (WBC) results below which this TIB method is still considered an overestimate.

Radionuclide	Absorption type	Time since last WBC (d)[a]	Organ of concern	WBC result must be less than (nCi):
Nb-95	M or S	90	Esophagus	14
Nb-95	M or S	180	Adrenals	4
			Urinary bladder	8
			Brain	12
			Breast	4
			Gall bladder	8
			Heart wall	3
			Kidneys	4
			Liver	13
			Muscle	6
			Ovaries	3
			Pancreas	5
			Testes	16
			Thyroid	8
			Red bone marrow	14
			Bone surface	12
			Stomach	4
			SI	3
			ULI	3
			LLI	4
			Skin	8
			Spleen	5
			Thymus	4
			Uterus	7
			ET	1
			Lung	3
			Colon	3
			Esophagus	4
Nb-95	M or S	365	Adrenals	0.1
			Urinary bladder	0.2
			Brain	0.3
			Breast	0.1
			Gall bladder	0.2
			Heart wall	0.1
			Kidneys	0.1
			Liver	0.3
			Muscle	0.2
			Ovaries	0.1
			Pancreas	0.1
			Testes	0.4
			Thyroid	0.2
			Red bone marrow	0.4
			Bone surface	0.3
			Stomach	0.1
			SI	0.1
			ULI	0.1
			LLI	0.1
			Skin	0.2
			Spleen	0.1
			Thymus	0.1
			Uterus	0.2
			ET	0.0
			Lung	0.1
			Colon	0.1
			Esophagus	0.1

Table 7-1 (Continued). Whole-body count (WBC) results below which this TIB method is still considered an overestimate.

Radionuclide	Absorption type	Time since last WBC (d)[a]	Organ of concern	WBC result must be less than (nCi):
Nb-95	M or S	730	Not applicable for all organs	
Ce-141	M or S	90	Adrenals	130
			Urinary bladder	240
			Brain	270
			Breast	106
			Gall bladder	180
			Heart wall	79
			Kidneys	170
			Liver	29
			Muscle	180
			Ovaries	107
			Pancreas	140
			Testes	320
			Thyroid	240
			Red bone marrow	104
			Bone surface	20
			Stomach	37
			SI	18
			ULI	6
			LLI	6
			Skin	200
			Spleen	180
			Thymus	108
			Uterus	200
			ET	3
			Lung	4
			Colon	6
			Esophagus	108
Ce-141	M or S	180	Adrenals	36
			Urinary bladder	65
			Brain	74
			Breast	27
			Gall bladder	50
			Heart wall	20
			Kidneys	46
			Liver	8
			Muscle	48
			Ovaries	27
			Pancreas	39
			Testes	89
			Thyroid	67
			Red bone marrow	29
			Bone surface	6
			Stomach	9
			SI	5
			ULI	1.4
			LLI	1.5
			Skin	57
			Spleen	45
			Thymus	27
			Uterus	57
			ET	0.8
			Lung	0.9
			Colon	1.4

Table 7-1 (Continued). Whole-body count (WBC) results below which this TIB method is still considered an overestimate.

Radionuclide	Absorption type	Time since last WBC (d)[a]	Organ of concern	WBC result must be less than (nCi):
Ce-141	M or S	180	Esophagus	27
Ce-141	M or S	365	Adrenals	1.3
			Urinary bladder	2.4
			Brain	2.7
			Breast	0.9
			Gall bladder	1.8
			Heart wall	0.7
			Kidneys	1.7
			Liver	0.3
			Muscle	1.7
			Ovaries	0.9
			Pancreas	1.4
			Testes	3.3
			Thyroid	2.4
			Red bone marrow	1.1
			Bone surface	0.2
			Stomach	0.3
			SI	0.2
			ULI	<0.1
			LLI	<0.1
			Skin	2.1
			Spleen	1.5
			Thymus	0.9
			Uterus	2.0
			ET	<010
			Lung	<0.1
			Colon	<0.1
			Esophagus	0.9
Ce-144	M or S	90	Adrenals	56
			Urinary bladder	59
			Brain	46
			Breast	40
			Gall bladder	89
			Heart wall	52
			Kidneys	53
			Liver	3
			Muscle	49
			Ovaries	60
			Pancreas	56
			Testes	52
			Thyroid	54
			Red bone marrow	13
			Bone surface	15
			Stomach	30
			SI	15
			ULI	4
			LLI	4
			Skin	38
			Spleen	55
			Thymus	52
			Uterus	63
			ET	5
			Lung	3
			Colon	4
			Esophagus	52

Table 7-1 (Continued). Whole-body count (WBC) results below which this TIB method is still considered an overestimate.

Radionuclide	Absorption type	Time since last WBC (d)[a]	Organ of concern	WBC result must be less than (nCi):
Ce-144	M or S	180	Adrenals	86
			Urinary bladder	90
			Brain	69
			Breast	60
			Gall bladder	136
			Heart wall	79
			Kidneys	80
			Liver	5
			Muscle	75
			Ovaries	91
			Pancreas	86
			Testes	78
			Thyroid	82
			Red bone marrow	20
			Bone surface	22
			Stomach	46
			SI	20
			ULI	6
			LLI	5
			Skin	58
			Spleen	83
			Thymus	78
			Uterus	95
			ET	6
			Lung	4
			Colon	5
			Esophagus	78
Ce-144	M or S	365	Adrenals	103
			Urinary bladder	108
			Brain	84
			Breast	72
			Gall bladder	160
			Heart wall	96
			Kidneys	97
			Liver	6
			Muscle	90
			Ovaries	110
			Pancreas	103
			Testes	95
			Thyroid	99
			Red bone marrow	24
			Bone surface	27
			Stomach	53
			SI	22
			ULI	6
			LLI	6
			Skin	70
			Spleen	100
			Thymus	95
			Uterus	110
			ET	7
			Lung	4
			Colon	6
			Esophagus	95

a. ±15%. The choice of days is meant to represent general measurement frequencies, such as quarterly, semiannually, or annually. It is recognized that workers did not get whole-body counts at intervals that were exact to the day.

Table 7-2 Urinary excretion results below which this TIB method is still considered an overestimate.

Radionuclide or analysis type	Nature of exposure	Time since last urinalysis (d)	Organ of concern	Excretion rate	
				pCi/d	pCi/L
Radiostrontium or Sr-90	Type F	90	Adrenals	270	193
			Urinary bladder	95	68
			Brain	190	136
			Breast	174	124
			Gall bladder	401	286
			Heart wall	250	179
			Kidneys	240	171
			Liver	916	654
			Muscle	214	153
			Ovaries	265	189
			Pancreas	266	190
			Testes	213	152
			Thyroid	228	163
			Red bone marrow	9	6
			Bone surface	12	9
			Stomach	210	150
			SI	219	156
			ULI	94	67
			LLI	85	61
			Skin	160	114
			Spleen	240	171
			Thymus	230	164
			Uterus	268	191
			ET	141	101
			Lung	3,200	2286
			Colon	88	63
			Esophagus	230	164
		180	Adrenals	540	386
			Urinary bladder	190	136
			Brain	380	271
			Breast	348	249
			Gall bladder	802	573
			Heart wall	500	357
			Kidneys	480	343
			Liver	1,832	1309
			Muscle	427	305
			Ovaries	529	378
			Pancreas	532	380
			Testes	426	304
			Thyroid	457	326
			Red bone marrow	17	12
			Bone surface	23	16
			Stomach	420	300
			SI	439	314
			ULI	189	135
			LLI	170	121
			Skin	319	228
			Spleen	479	342
Radiostrontium or Sr-90 cont.	Type F	180	Thymus	459	328
			Uterus	537	384
			ET	282	201
			Lung	6400	4571
			Colon	176	126
			Esophagus	459	328
		365	Adrenals	204	146
			Urinary bladder	72	51
			Brain	144	103
			Breast	132	94
			Gall bladder	304	217
			Heart wall	189	135
			Kidneys	182	130
			Liver	694	496

Table 7-2 (Continued). Urinary excretion results below which this TIB method is still considered an overestimate.

Radionuclide or analysis type	Nature of exposure	Time since last urinalysis (d)	Organ of concern	Excretion rate	
				pCi/d	pCi/L
Radiostrontium or Sr-90 cont.	Type F	365	Muscle	162	116
			Ovaries	200	143
			Pancreas	202	144
			Testes	161	115
			Thyroid	173	124
			Red bone marrow	7	5
			Bone surface	9	6
			Stomach	159	114
			SI	166	119
			ULI	71	51
			LLI	65	46
			Skin	121	86
			Spleen	182	130
			Thymus	174	124
			Uterus	203	145
			ET	107	76
			Lung	2,424	1731
			Colon	66	47
			Esophagus	174	124
		730	Adrenals	79	56
			Urinary bladder	28	20
			Brain	55	39
			Breast	51	36
			Gall bladder	117	84
			Heart wall	73	52
			Kidneys	70	50
			Liver	267	191
			Muscle	62	44
			Ovaries	77	55
			Pancreas	78	56
			Testes	62	44
			Thyroid	67	48
			Red bone marrow	3	2
			Bone surface	3	2
Radiostrontium or Sr-90 cont.	Type F	730	Stomach	61	44
			SI	64	46
			ULI	27	19
			LLI	25	18
			Skin	47	34
			Spleen	70	50
			Thymus	67	48
			Uterus	78	56
			ET	41	29
			Lung	932	666
			Colon	26	19
			Esophagus	67	48
				µg/d	**µg/L**
Elemental U	U facility, natural U, type F	NA, assumes chronic exposure for at least one year. Sampling during work days.	Adrenals	1,300	927
			Urinary bladder	1,200	863
			Brain	1,290	920
[Note change in units]			Breast	1,290	923
			Gall bladder	1,290	922
			Heart wall	1,300	927
			Kidneys	240	172
			Liver	3,050	2180
			Muscle	1,290	922
			Ovaries	5,320	3800
			Pancreas	1,290	924
			Testes	5,410	3870
			Thyroid	1,290	921

Table 7-2 (Continued). Urinary excretion results below which this TIB method is still considered an overestimate.

Radionuclide or analysis type	Nature of exposure	Time since last urinalysis (d)	Organ of concern	Excretion rate	
				pCi/d	pCi/L
Elemental U cont.	U facility, natural U, type F	NA, assumes chronic exposure for at least one year. Sampling during work days.	Red bone marrow	5,390	3850
			Bone surface	6,910	4940
			Stomach	1,280	918
			SI	1,280	911
			ULI	1,240	887
			LLI	1,320	942
			Skin	1,290	920
			Spleen	1,290	923
			Thymus	1,290	924
			Uterus	1,290	920
			ET	535,000	382000
			Lung	827,000	591000
			Colon	1,300	926
			Esophagus	1,290	924
	LEU			Multiply results for natural U by 0.031	
	HEU			Multiply results for natural U by 0.01	
Elemental U	U facility, natural U, type F	Sampling on third day of no exposure, e.g., Monday morning	Adrenals	376	269
			Urinary bladder	351	250
			Brain	374	267
			Breast	375	268
			Gall bladder	374	267
Elemental U cont.	U facility, natural U, type F	Sampling on third day of no exposure	Heart wall	376	269
			Kidneys	70	50
			Liver	885	632
			Muscle	374	267
			Ovaries	1,540	1100
			Pancreas	375	268
			Testes	1,570	1120
			Thyroid	374	267
			Red bone marrow	1,560	1120
			Bone surface	2,000	1430
			Stomach	373	266
			SI	370	264
			ULI	360	257
			LLI	382	273
			Skin	373	267
			Spleen	375	268
			Thymus	375	268
			Uterus	374	267
			ET	155,000	111000
			Lung	240,000	171000
			Colon	376	269
			Esophagus	375	268
	LEU			Multiply results for natural U by 0.31	
	HEU			Multiply results for natural U by 0.01	
Elemental U	Alpha facility, any absorption type	Because intakes at alpha facilities are considered acute for overestimates, any detection of U above natural background will have to be evaluated separately from OTIB-018.			

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ATTACHMENT A
AIR CONCENTRATION LIMITS PRIOR TO 1953 OR OTHER LIMITS

Air concentration limits established for a site prior to 1953 can be used to extend the applicability of this TIB to those earlier years.

Because Hanford Site tolerance levels were used to control intakes, they fit the criteria of air concentration limit. The alpha and beta particular air concentration limit at Hanford from at least 1949 to 1952 were 1×10^{-12} μCi/mL for alpha in nonuranium facilities, 1×10^{-9} μCi/mL for beta, and 1.5×10^{-4} μg U/mL (which converts to 1.1×10^{10} μCi/mL) for uranium facilities (Patterson 1949). These concentrations are less than the concentrations listed in Table 4-1; thus, the use of Table 4-1 values is a plausible overestimate. Therefore, this TIB approach can be used unmodified for Hanford claims from 1949 to the present. Higher tolerance levels of 1×10^{-8} μCi/mL for beta-emitters and 4×10^{-11} μCi/mL for plutonium were in place at least by October 1945 (Cantril 1945).

Oak Ridge National Laboratory (ORNL) used the concept of tolerance levels for air concentrations prior to 1953. By July 1944, ORNL had established an air concentration limit of 5×10^{-10} μg/mL (3×10^{-11} μCi/mL) for plutonium based on alpha counting (Parker 1944a). A limit for beta-emitters in air (based on tolerance in the thyroid for ^{131}I even though they might not have been collecting iodine properly on their filters) was established at least by 1944 at 1×10^{-7} μCi/mL (Parker 1944b).

Los Alamos also established tolerance levels for air concentrations for specific buildings as early as 1948. The tolerance levels varied by building, apparently based on the predominant radionuclide used in the building. For beta-emitters the limit was stated as 1×10^{-7} μCi/ml in one reference (LANL 1948). The air concentration limit for alpha-emitters other than uranium or ^{210}Po was 70 dpm/m^3 (3.2×10^{-11} μCi/ml) for 1948 through 1952 (LANL 1947, 1950, 1952). This was actually the highest limit as many buildings had lower limits. For uranium buildings the highest air concentration was 400 dpm/m^3 (1.8×10^{-10} μCi/ml) reported in both late 1947 and 1950 so assumed to apply throughout the period 1948-1950 (LANL 1947, 1950). For ^{210}Po, which was a source of exposure at Los Alamos only for 1948-53 (ORAU 2004a), the air concentration limit was the same throughout the period at 1400 dpm/m^3 (6.3×10^{-10} μCi/ml) (LANL 1947,1950, 1954). The concentration limits for alpha-emitters and beta-emitters changed in 1953. The highest air concentration limit for a plutonium facility was 9 dpm/m^3 (4×10^{-12} μCi/ml) (LANL 1954) which continued to show in reports in 1961 and 1963 (LANL 1969). Fission products were not in heavy use at Los Alamos; the only limiting air concentration for beta-emitters found after 1953 (actually referred to as fission products) was listed at 6700 dpm/m^3 (3×10^{-9} μCi/ml) in a monthly report in 1961(LANL 1969). It was assumed that this limit applied from 1953 to 1966. The limits for uranium in reports during and after 1951 (LANL 1952, 1969) were less than the NCRP value in Table 4-2 so the NCRP value was assumed to apply, in other words not adjustment from the default value in Table 4-2 was needed.

The Rocky Flats site used two air concentration limits for plutonium operations throughout its operational period: 2×10^{-12} μCi/ml was used for plutonium metal operations and 4×10^{-12} μCi/ml was used for plutonium chemistry operations. These values were used through 1994 (ORAU 2004b).

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ATTACHMENT B
VALIDATION OF THE BASIC ASSUMPTION IN THIS TIB

The basis for this TIB is that weapons sites with a radiological control program controlled exposures to airborne radioactive contaminates to less than an applicable limit. The limit varied with time and isotope of concern. This TIB acknowledges that exposures to concentrations greater than these limits did occur. However, the assumption is that, when integrated over time, the average exposure to any individual was less than that which would occur if someone were exposed to the limit continuously. The assumption is tested in the following analysis. Note that this analysis does not change the assumption but merely tests the validity of it.

B.1 URANIUM

Uranium urinalysis data from the Y-12 site was used to test the validity of the assumption. The Y-12 site processed uranium extensively. The data consisted of hundreds of urine samples each month from 1952 through 1985. Because this TIB provides mean values that are to be used in a lognormal distribution, the median result was used in the comparison. Figure B-1 plots these median sample results as well as the uranium result predicted by this TIB for an inhalation of type M uranium.

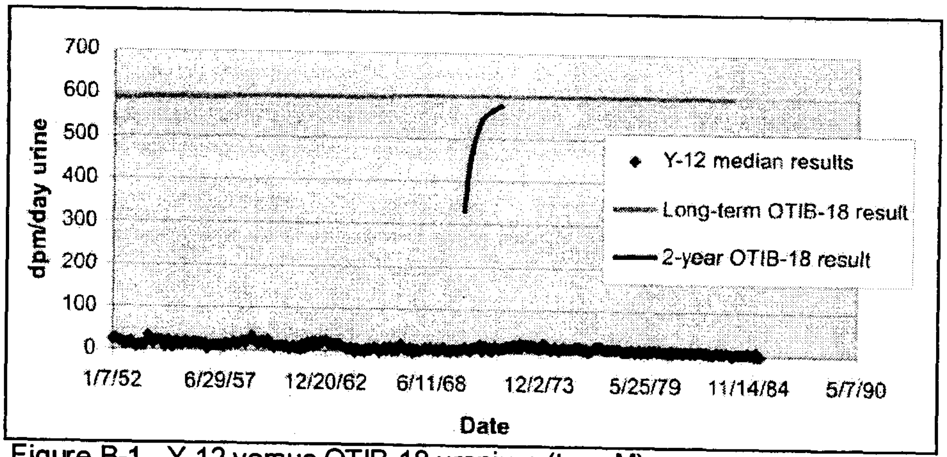

Figure B-1. Y-12 versus OTIB-18 uranium (type M).

The TIB result is considerably higher than the actual Y-12 median urine sample result. The 2-year result begins 1 month after the start of the chronic intake and is approximately 10 times higher than the highest actual median result. This verifies that for Y-12 uranium intakes, either the assumed intake is an overestimate or the lung absorption of the uranium is considerably slower than type M.

Figure B-2 shows the same information using an absorption type S assumption for the TIB values. This graph indicates that for a long-term exposure, it can be shown that this TIB overestimates actual results even if absorption type S is assumed. The short-term exposure, on the other hand, indicates that it could take as long as 3 years to reach the level of the highest median urinalysis result. This implies that this TIB approach might not be an overestimate for type S exposures of duration that are less than 3 years.

B.2 PLUTONIUM

The Rocky Flats Plant worked extensively with plutonium. Plutonium urine samples from Rocky Flats were evaluated from 1953 through 1969. For every year, the median sample was recorded as zero.

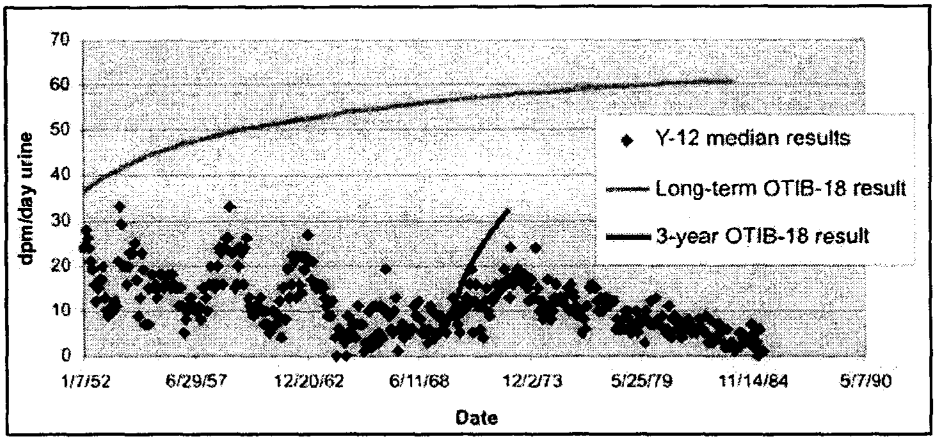

Figure B-2. Y-12 versus OTIB-18 uranium (type S).

The lowest recorded positive value varied through the years from 0.02 dpm/day (0.009 pCi/day) to 0.09 dpm/day (0.04 pCi/day) with most years at 0.02 dpm/day. Figure B-3 shows the assumed intake in this TIB for both a type S and a type M inhalation. The figure also shows the 0.009-pCi/day and the 0.04-pCi/day values. The figure indicates that the TIB values would be likely to exceed the measured data soon after the start of a chronic type M inhalation. The type S inhalation, however, might not result in detectable urine samples.

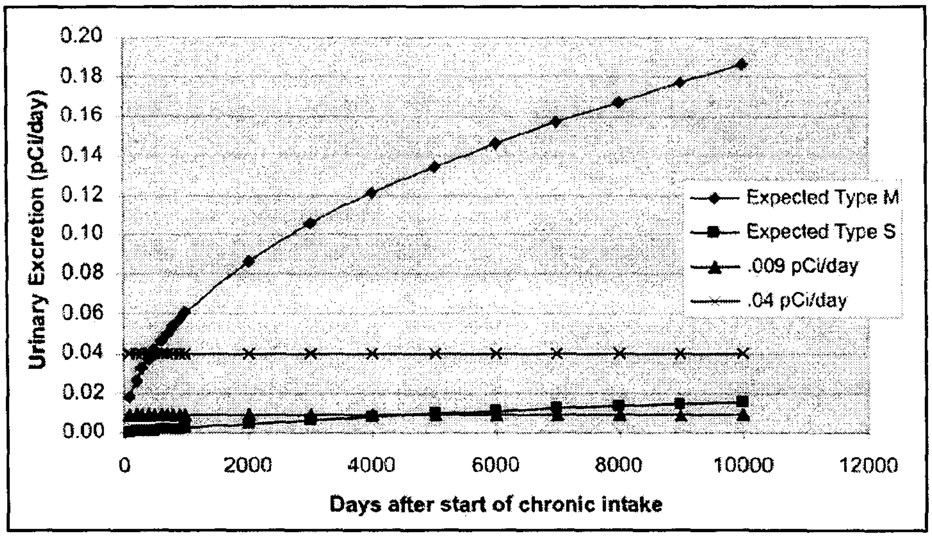

Figure B-3. Rocky Flats Pu expected urine versus actual results.

The Hanford site produced much of the plutonium used by the weapons program. It appears to be reasonable, therefore, to review Hanford site plutonium urine results against the assumptions in this TIB.

The TIB analysis evaluated plutonium urine samples from Hanford from 1953 through 1969. The median sample for each year varied. The median result was never more than 0.023 pCi/day and for most years was less than that. Figure B-4 shows the assumed intake in this TIB for both a type S and a type M inhalation. The figure also shows the 0.023-pCi/day value. As with the Rocky Flats data, the graph indicates that the TIB values would be likely to exceed the measured data soon after the start of a chronic type M inhalation but a type S inhalation might not result in a detectable urine sample.

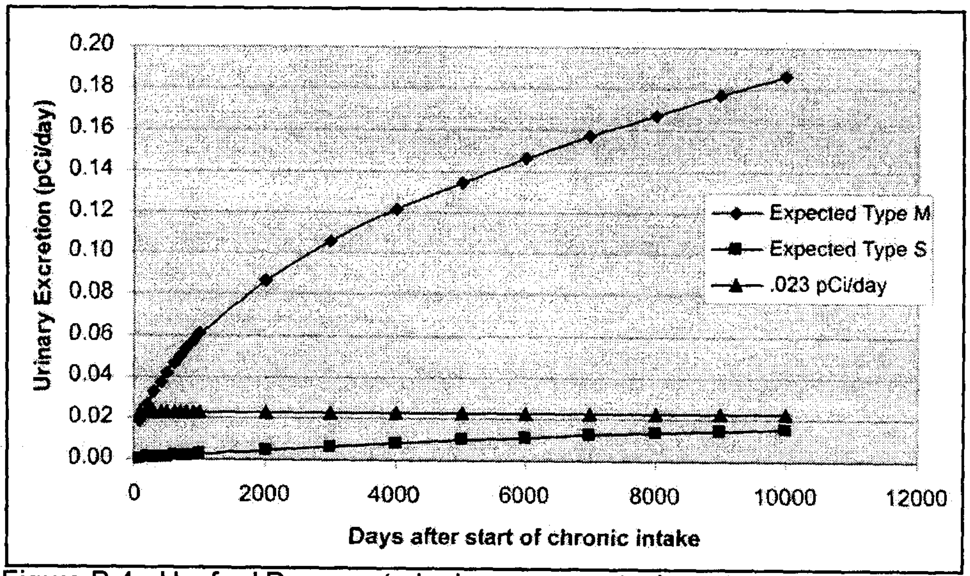

Figure B-4. Hanford Pu expected urine versus actual results.

B.3 FISSION PRODUCTS

The fission product that was primarily sampled in urine at Hanford was ^{90}Sr. In the 1960s, the recording limit at Hanford for ^{90}Sr urine samples appears to be 16.7 pCi/day. More than 95% of the samples in 1966, 1967, and 1968 were recorded as less than this limit. After that period, the recording limit decreased and enabled the evaluation of the median sample result.

Figure B-5 shows the median and 95th-percentile urine result predicted from the values used in this TIB for a chronic inhalation of type F ^{90}Sr. The 95th-percentile results were determined using the assumption in this TIB that the intakes are lognormally distributed with a geometric standard deviation of three. The figure also shows the recording limit of 16.7 pCi/day as well as the median sample results for 1970 through 1974.

Figure B-5 indicates that the results predicted from the 95th percentile of the intake in this TIB exceed the reporting limit from the 1960s. More than 95% of the values were reported as less than this value. In addition, it shows that when the reporting limit was reduced, the median result was less than the mean result predicted from the intake in this TIB.

B.4 CONCLUSION

The analysis discussed above indicates that the assumption made in this TIB is valid if the material to which an individual is exposed has type F or type M absorption characteristics. Because these are the limiting absorption types for systemic organs, it appears the assumption is valid for systemic organs.

Type S, however, is the limiting absorption type (when comparing the same intake values for different absorption types) for the respiratory tract. Due to the detection limits of urinalysis, it cannot be shown

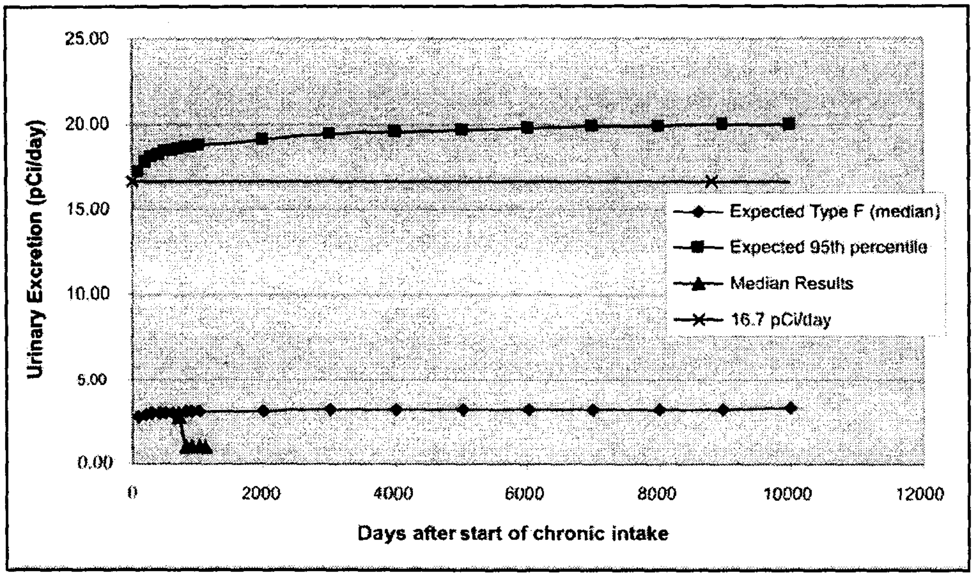

Figure B-5. Hanford Sr-90 expected urine versus actual results.

that the values of this TIB are an overestimate for the respiratory tract. However, there are several important issues to point out.

- While comparing the recording limit to the TIB mean cannot show the TIB to be an overestimate, continuous exposures at these levels would be reportable in many of the long-term workers. Levels below the reporting limit were seen in a large percent of the sampled population, much larger than 50%.

- Long-term exposure to these levels requires that an employee would be working into the modern era. In more modern eras, chest counts, whole-body counts, and more sensitive urinalysis would indicate if the average worker were exposed to these levels for a long duration.

- The data analyzed was obviously for monitored workers. The monitored population is probably the population that radiological control personnel believed to have the highest exposure potential.

Taken together, this implies that while it might not be possible to demonstrate that the average radiological worker was exposed to a lower concentration of type S material, it can be seen that it is very possible. Therefore this TIB should not be considered an overestimate for respiratory tract dose unless other information, such as chest counts or whole-body counts, is available for the individual.

However, the TIB assumes a continuous exposure every working minute of every day. For individuals who were only intermittently exposed, a considerably higher air concentration would have had to be routinely present to equal the same intake. Intakes considerably higher than those listed in this TIB would have been detectable in the urine results of those present more frequently in those areas. This TIB should, therefore, be considered an overestimate for intermittently exposed workers regardless of the organ of concern.

ATTACHMENT C
SITE APPLICABILITY

This TIB is applicable to all DOE sites with the limitations listed in the TIB and exceptions listed below. Table 4 contains three inventory lists of radionuclides. Dose reconstructors should use these lists as follows:

- The alpha list should apply alone to sites or facilities that primarily handled plutonium.
- The uranium list should be applied alone to sites or facilities that primarily handled uranium.
- The alpha plus the beta list should be applied to reactor sites, national laboratories, and any other sites.

Some large sites contained facilities that performed different functions. For example, the Hanford Site is a reactor site but also had fuel fabrication facilities. The fabrication facilities normally controlled air concentrations to uranium limits. Therefore, the uranium list should be applied to individuals working in those facilities.

Some sites operated with particular nuclides in campaigns. Once the campaign ended, the nuclide was no longer routinely handled. For example, the Fernald site produced thorium products during several campaigns but did not routinely handle it between these campaigns. Therefore, it is permissible to remove thorium from the list of possible radionuclides during times when the individual was not exposed to thorium. This can be done in other situations when there is clear evidence that the individual was not exposed to a particular radionuclide.

Exception
Until 1962, the Nevada Test Site performed atmospheric testing of nuclear weapons. These tests had the potential to create high airborne concentrations for relatively short periods. While it is possible that the continuous assumption made in this TIB will overestimate these intermittent exposures, further evaluation is necessary to verify this. Therefore, this TIB is not currently applicable to exposures at the Nevada Test Site prior to 1963.

ORAU TEAM
Dose Reconstruction Project for NIOSH

Oak Ridge Associated Universities | Dade Moeller & Associates | MJW Corporation

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Concurrence:	Signature on File Edward F. Maher, Task 5 Manager	Concurrence Date:	09/13/2007
Concurrence:	Signature on File Kate Kimpan, Project Director	Concurrence Date:	09/27/2007
Approval:	Brant A. Ulsh Signature on File for James W. Neton, Associate Director for Science	Approval Date:	10/01/2007

☐ New ☐ Total Rewrite ☒ Revision ☐ Page Change

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
06/29/2004	00	New technical basis document for the Oak Ridge National Laboratory – Occupational Internal Dose. First approved issue. Initiated by Robert E. Burns, Jr.
05/30/2006	00 PC-1	Page change revision to update required language in the Introduction on page 8 in Section 5.1. Adds Section 5.4 on pages 28 and 29 on using air monitoring data to assign best estimates of internal dose for unmonitored ORNL workers during the 1944-1947 period. Makes reference changes on pages 30 and 32 in the Reference Section. Adds Attachment B, on pages 67, 68, 69, and 70, which gives the technical justification for this protocol. No changes occurred as a result of internal formal review. Incorporates NIOSH formal review comments. No sections were deleted. This revision results in no change to the assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Elizabeth M. Brackett and Thomas R. La Bone. Approval: Signature on File 05/22/2006 John M. Byrne, Document Owner Signature on File 05/22/2006 John M. Byrne, Task 3 Manager Keith McCartney Signature on File for 05/22/2006 Edward F. Maher, Task 5 Manager Signature on File 05/22/2006 Kate Kimpan, Project Director Signature on File 05/30/2006 James W. Neton, Associate Director for Science
10/01/2007	01	Approved Revision 01 initiated to include Attributions and Annotations, Purpose and Scope sections. Revised Section 5.4. Made editorial changes. No further changes occurred as a result of formal internal review. Incorporates formal NIOSH review comments. Training required: As determined by the Task Manager. Initiated by Robert E. Burns, Jr.

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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

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

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
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
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 Notes on Measurements of Alpha Emitters

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
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
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
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

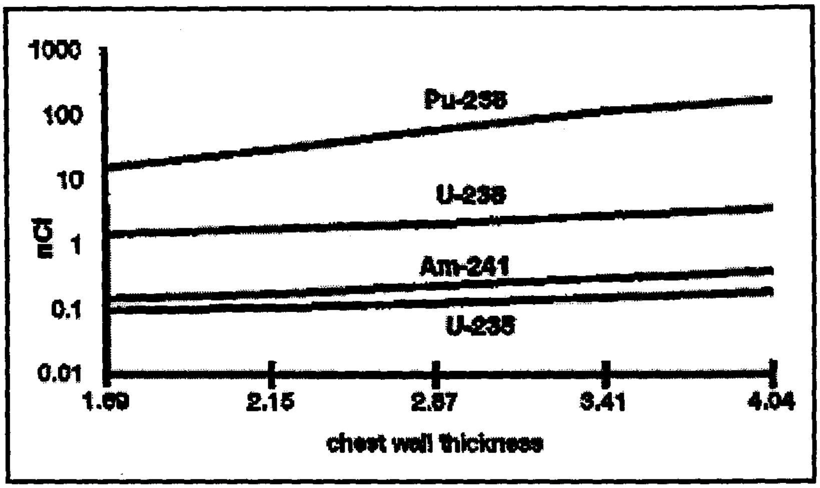
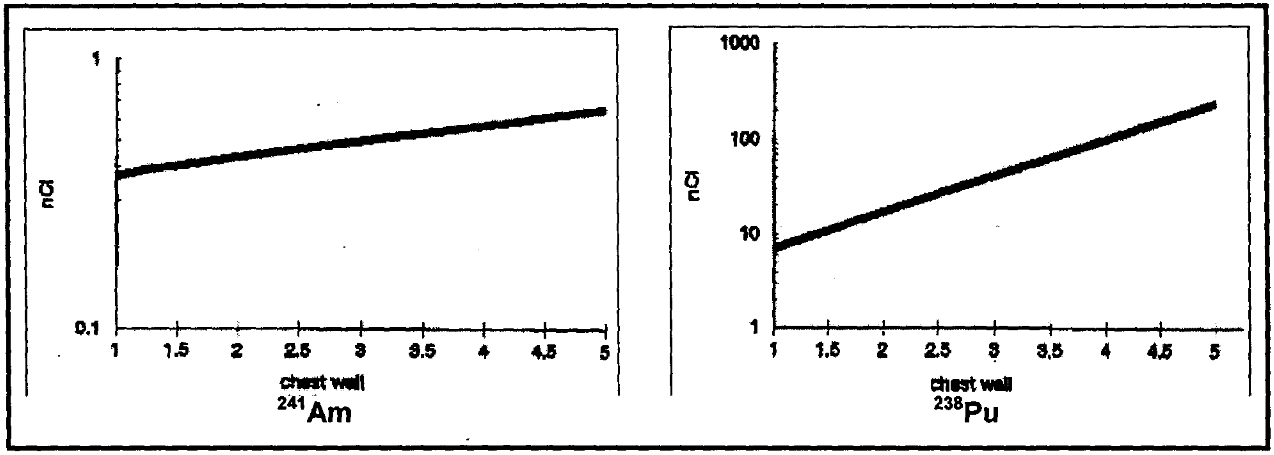


Figure 5-1. Current MDA values (nCi) for ORNL germanium lung counter versus chest-wall thickness (cm) (Watts et al. 1995).



^{241}Am

^{238}Pu

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]

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[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.)

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- 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):

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

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)

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

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.

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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																			

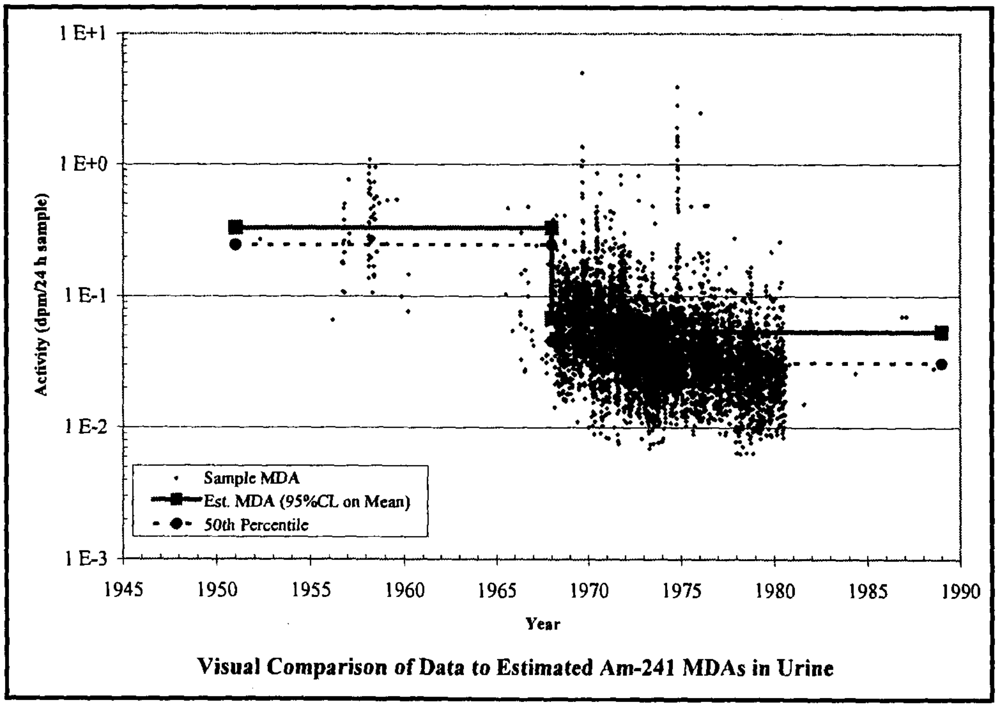

Figure A-1. MDA data for americium-241 in urine (part 1 of 3).

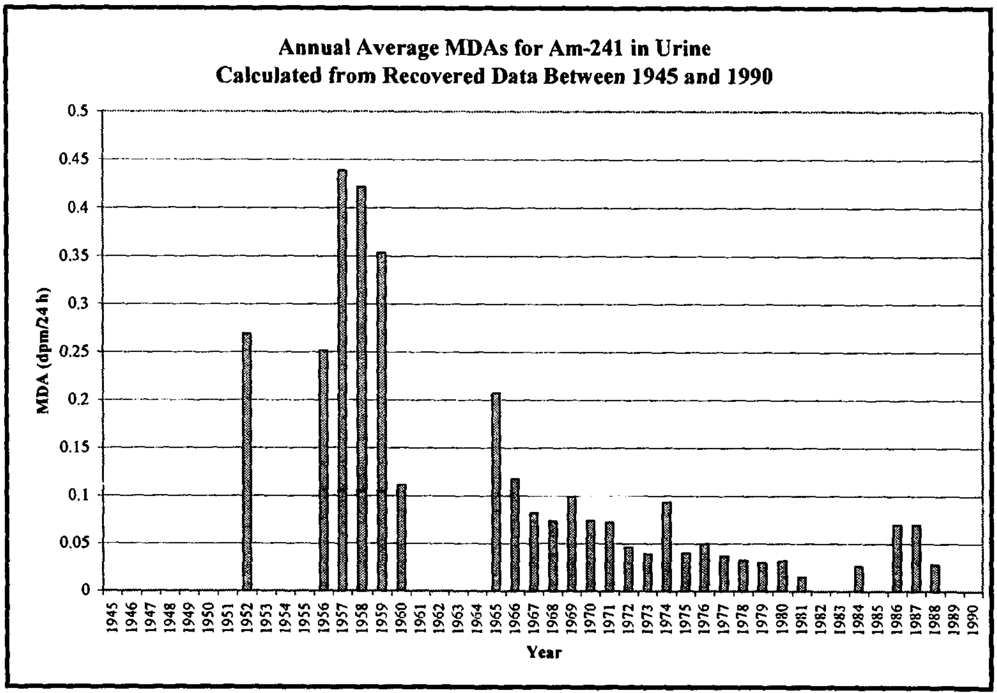

Figure A-1. MDA data for americium-241 in urine (part 2 of 3).

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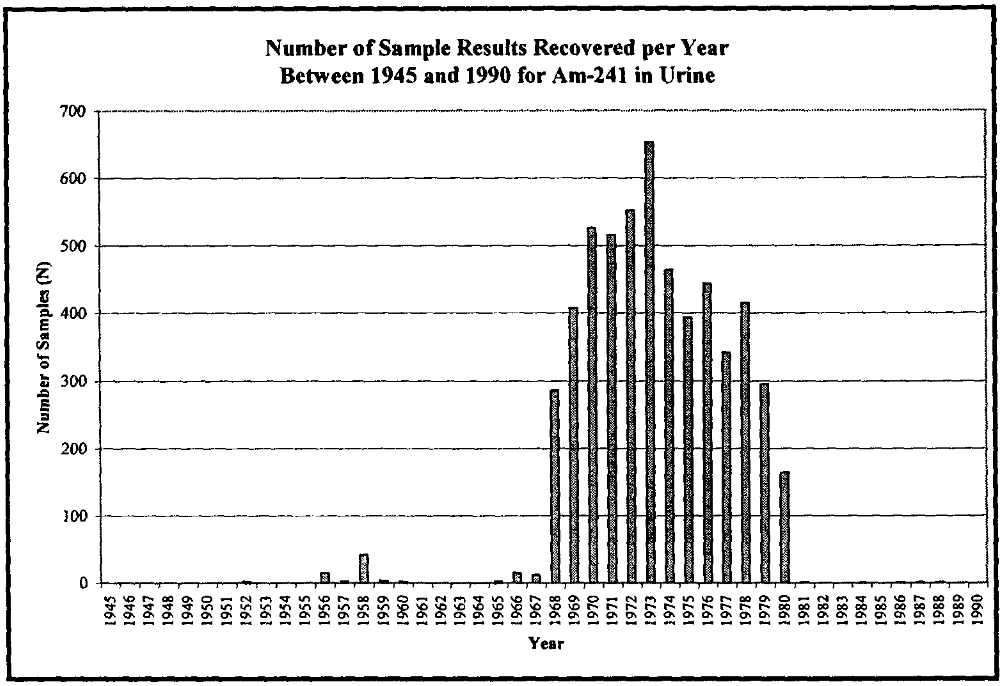

Figure A-1. MDA data for americium-241 in urine (part 3 of 3).

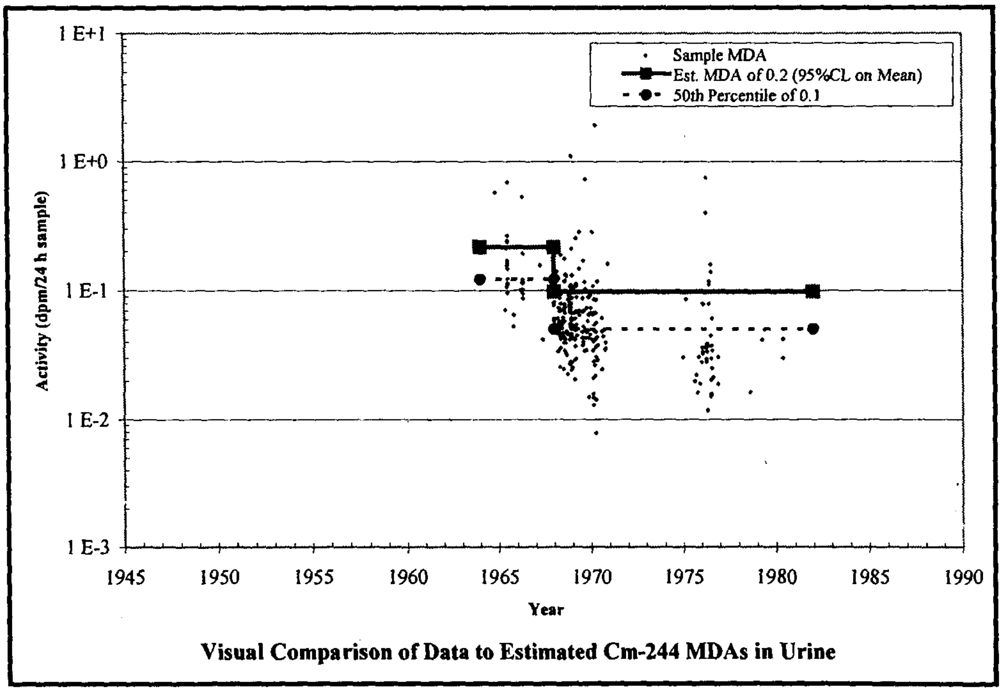

Figure A-2. MDA data for curium-244 in urine (part 1 of 3) (some later tables are "in feces").

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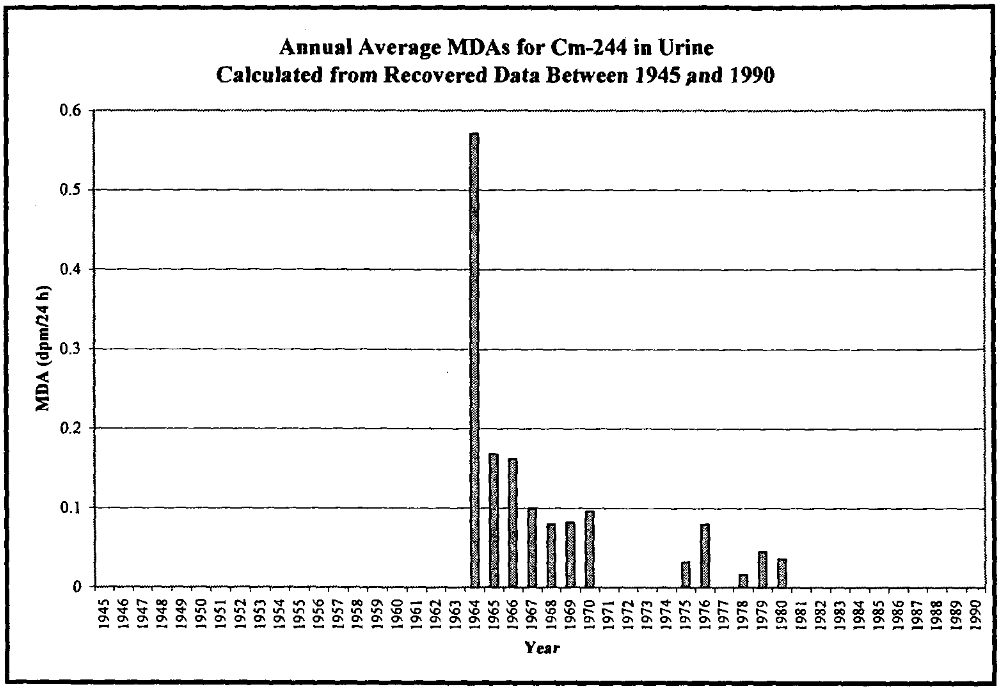

Figure A-2. MDA data for curium-244 in urine (part 2 of 3).

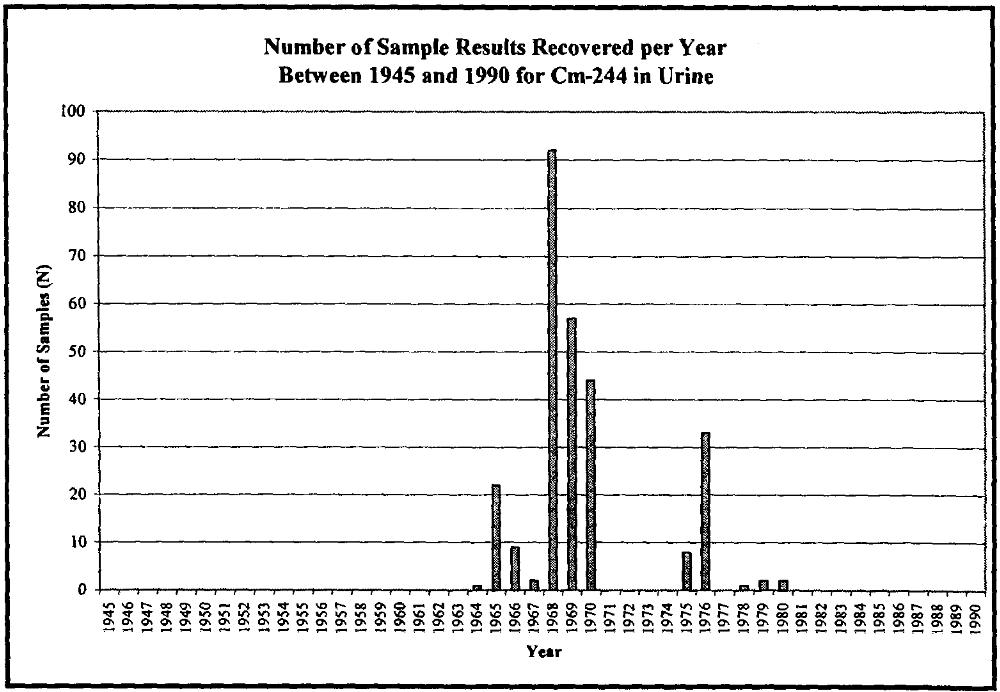

Figure A-2. MDA data for curium-244 in urine (part 3 of 3).

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CALCULATION OF MDAS FROM HISTORICAL DATA

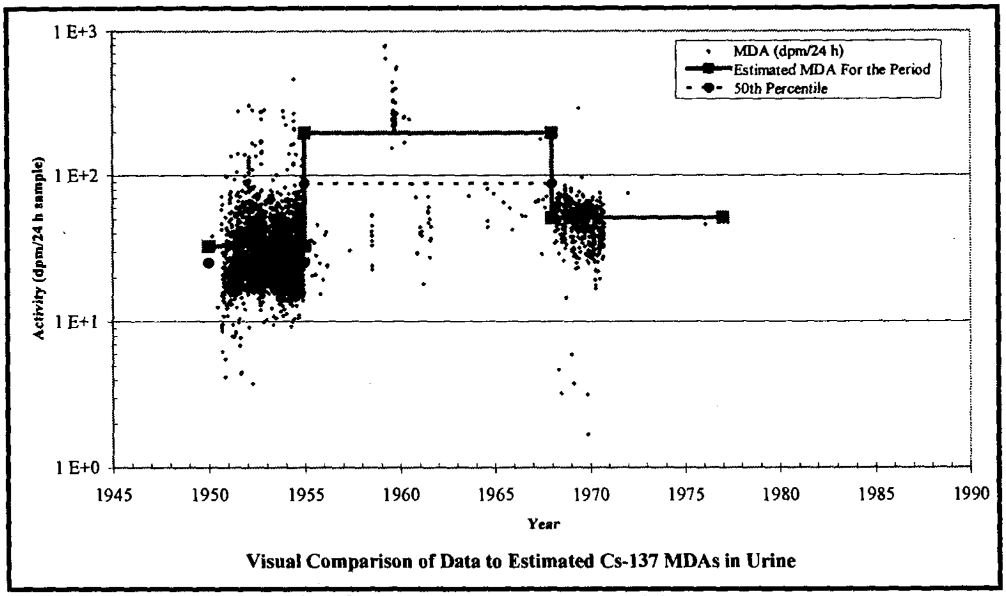

Figure A-3. MDA data for Cs-137 in urine (part 1 of 3).

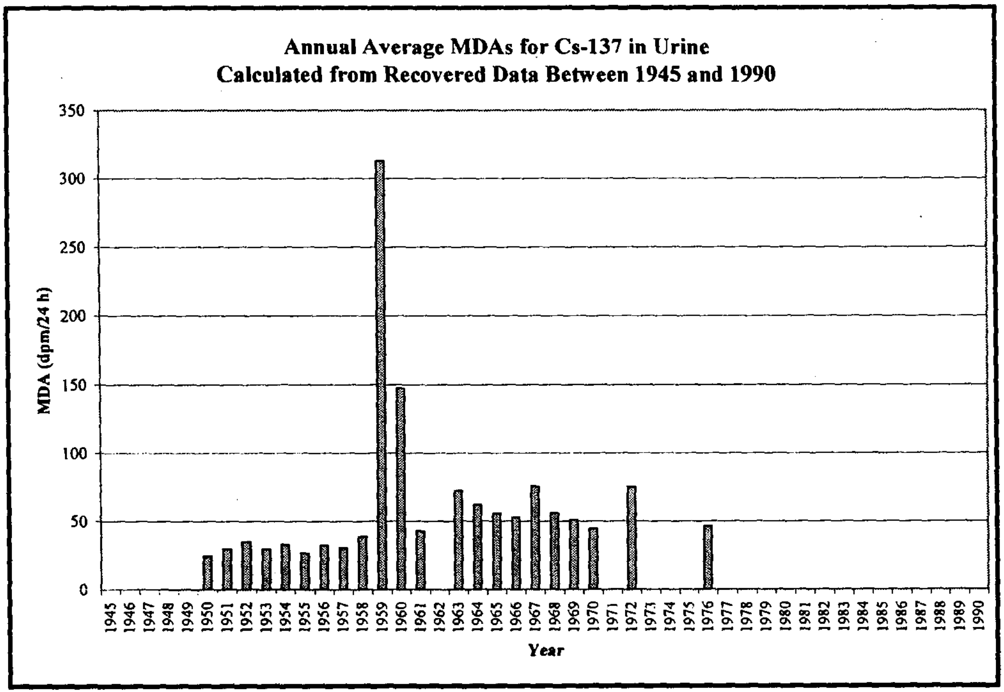

Figure A-3. MDA data for Cs-137 in urine (part 2 of 3).

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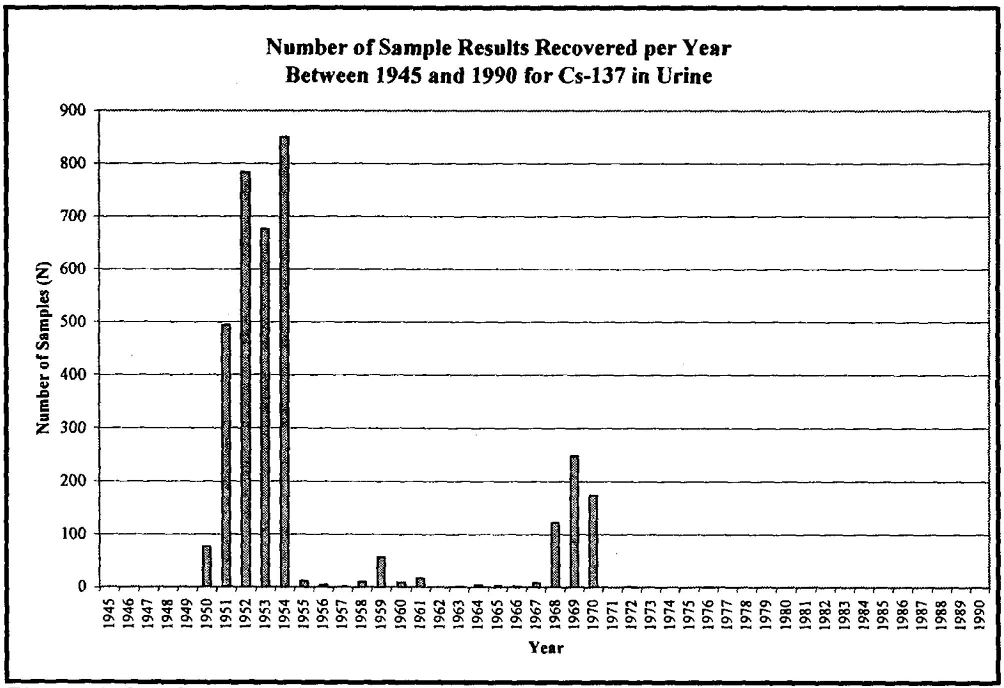

Figure A-3. MDA data for Cs-137 in urine (part 3 of 3).

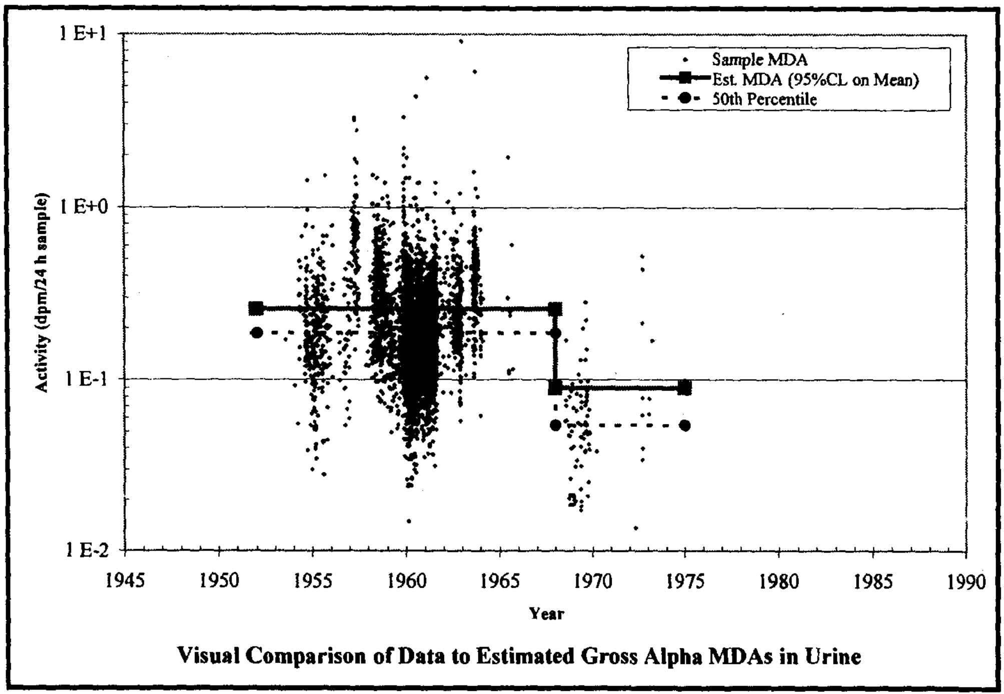

Figure A-4. MDA data for gross alpha in urine (part 1 of 3).

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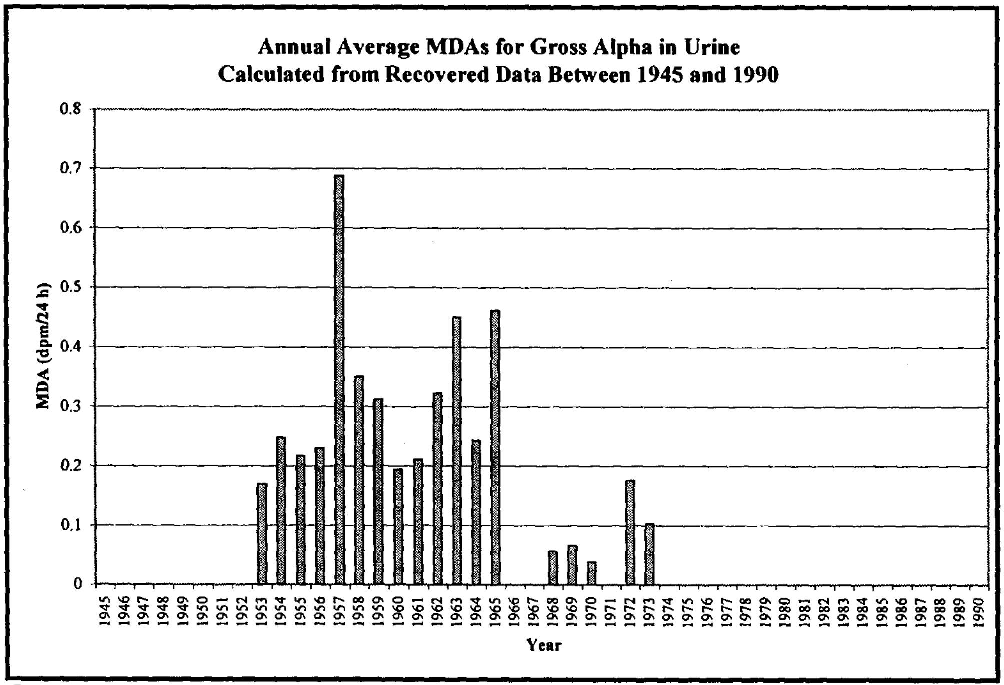

Figure A-4. MDA data for gross alpha in urine (part 2 of 3).

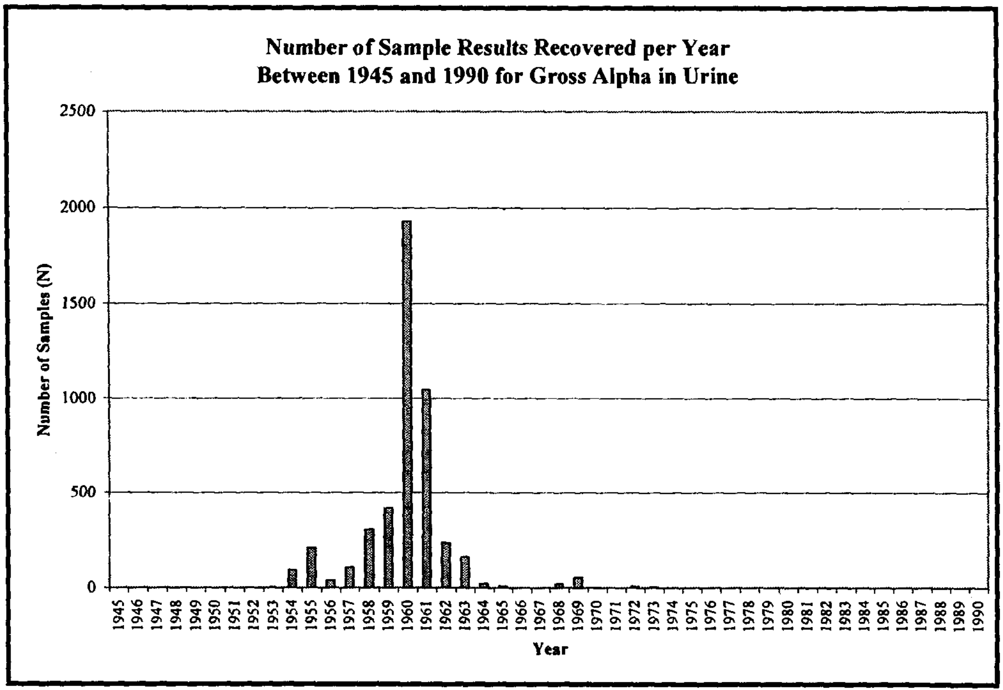

Figure A-4. MDA data for gross alpha in urine (part 3 of 3).

ATTACHMENT A
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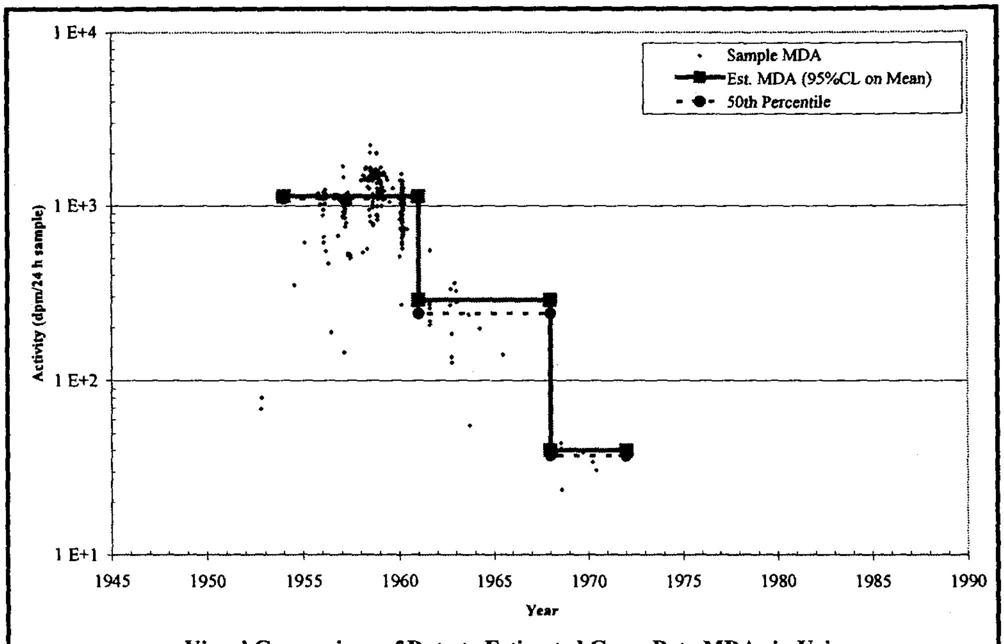

Figure A-5. MDA data for gross beta in urine (part 1 of 3).

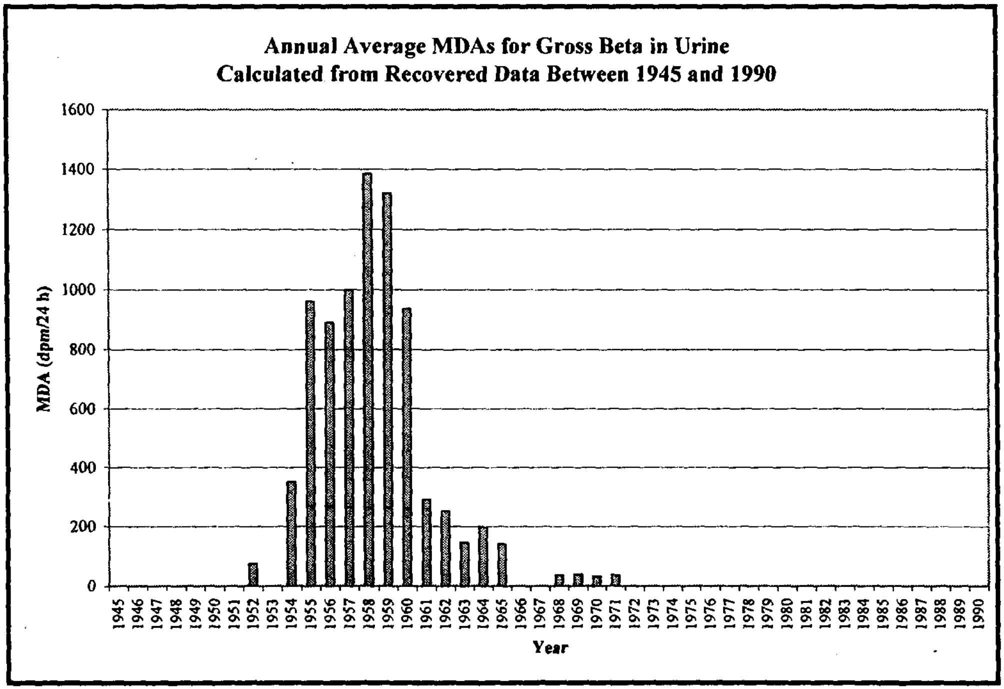

Figure A-5. MDA data for gross beta in urine (part 2 of 3).

ATTACHMENT A
CALCULATION OF MDAS FROM HISTORICAL DATA

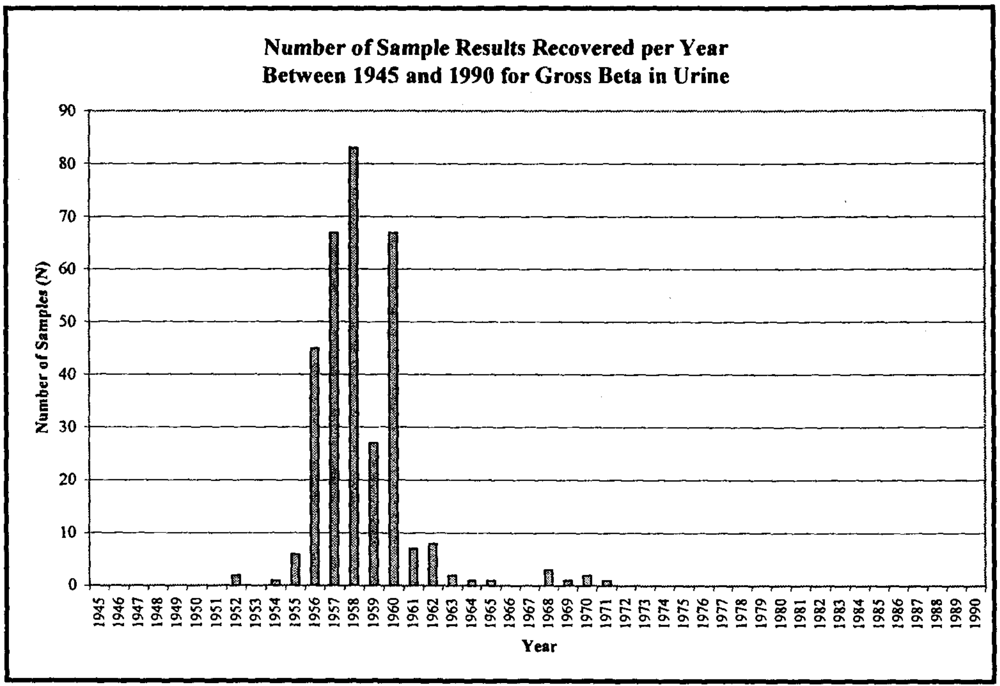

Figure A-5. MDA data for gross beta in urine (part 3 of 3).

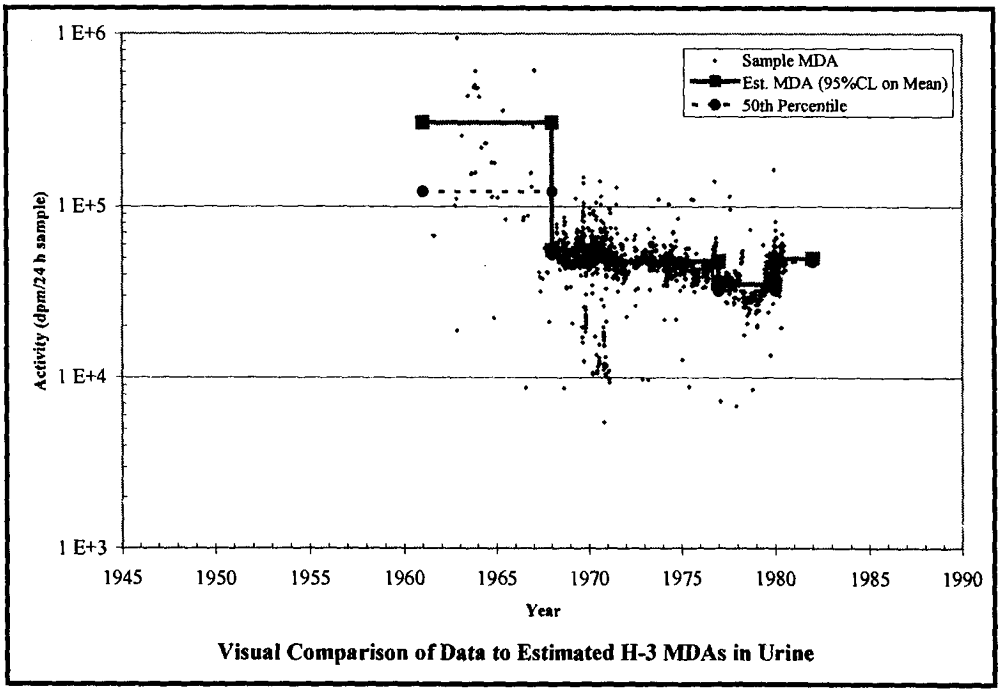

Figure A-6. MDA data for H-3 in urine (part 1 of 3).

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CALCULATION OF MDAS FROM HISTORICAL DATA

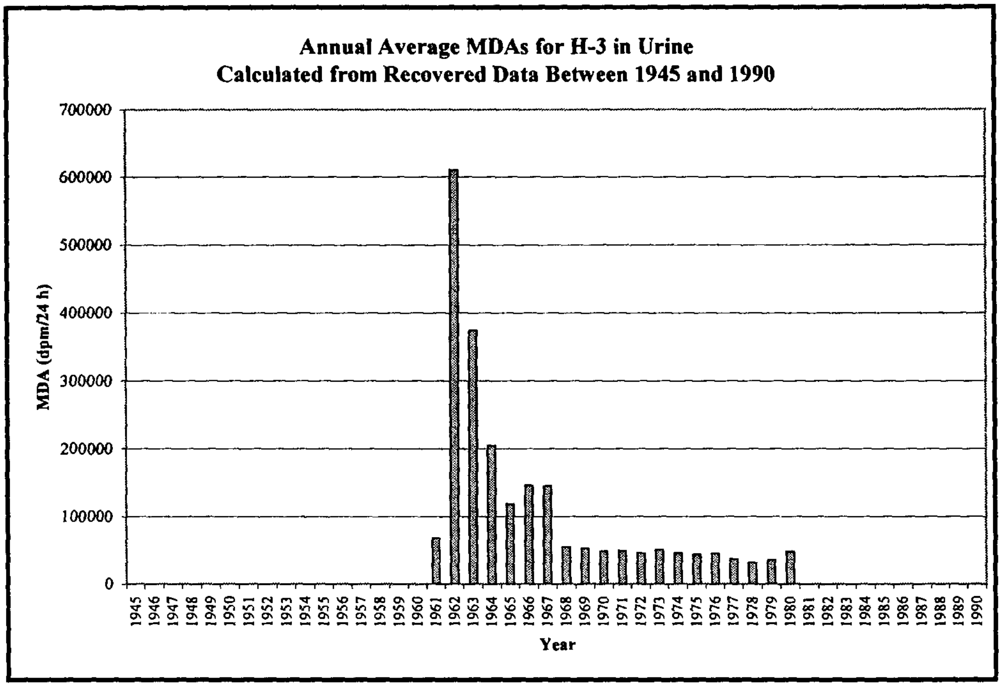


Figure A-6. MDA data for H-3 in urine (part 2 of 3).

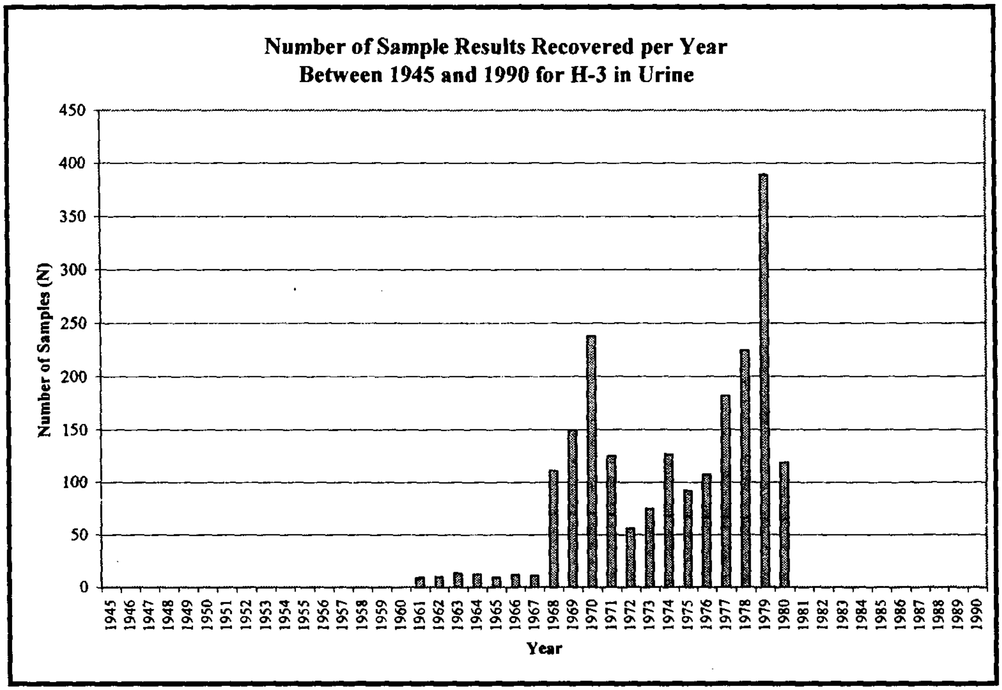
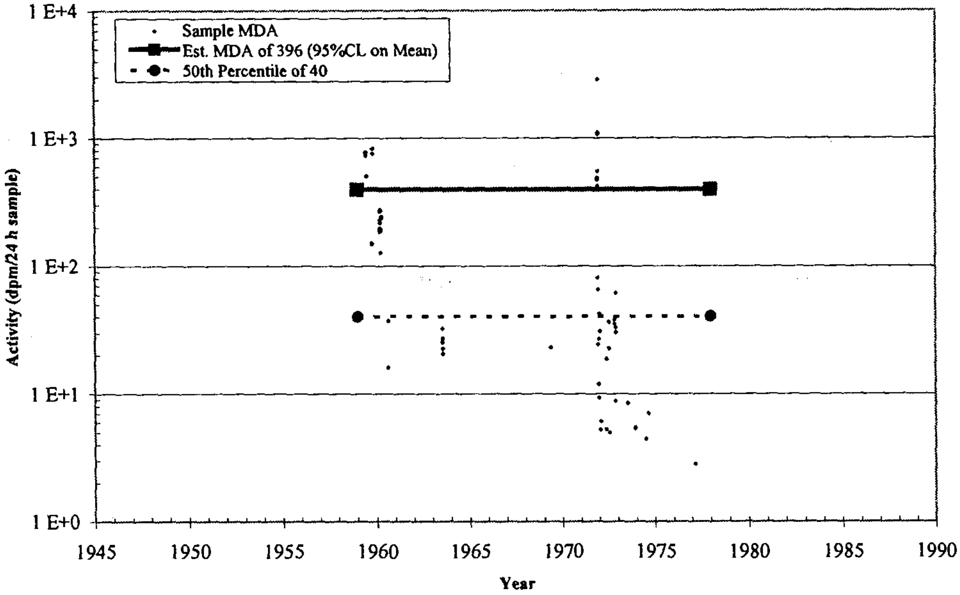


Figure A-6. MDA data for H-3 in urine (part 3 of 3).

ATTACHMENT A
CALCULATION OF MDAS FROM HISTORICAL DATA



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).

Annual Average MDAs for Pm-147 in Urine
Calculated from Recovered Data Between 1945 and 1990

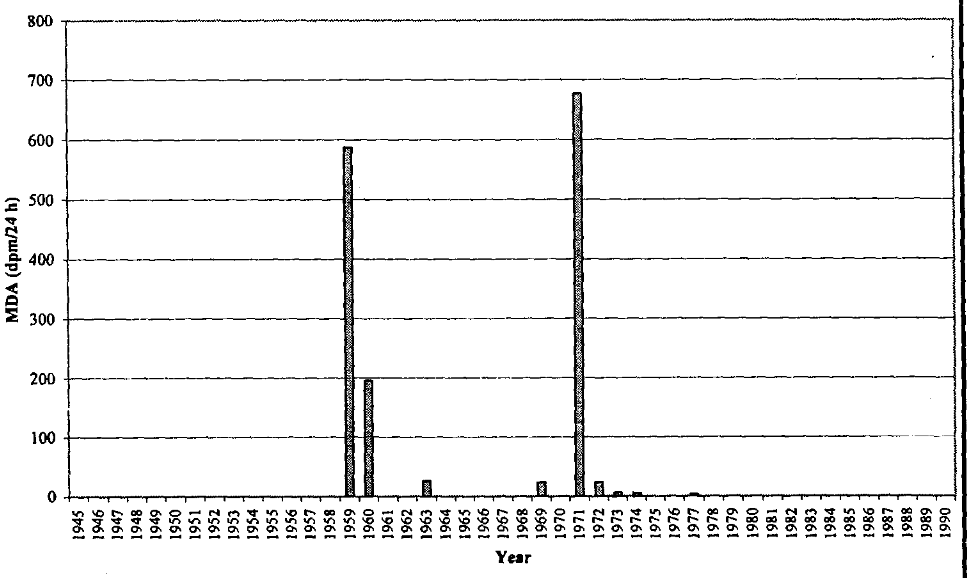


Figure A-7. MDA data for Pm-147 in urine (part 2 of 3).

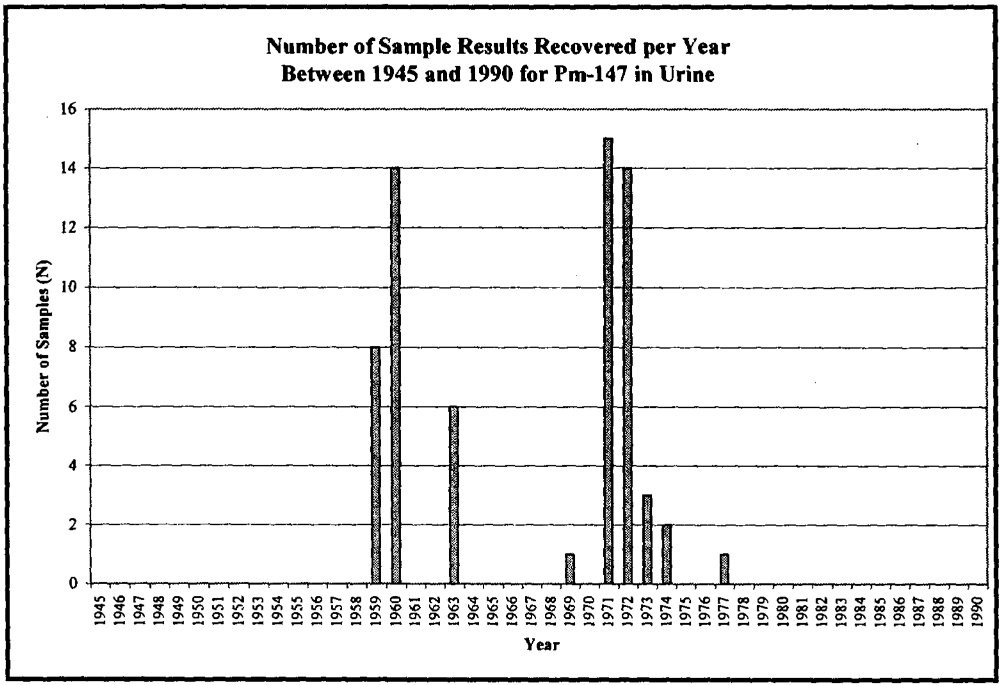


Figure A-7. MDA data for Pm-147 in urine (part 3 of 3).

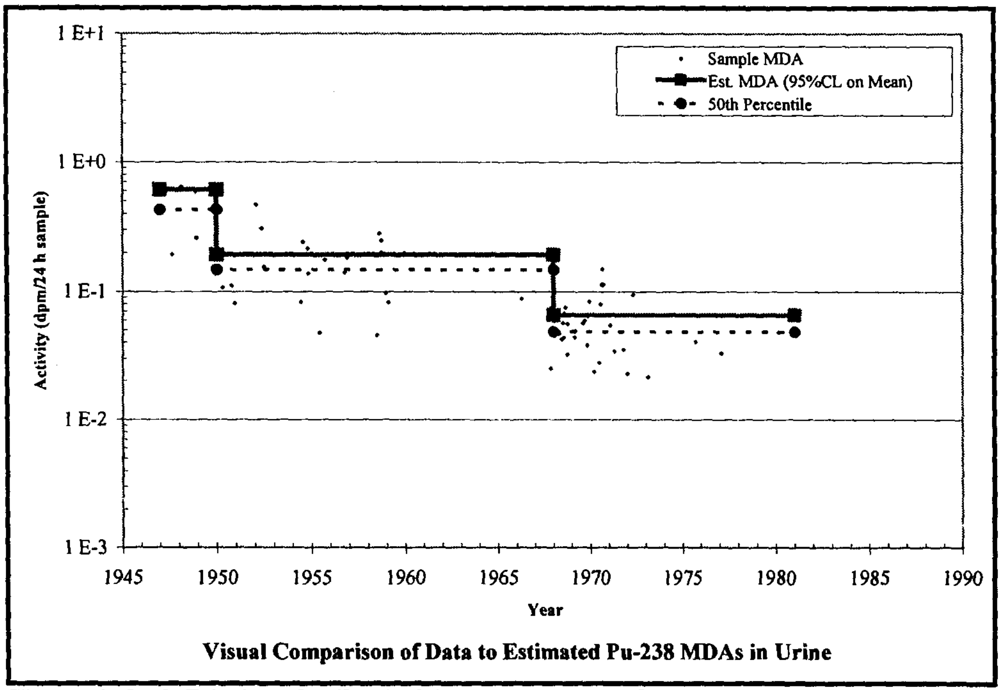


Figure A-8. MDA data for Pu-238 in urine (part 1 of 3).

ATTACHMENT A
CALCULATION OF MDAS FROM HISTORICAL DATA

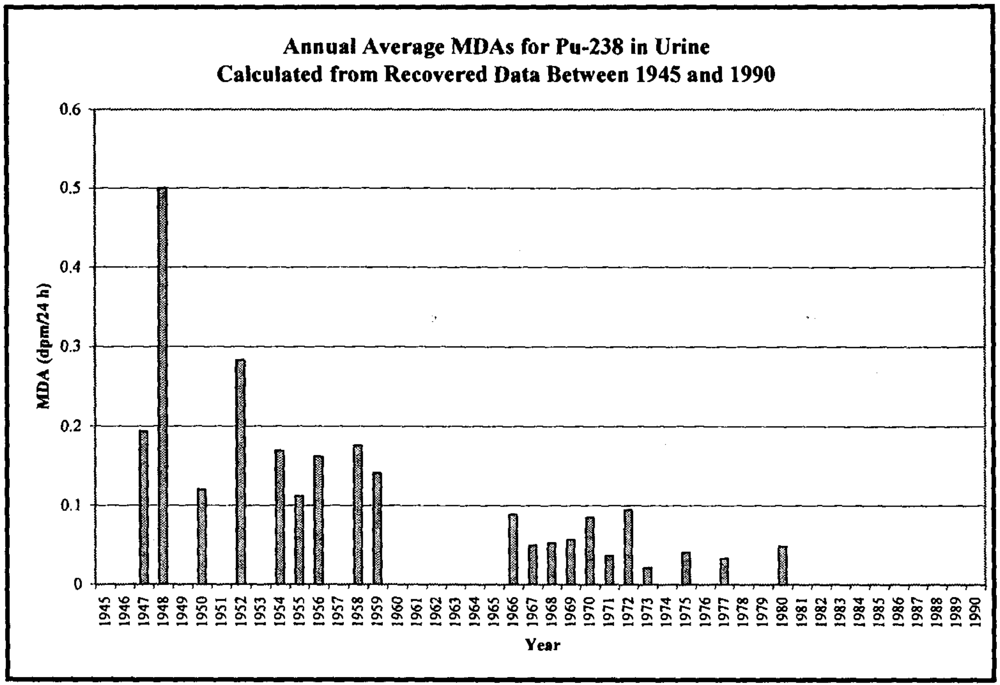


Figure A-8. MDA data for Pu-238 in urine (part 2 of 3).

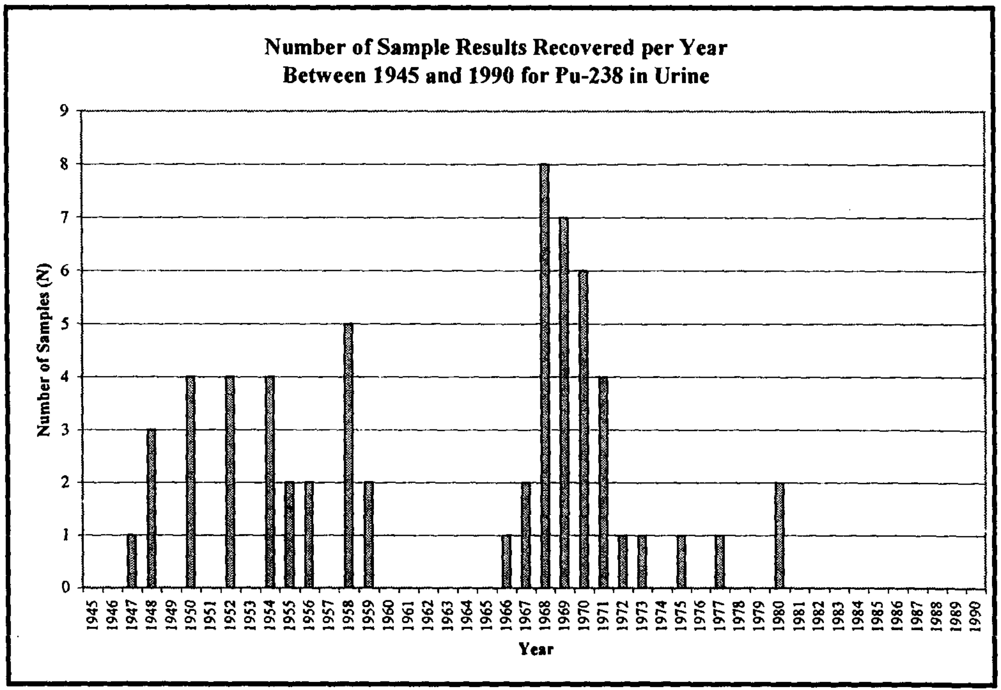


Figure A-8. MDA data for Pu-238 in urine (part 3 of 3).

ATTACHMENT A
CALCULATION OF MDAS FROM HISTORICAL DATA

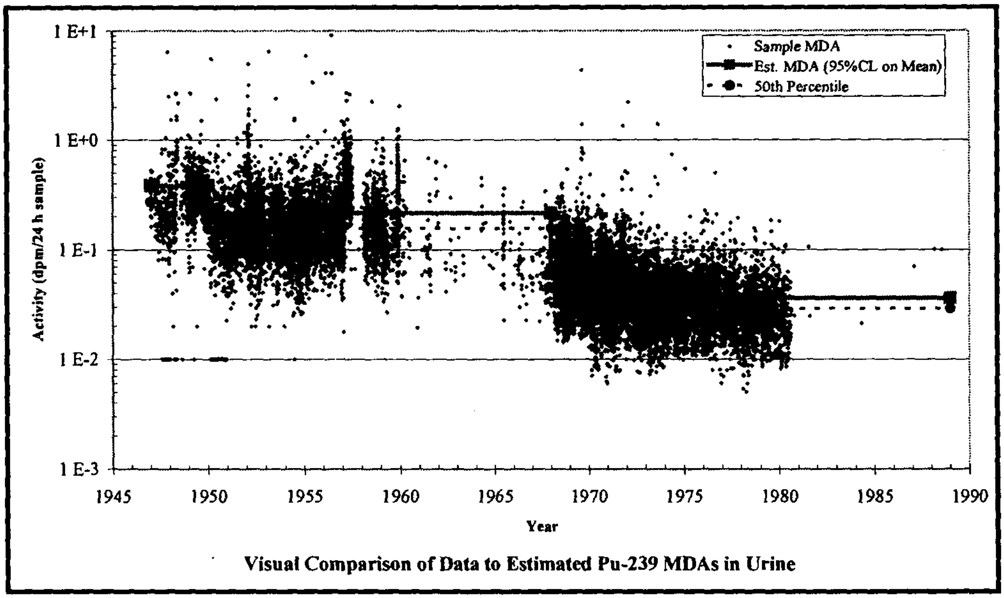


Figure A-9. MDA data for Pu-239 in urine (part 1 of 3).

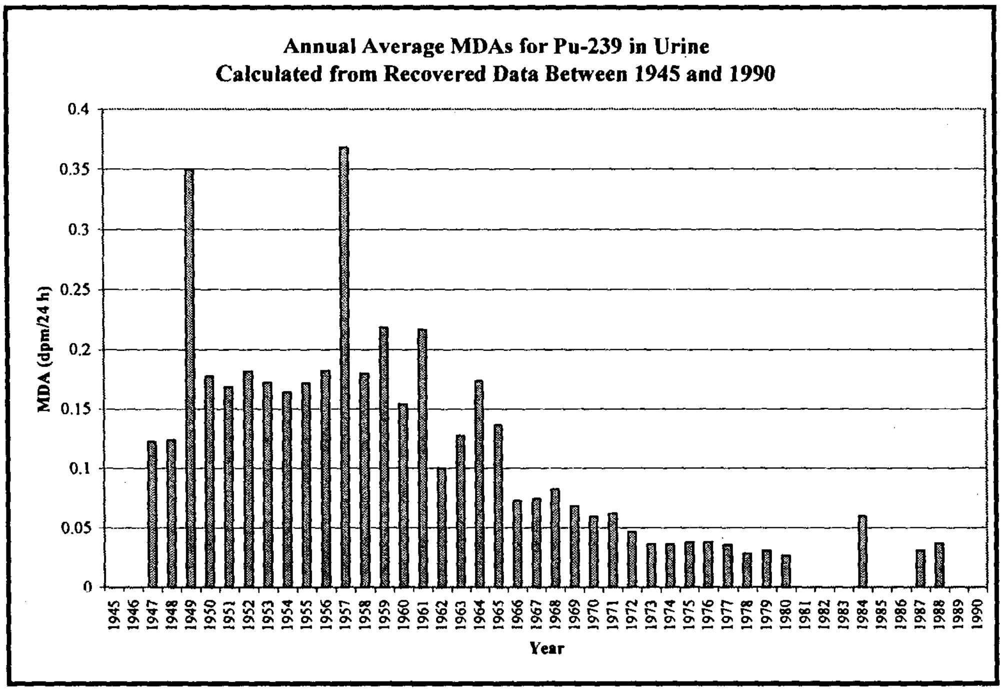


Figure A-9. MDA data for Pu-239 in urine (part 2 of 3).

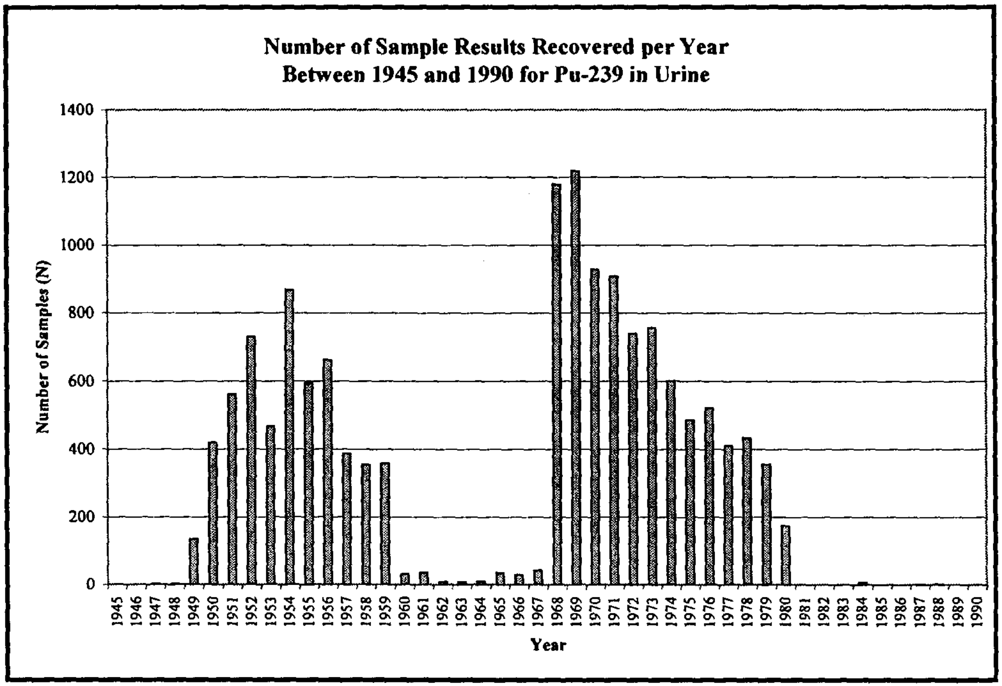


Figure A-9. MDA data for Pu-239 in urine (part 3 of 3).

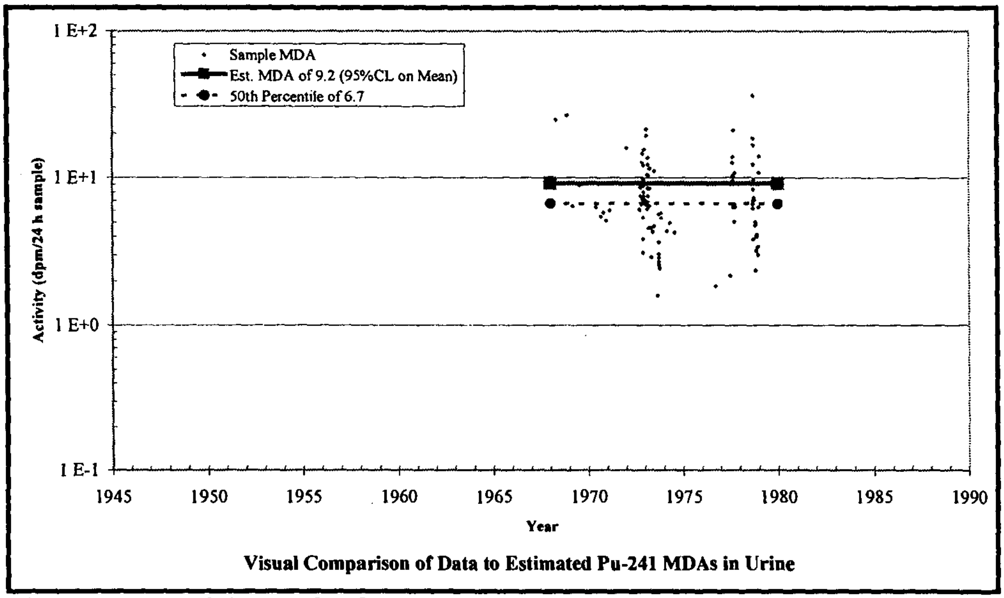


Figure A-10. MDA data for Pu-241 in urine (part 1 of 3).

ATTACHMENT A
CALCULATION OF MDAS FROM HISTORICAL DATA

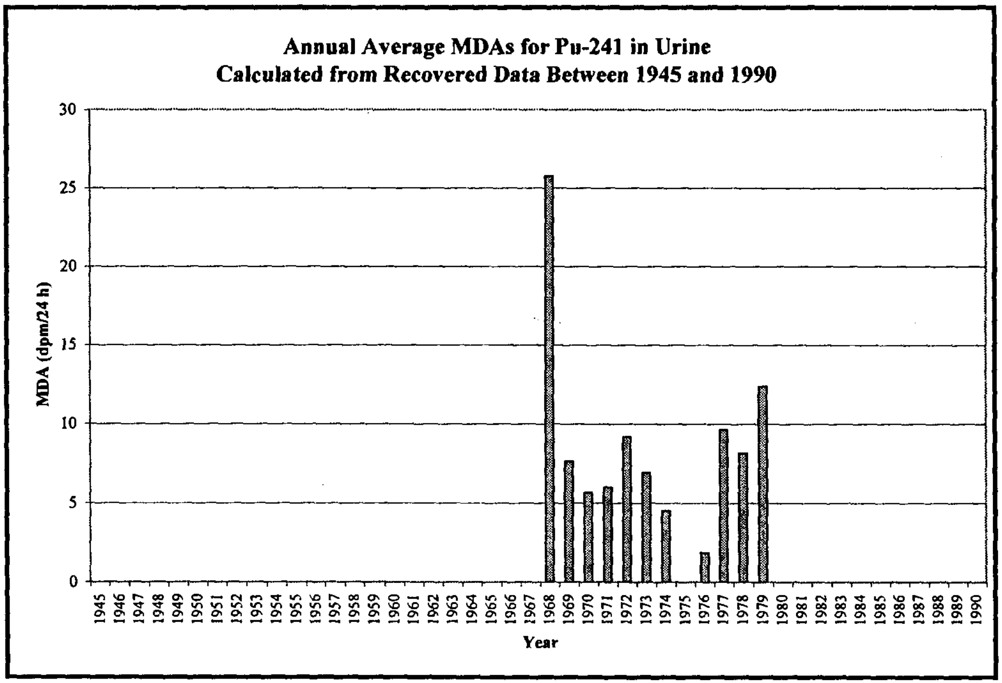


Figure A-10. MDA data for Pu-241 in urine (part 2 of 3).

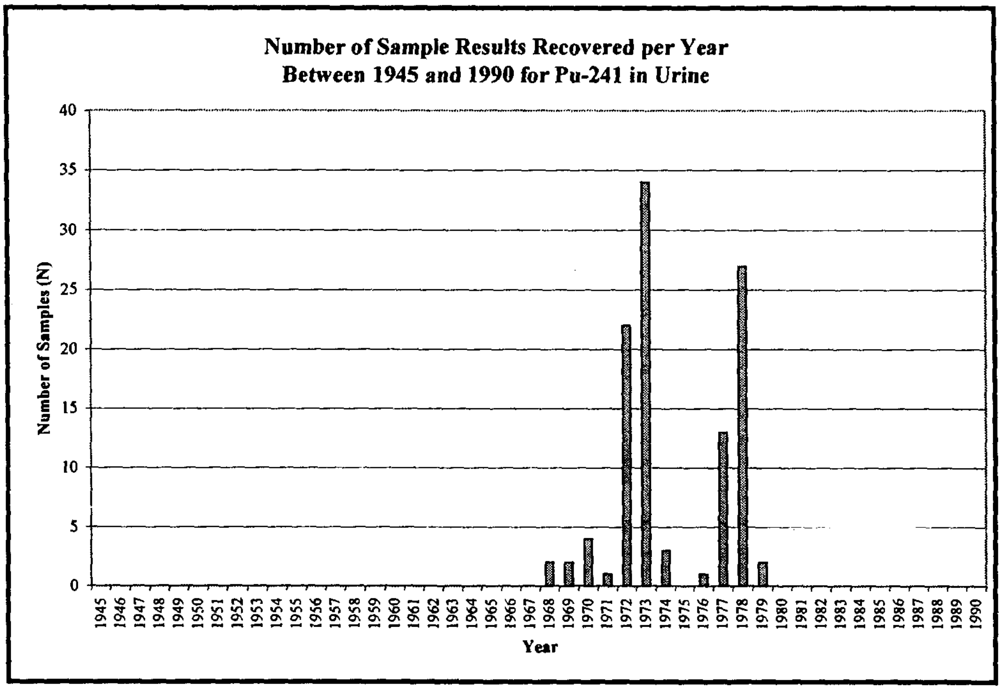


Figure A-10. MDA data for Pu-241 in urine (part 3 of 3).

ATTACHMENT A
CALCULATION OF MDAS FROM HISTORICAL DATA

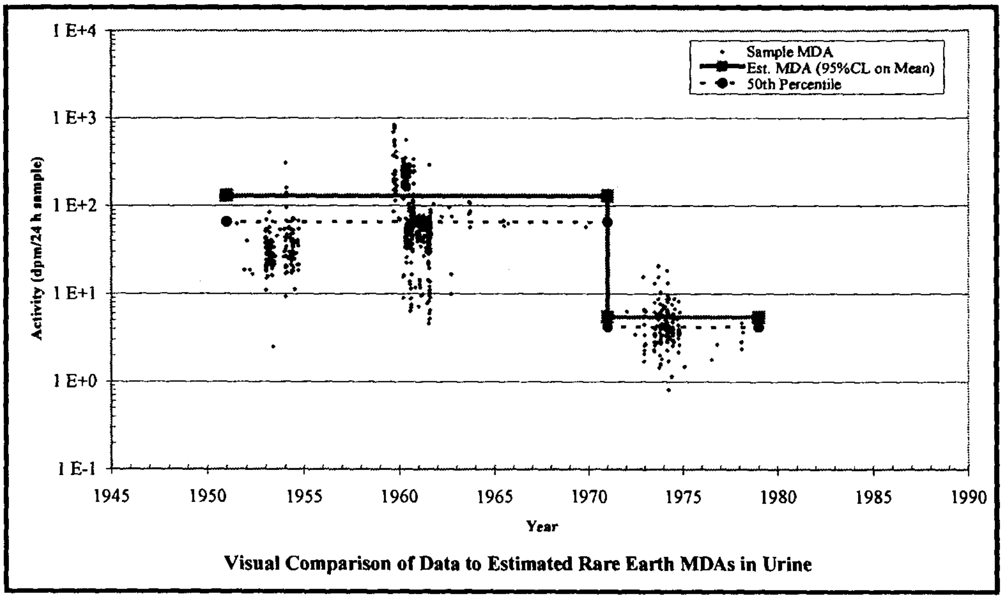


Figure A-11. MDA data for rare earths in urine (part 1 of 3).

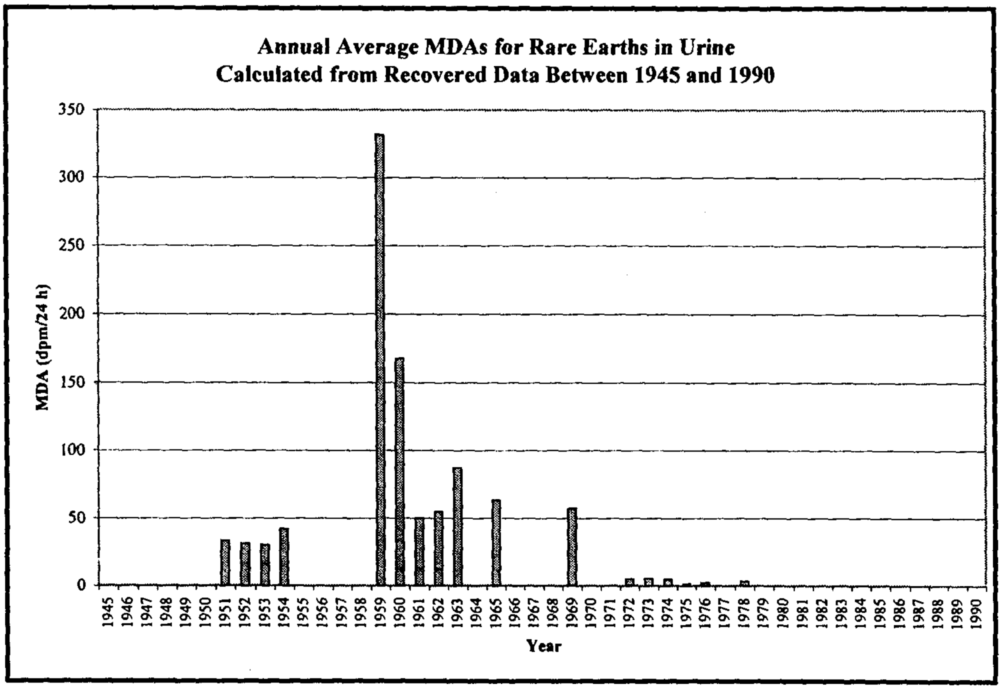


Figure A-11. MDA data for rare earths in urine (part 2 of 3).

ATTACHMENT A
CALCULATION OF MDAS FROM HISTORICAL DATA

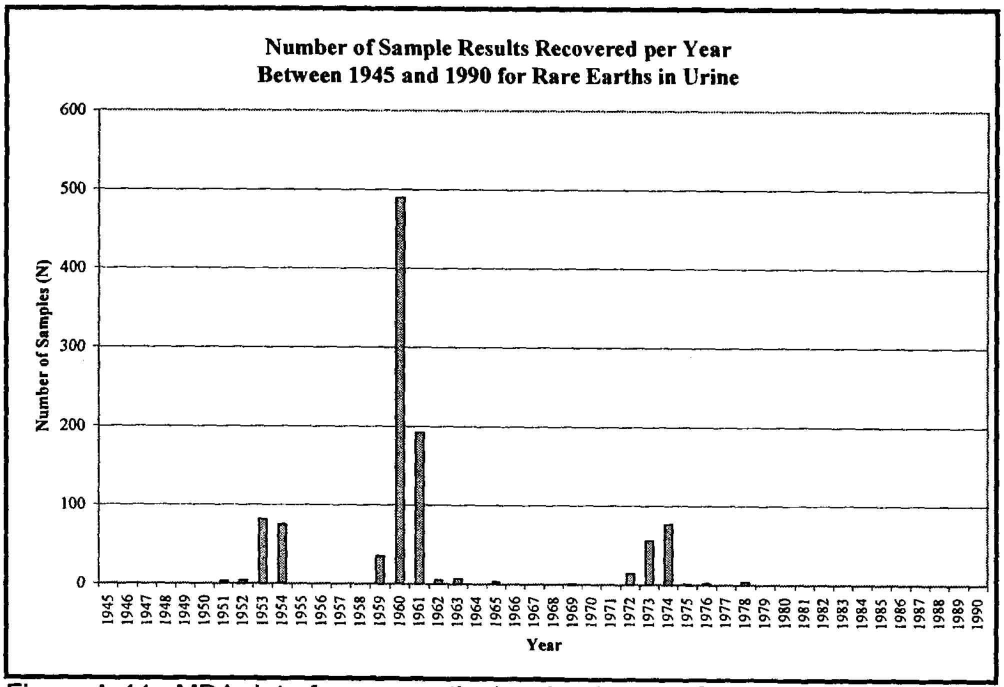


Figure A-11. MDA data for rare earths in urine (part 3 of 3).

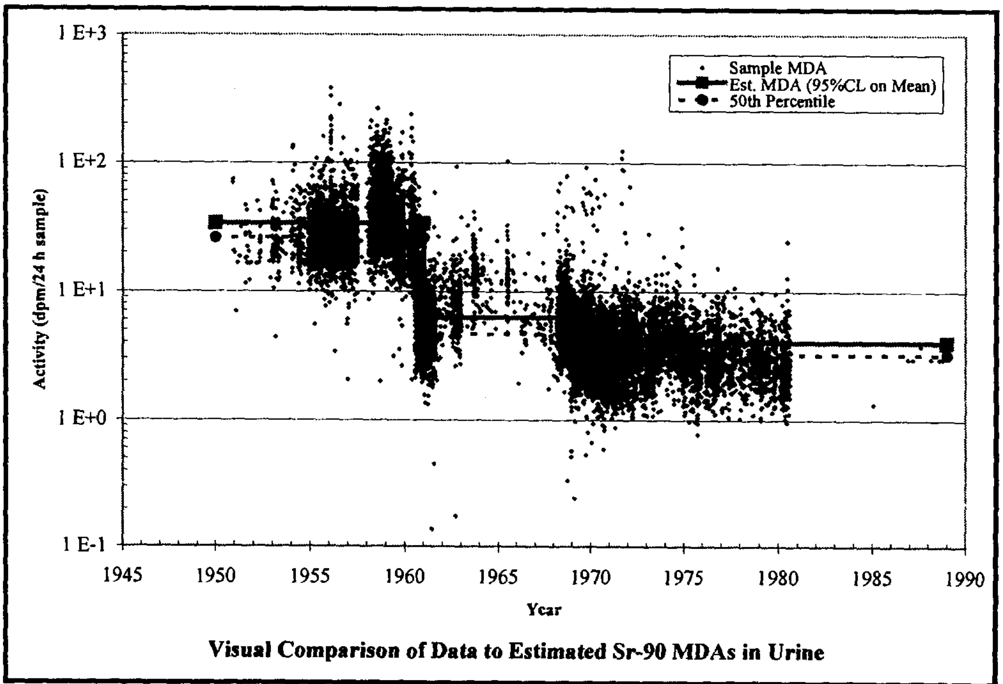


Figure A-12. MDA data for Sr-90 in urine (part 1 of 3).

ATTACHMENT A
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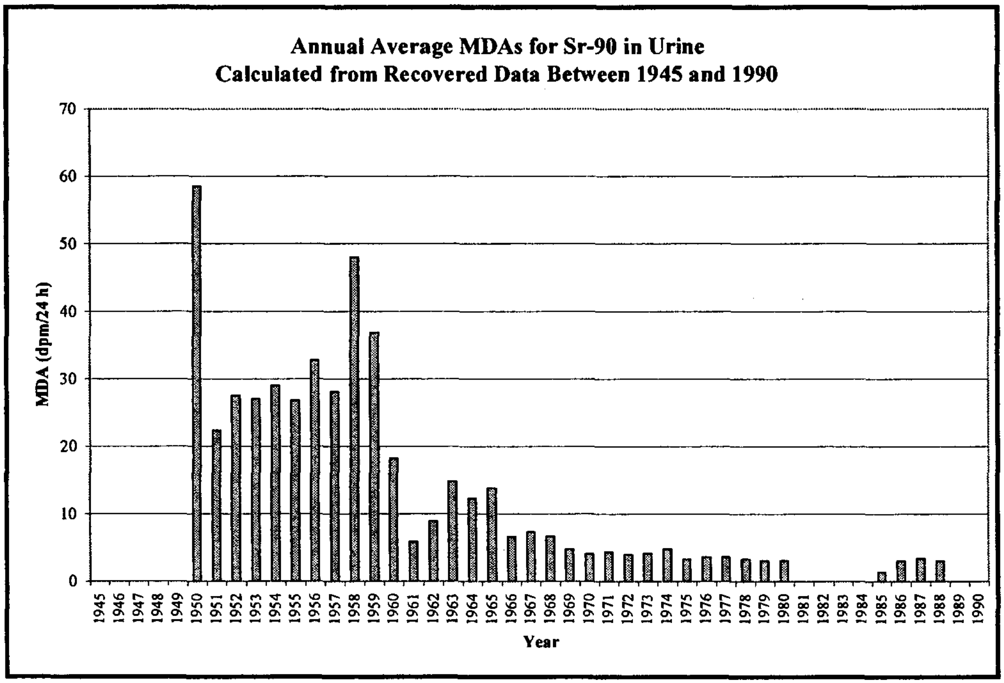

Figure A-12. MDA data for Sr-90 in urine (part 2 of 3).

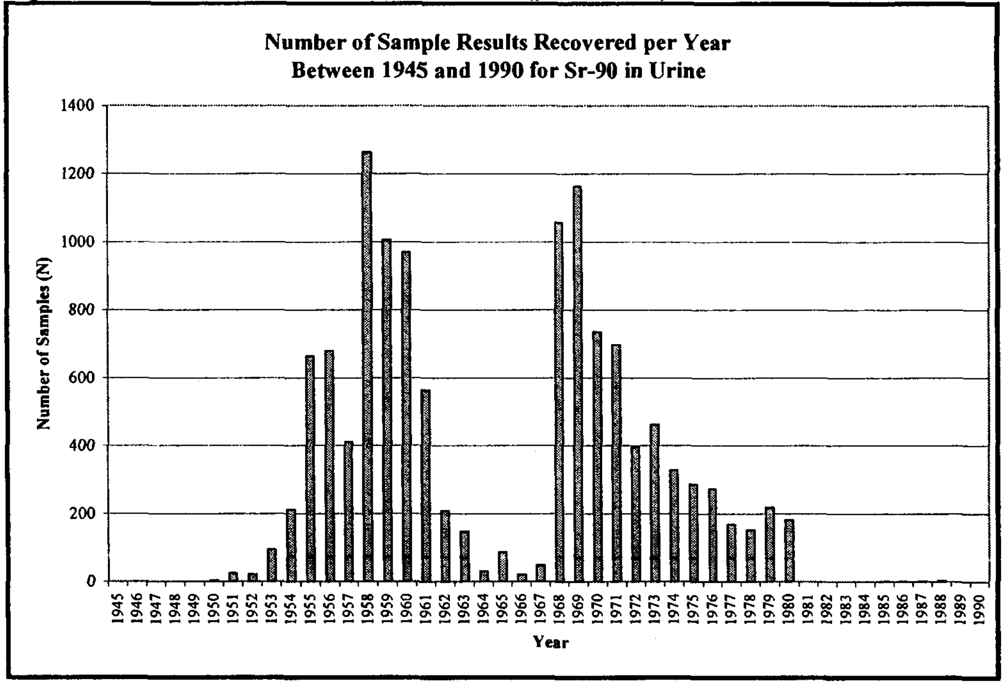

Figure A-12. MDA data for Sr-90 in urine (part 3 of 3).

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CALCULATION OF MDAS FROM HISTORICAL DATA

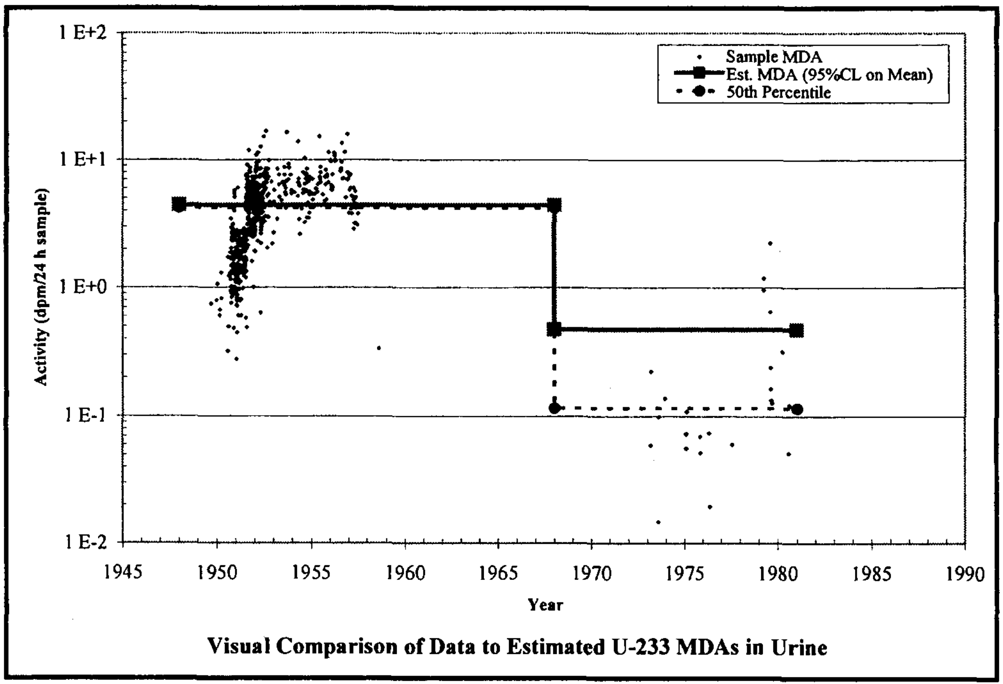

Figure A-13. MDA data for U-233 in urine (part 1 of 3).

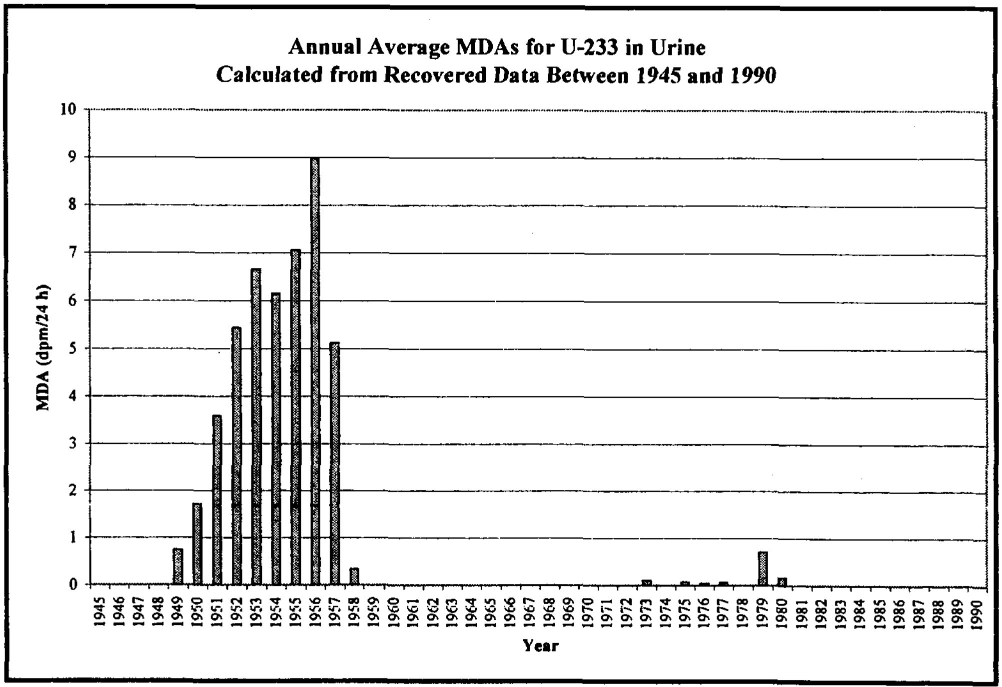

Figure A-13. MDA data for U-233 in urine (part 2 of 3).

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CALCULATION OF MDAS FROM HISTORICAL DATA

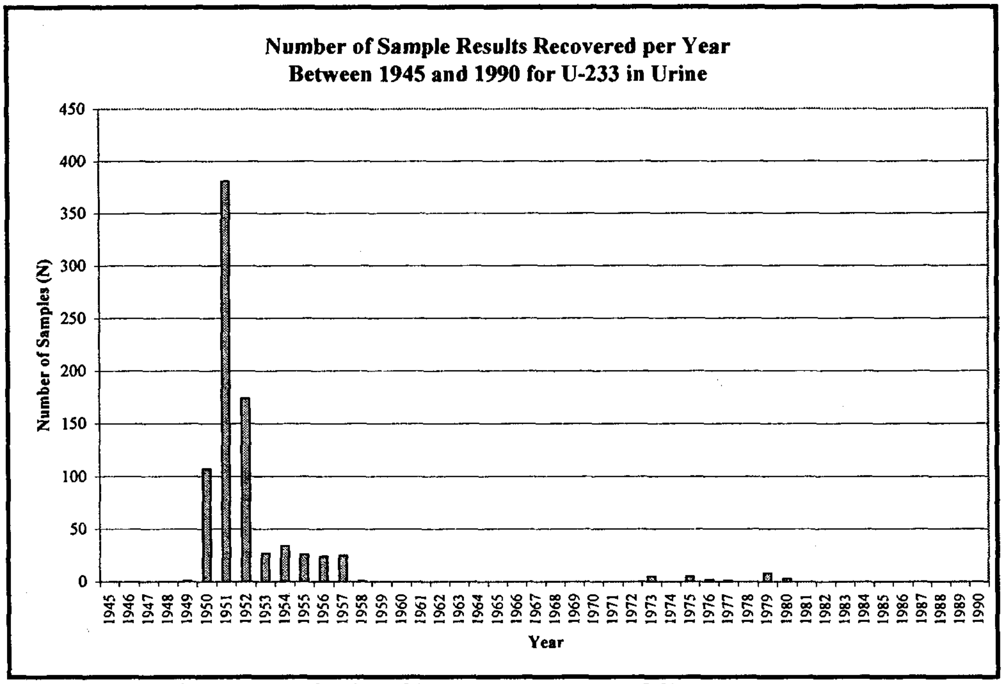

Figure A-13. MDA data for U-233 in urine (part 3 of 3).

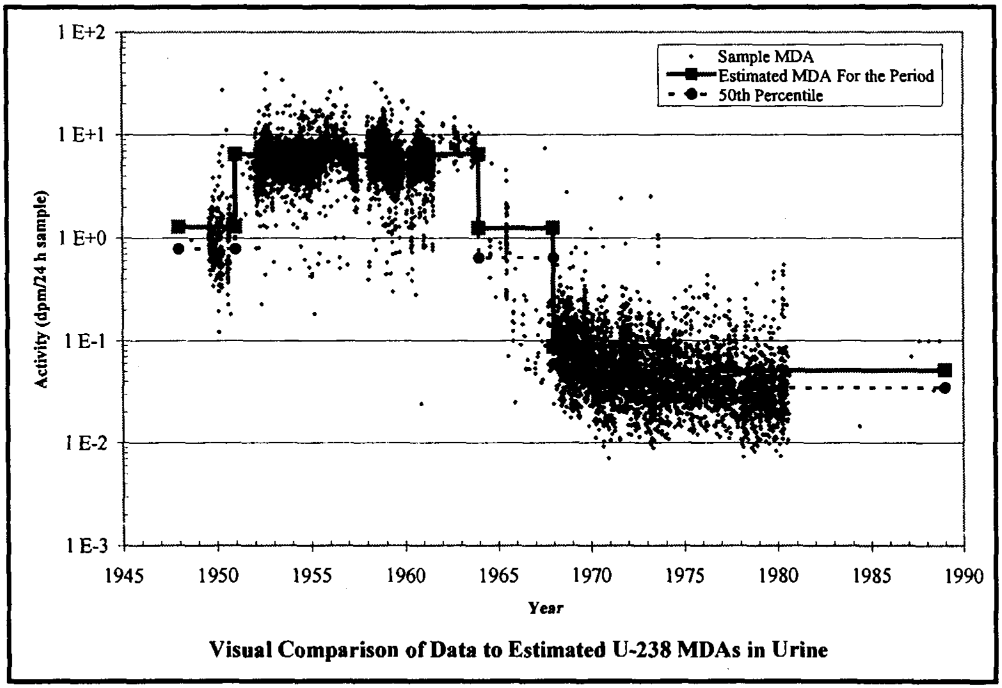

Figure A-14. MDA data for U-238 in urine (part 1 of 3).

ATTACHMENT A
CALCULATION OF MDAS FROM HISTORICAL DATA

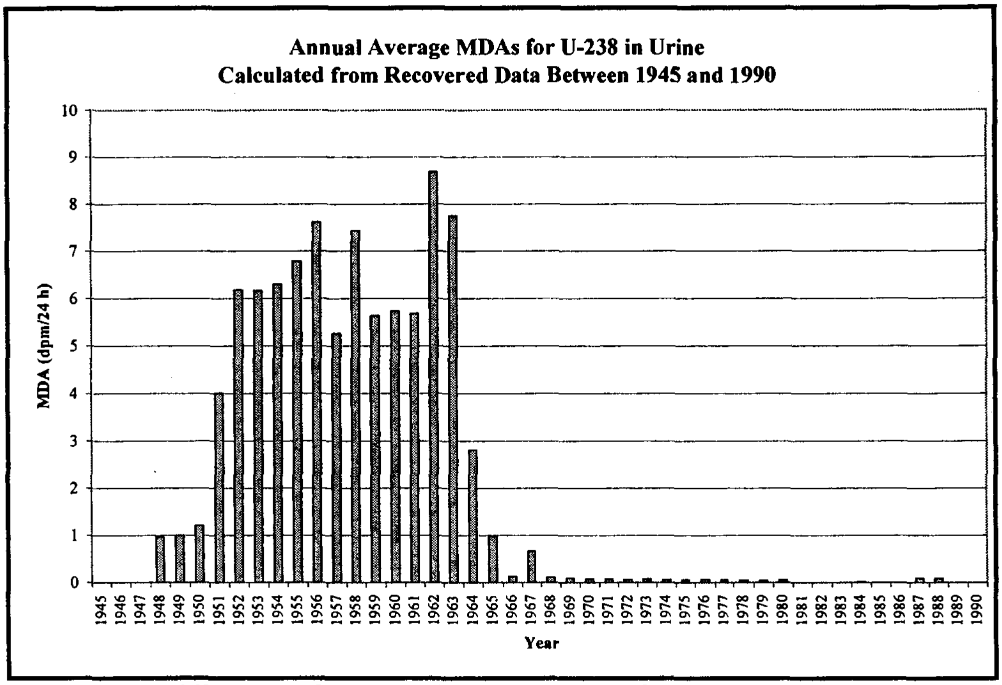

Figure A-14. MDA data for U-238 in urine (part 2 of 3).

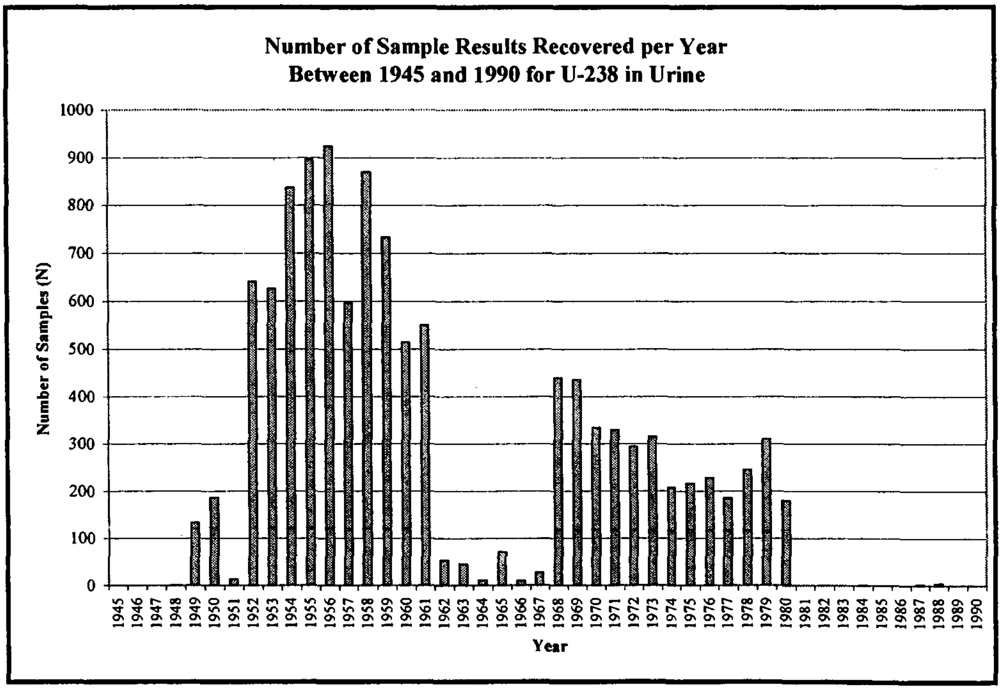

Figure A-14. MDA data for U-238 in urine (part 3 of 3).

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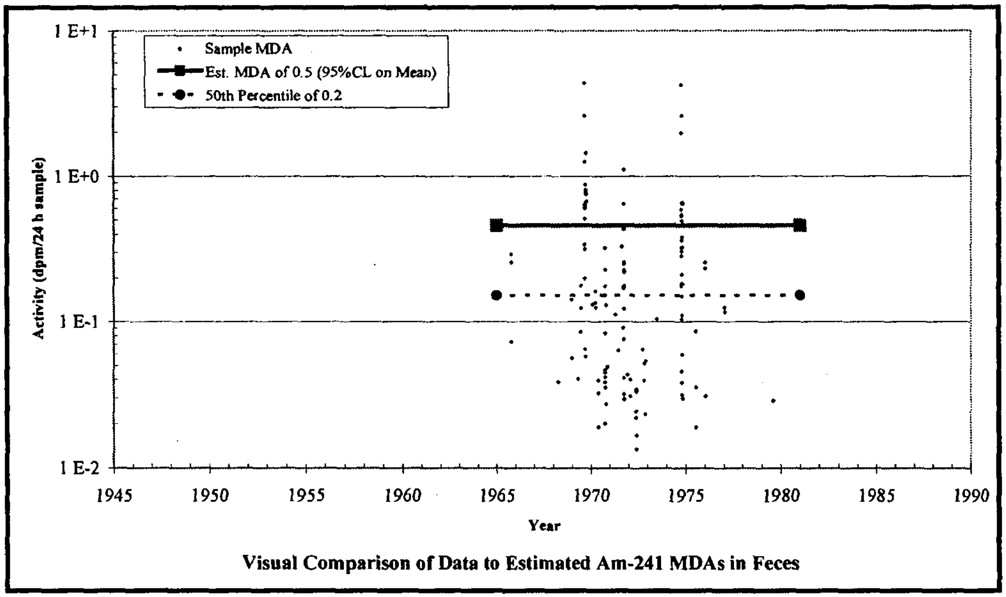

Figure A-15. MDA data for Am-241 in feces (part 1 of 3).

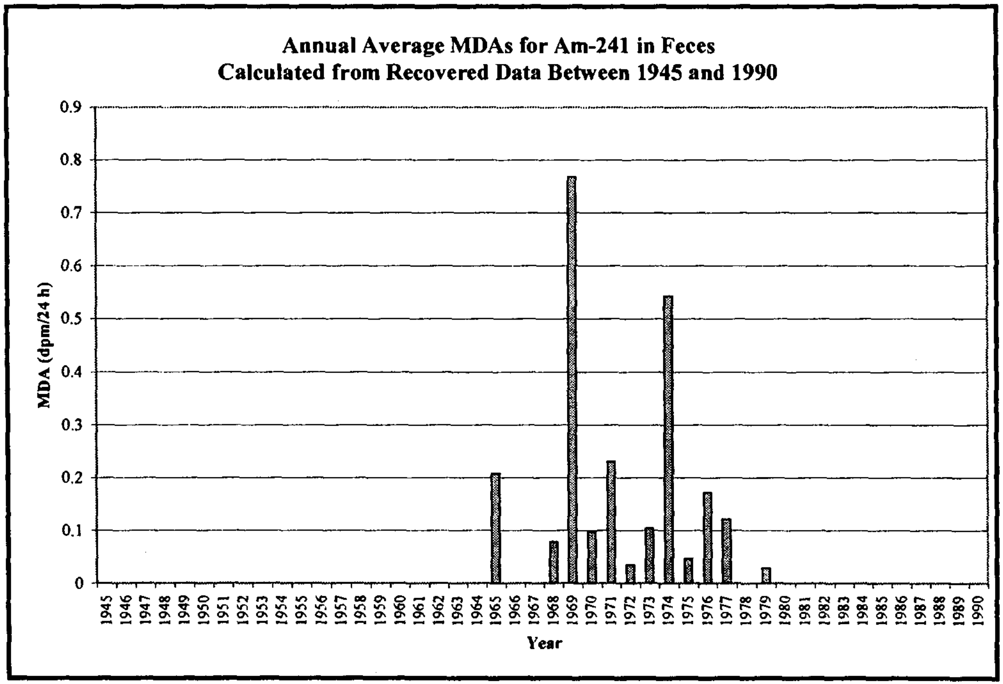

Figure A-15. MDA data for Am-241 in feces (part 2 of 3).

ATTACHMENT A
CALCULATION OF MDAS FROM HISTORICAL DATA

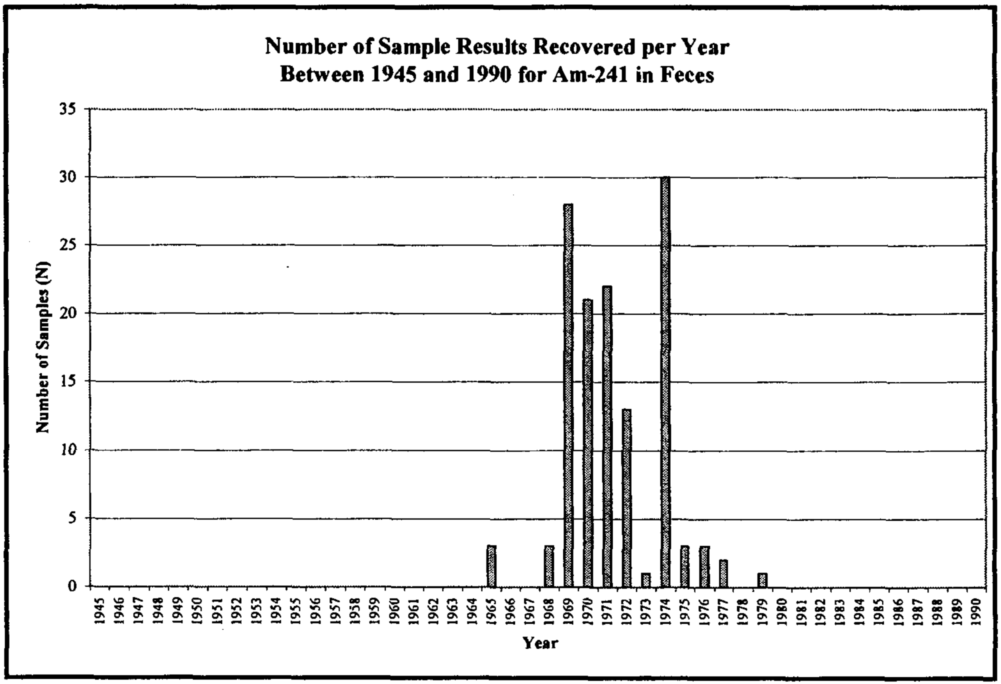

Figure A-15. MDA data for Am-241 in feces (part 3 of 3).

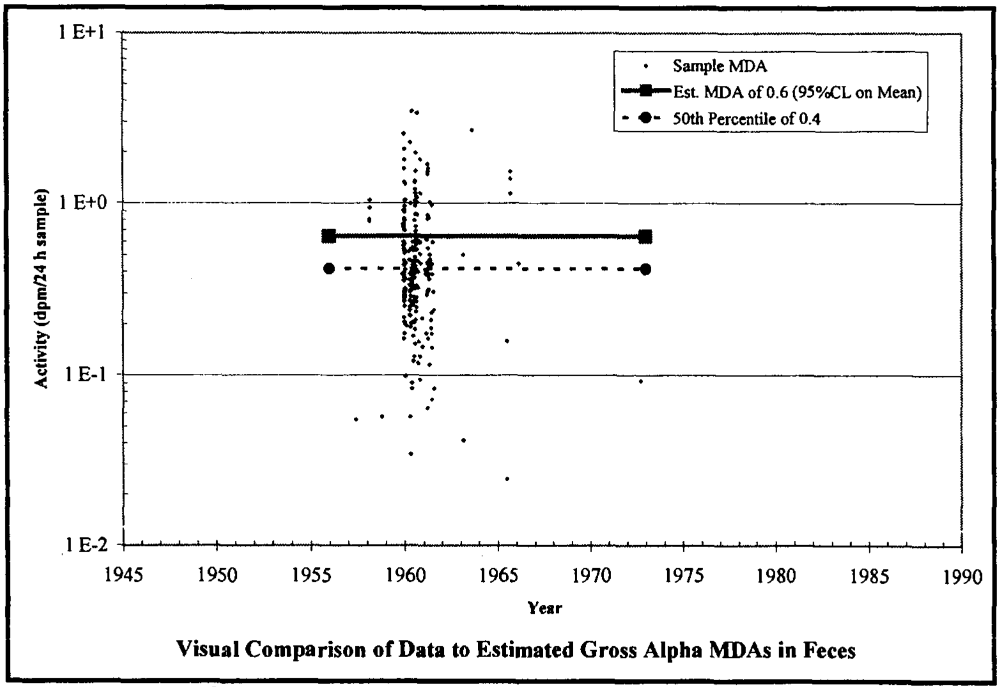

Figure A-16. MDA data for gross alpha in feces (part 1 of 3).

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CALCULATION OF MDAS FROM HISTORICAL DATA

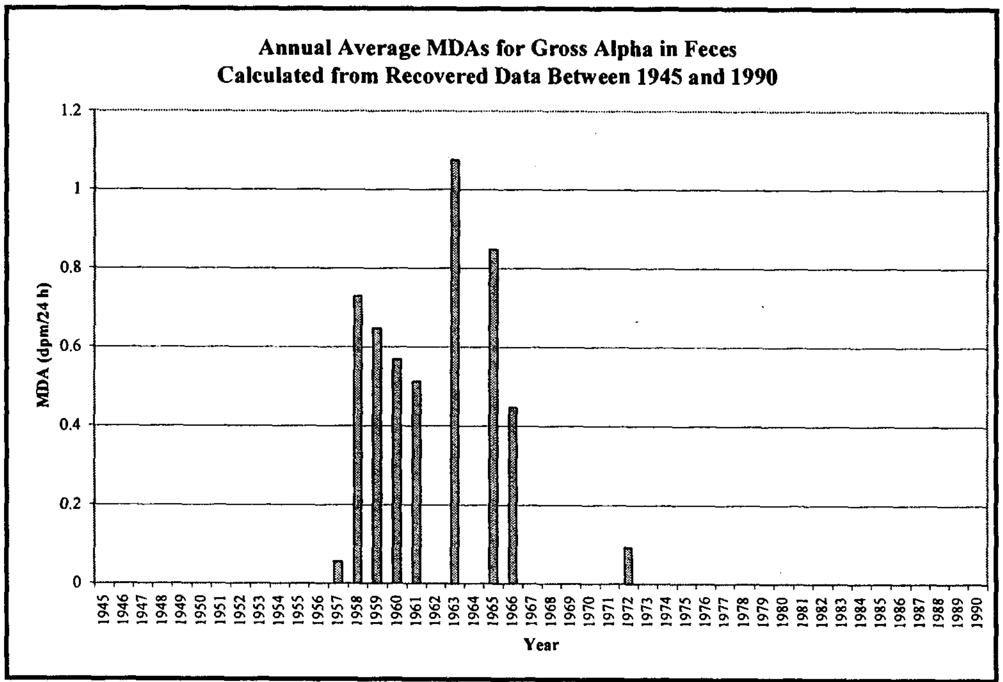

Figure A-16. MDA data for gross alpha in feces (part 2 of 3).

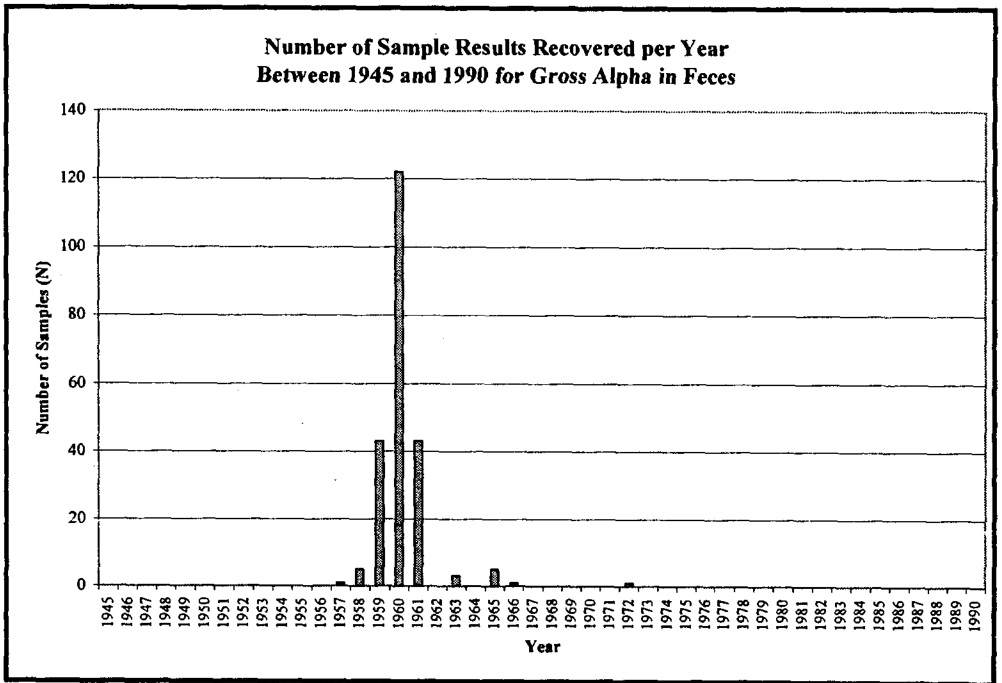

Figure A-16. MDA data for gross alpha in feces (part 3 of 3).

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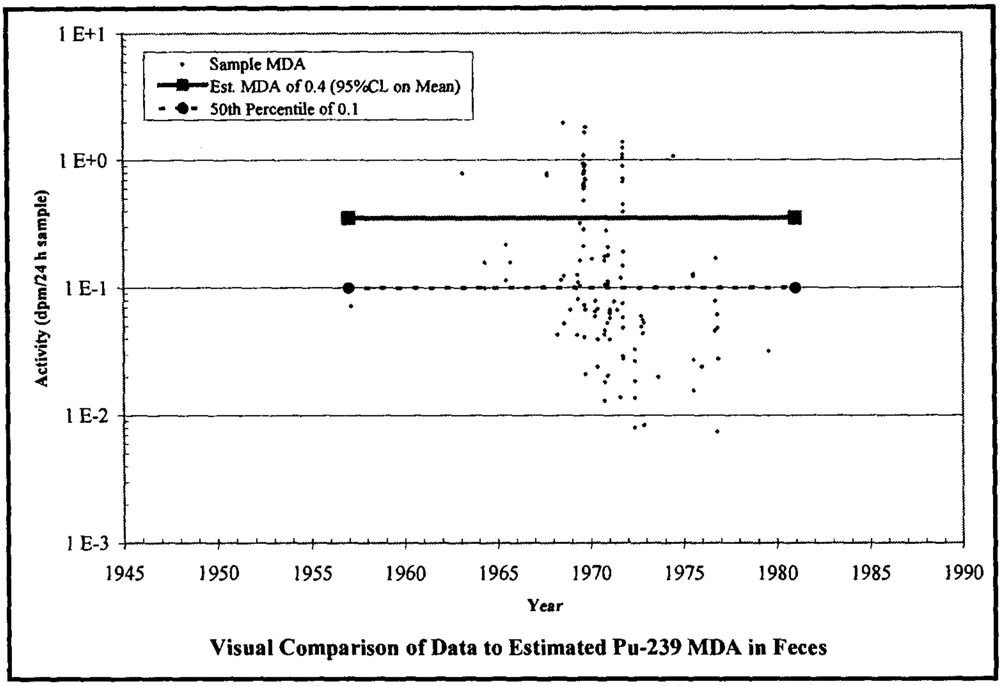

Figure A-17. MDA data for Pu-239 in feces (part 1 of 3).

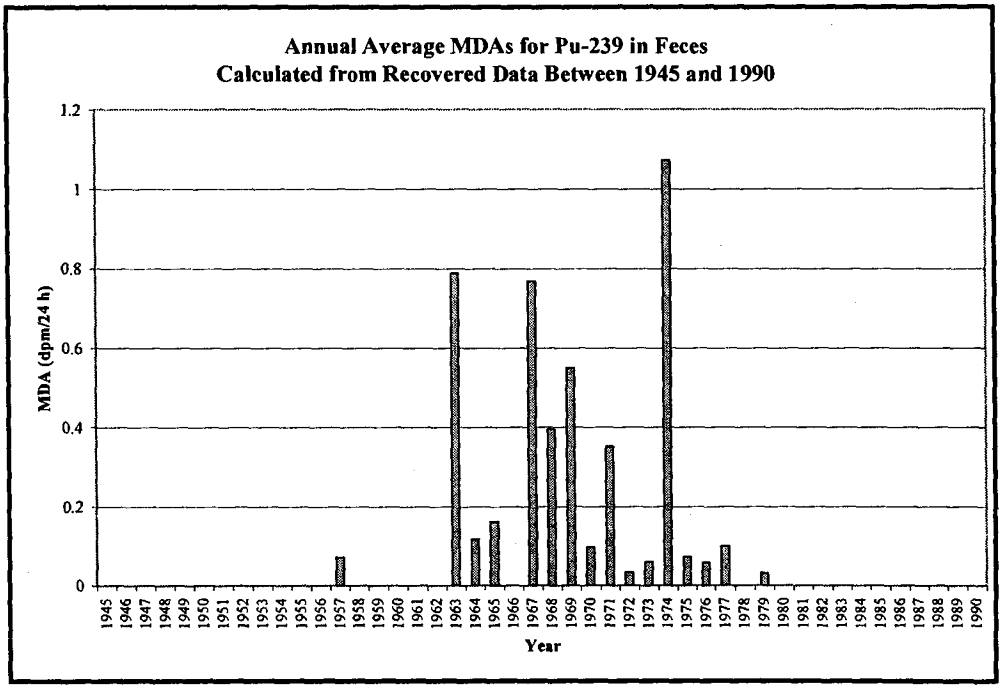

Figure A-17. MDA data for Pu-239 in feces (part 2 of 3).

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CALCULATION OF MDAS FROM HISTORICAL DATA

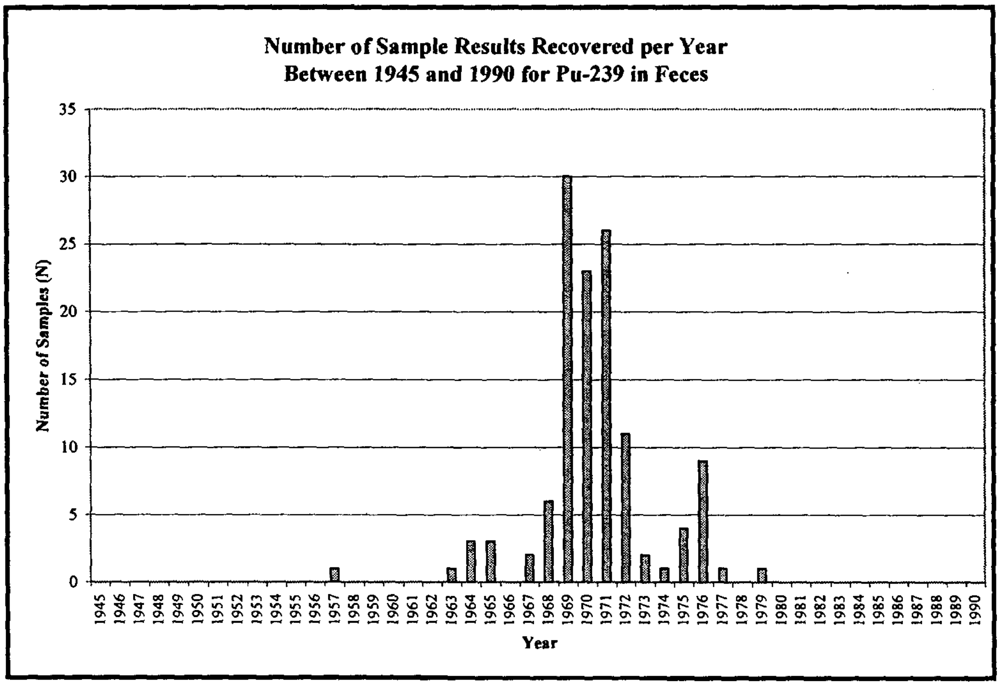

Figure A-17. MDA data for Pu-239 in feces (part 3 of 3).

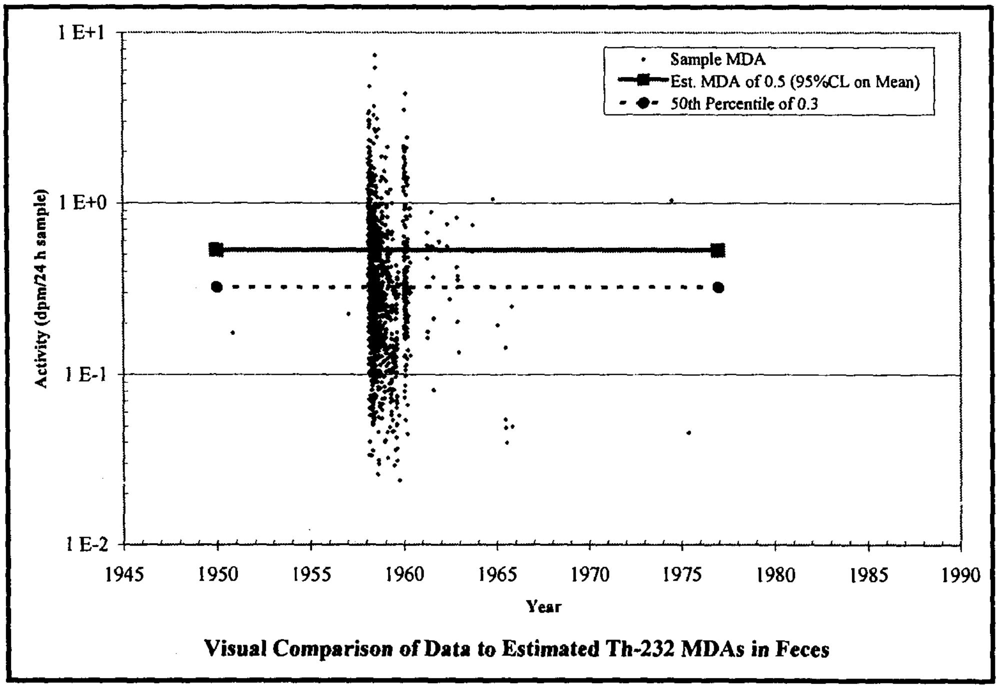

Figure A-18. MDA data for Th-232 in feces (part 1 of 3).

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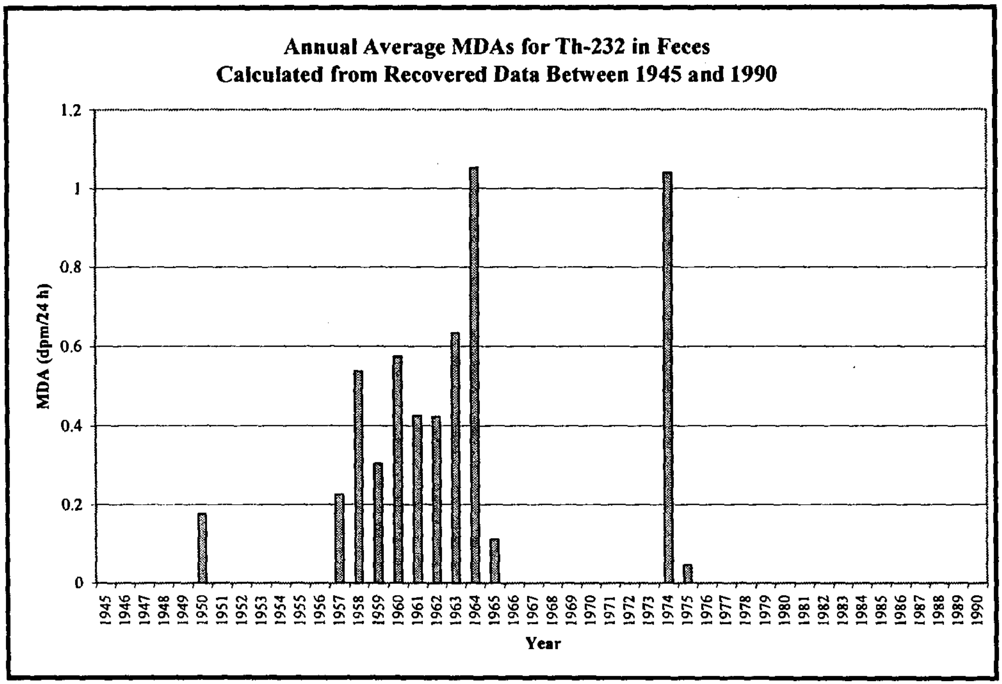

Figure A-18. MDA data for Th-232 in feces (part 2 of 3).

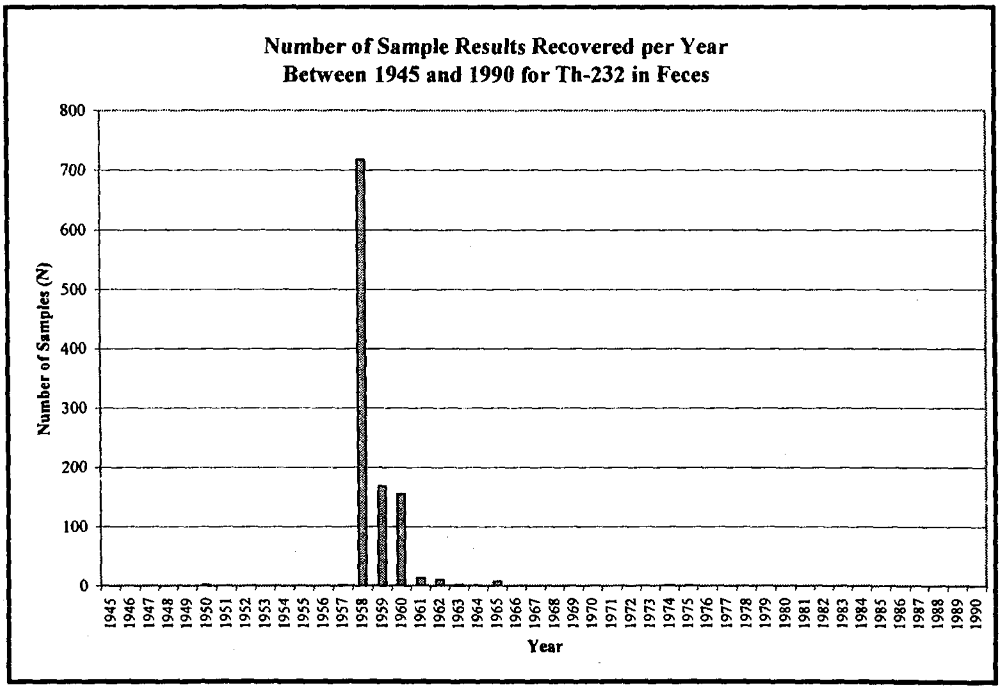

Figure A-18. MDA data for Th-232 in feces (part 3 of 3).

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

LIST OF FIGURES

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

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

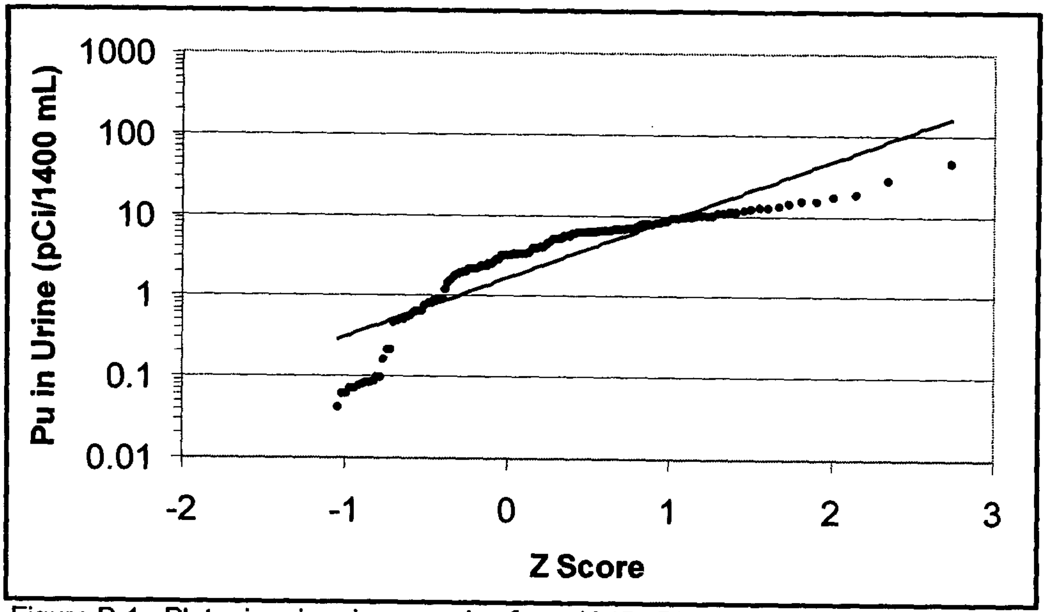

Figure B-1. Plutonium in urine samples from 1945. The geometric mean = 1.7 pCi/day, the GSD = 5.3, and r^2 = 0.79.

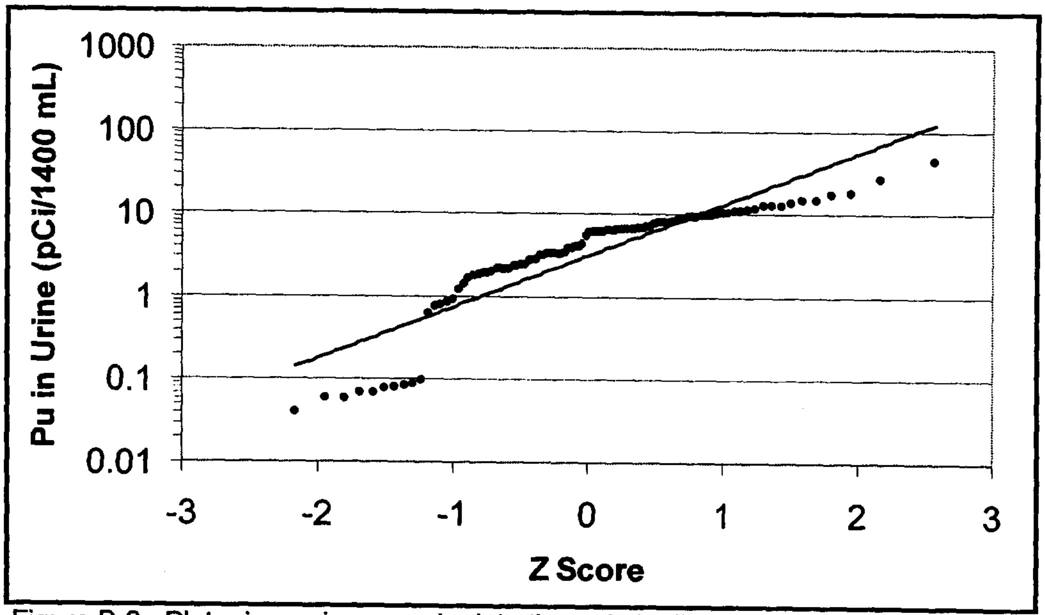

Figure B-2. Plutonium urine sample data through April 18, 1945. The geometric mean = 3.1 pCi/day, the GSD = 4.2, and r^2 = 0.82.

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

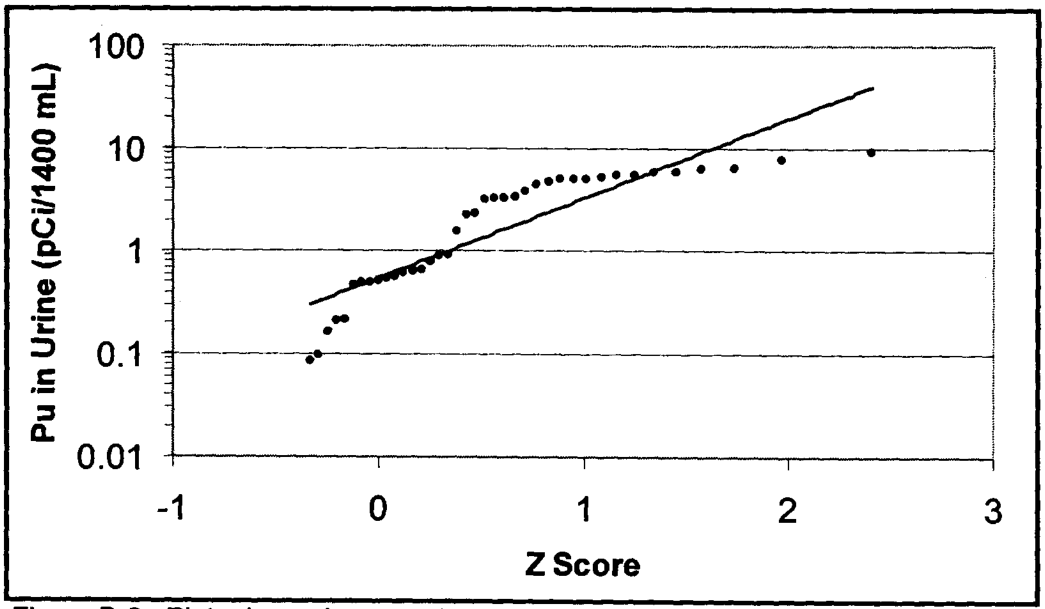

Figure B-3. Plutonium urine sample data for April 25-May 30, 1945. The geometric mean = 0.54 pCi/day, the GSD = 6.0, and r^2 = 0.80.

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.

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

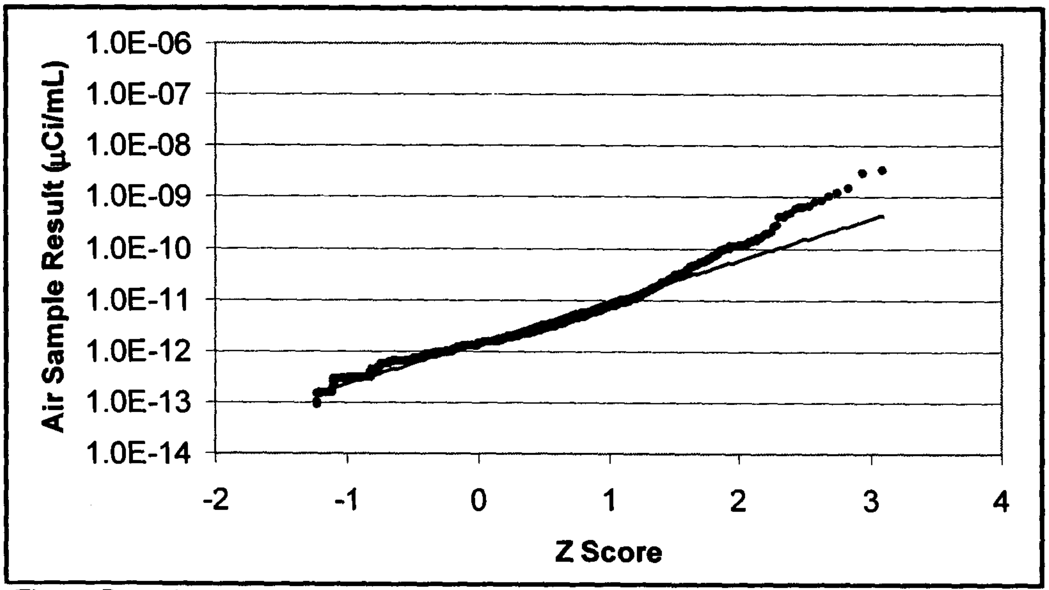

Figure B-4. Alpha air-monitoring results for 1944–1947. The geometric mean = 1.4 × 10^{-12} µCi/mL, the GSD = 6.35, and r^2 = 0.97.

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.