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Approval:	Signature on File James W. Neton, Associate Director for Science	Approval Date:	01/03/2012

☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

FOR DOCUMENTS MARKED AS A TOTAL REWRITE, REVISION, OR PAGE CHANGE, REPLACE THE PRIOR REVISION AND DISCARD / DESTROY ALL COPIES OF THE PRIOR REVISION.

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
06/22/2007	00	Approved new Site Profile for Ames Laboratory. Incorporates formal internal and NIOSH review comments. Adds Glossary and Attributions and Annotations section. There is no change to the assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Jerome B. Martin.
08/20/2008	00 PC-1	Approved page change revision made to include SEC-00075 on pages 10 and 11 in Section 1.3 where Sections 1.3.1, 1.3.2 and Table 1-2 were added. NIOSH required language was revised on pages 9 and 10 in Section 1.1. Added reference on page 74. Incorporates formal internal review comments. As a result of formal internal review, editorial changes were made on the following pages: 8 in Acronym and Abbreviations Section; 11 (Table 1-2), 25 (Table 4-1), 30 (Table 4-2), 32 (Table 4-6), 34 (Table 5-1), 36 (Table 5-2 and 5-3), 48 (Table 5-6), 51 (Table 6-1), in Sections 1.3, 4.2, 4.4, 4.7, 5.0, 5.1, 5.4, and 6.2. SEC changes occurred on pages 11 (Table 1-1), 32, 33, 37-39 (Table 5-4), 40, 41, 46 (Table 5-5), 49 (Table 5-7), 50, 57, 60, 61 (Table 6-4), 63 (Figure 6-2) in Sections 1.3, 4.7, 5.0, 5.1, 5.2, 5.3, 5.5, 6.1, and 6.3. Incorporates formal NIOSH review comments on page 2 (Publication Record), 30 (Table 4-2), 41, 48 (Table 5-6), and 74 in Sections 4.4, 5.2, 5.4, and Reference Section. References were updated on pages 10, 11, 19, 29, 30, 32, 33, 37, 40, 41, 46, 47, 50-52, 57-62, 64, 65, 70, 74-76 in Sections 1.3, 2.3, 4.4, 4.7, 5.0, 5.1, 5.2, 5.4, 6.1, 6.2, 6.3, 6.4, 7.0, and Reference Section. No sections were deleted. Training required: As determined by the Task Manager. Initiated by Karen S. Kent. Signature on File 08/12/2008 Karen S. Kent, Document Owner Signature on File 08/23/2008 John M. Byrne, Task 3 Manager Signature on File 08/12/2008 Edward F. Maher, Task 5 Manager Signature on File 08/19/2008 Kate Kimpan, Project Director Signature on File 08/20/2008 James W. Neton, Associate Director for Science
12/18/2009	01	Approved revision to add LAT chest doses for all periods, skin doses for various parts of the body, and an explanation of terminology. Sections 3.0 and 3.1 were modified as a result of these changes. Tables 3-1 through 3-6 were modified and Tables 3-7 and 3-8 were added. The definition of occupational medical dose was revised and three reference titles were changed. Wording in Section 6.2.1 about external uranium dose was modified. Revision made to add unmonitored doses prior to 1953. Section 6.2.1 reworded to remove OTIB-0004 discussion and add new methodology. Section 6.3.1 was modified to include two eras of unmonitored doses. Figure 6-2 was altered to reflect the new unmonitored dose methodology for doses prior to 1953. Table 5-1 was separated into three tables. Changes

		were made to Table 5-7 (now 5-8) to clarify and correct intake values. Wording was added to Section 1.3 to caution the Dose Reconstructor on when to label the DR as a partial DR. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite or the document. Training required: As determined by the Objective Manager. Initiated by Jackson R. Ellis.
01/14/2011	02	Revision initiated to include SEC-00166. Section 1.3.3 and Table 1-3 added. Section 2.1.6 updated to note the end of hot cell operations in the Research Building. Paragraph added in Section 5.0 to discuss SEC-00166. Section 5.1.1.3 updated with internal intake information from SEC-00166. Table 5-4 and 5-8 updated with updated intake rates. Section 5.4.1 updated to remove statement that intakes from the Research Building hot cell would be negligible. Removed A&A point. Added SEC-00166 to the reference section as NIOSH 2010. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Jackson R. Ellis.
01/03/2012	03	Revision initiated to include SEC-00185. Section 1.3.4 was added. Section 1.3 edited to update partial dose reconstruction information and to introduce new SEC. Sections 1.3.1 and 1.3.2 were edited to show inclusion of all workers in those SECs. Expansion of class was inserted in sections of the document as necessary. Section 5.4.1 edited to state environmental doses should be applied to certain Research Building workers. Battelle 2006a was updated to Battelle 2011 and Battelle 2006b was cancelled and replaced with other references. Changes were made to several sections and tables in Sections 5.1 and 6.3. Section 6.3.1.2 was updated to reduce missed dose component of beta doses in Table 6-7. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Jackson R. Ellis.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
ALRR	Ames Laboratory Research Reactor
ANL-E	Argonne National Laboratory–East
AP	anterior-posterior
AWE	atomic weapons employer
BNL	Brookhaven National Laboratory
Ci	curie
CFR	Code of Federal Regulations
cm	centimeter
d	day
D&D	decontamination and decommissioning
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ft	foot
g	gram
GSD	geometric standard deviation
HEPA	high-efficiency particulate air
hr	hour
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
INL	Idaho National Laboratory
IREP	Interactive RadioEpidemiological Program
ISU	Iowa State University
keV	kiloelectron-volt, 1,000 electron-volts
kg	kilogram
kV	kilovolt
L	liter
LAT	lateral
LBNL	Lawrence Berkeley National Laboratory
m	meter
MDA	minimum detectable activity
MDL	minimum detection level
MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
mi	mile
min	minute
mL	milliliter
mR	milliroentgen

mrad	millirad
MRD	minimum recordable dose
mrem	millirem
mrep	millirep
MTR	Materials Test Reactor
MW	megawatt
N/A	Not applicable
NCA	Nucleonic Corporation of America
NCRP	National Council on Radiological Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NTA	Eastman Kodak nuclear track emulsion, type A
NVLAP	National Voluntary Laboratory Accreditation Program
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
pCi	picocurie
POC	probability of causation
R&D	research and development
s	second
SEC	Special Exposure Cohort
SLAC	Stanford Linear Accelerator Center
SRDB Ref ID	Site Research Database Reference Identification (number)
SRS	Savannah River Site
TASF	Technical and Administrative Services Facility
TBD	technical basis document
TLD	thermoluminescent dosimeter
U.S.C.	United States Code
yr	year
β	beta
γ	gamma
µCi	microcurie
µg	microgram
µm	micrometer
§	section or sections

1.0 INTRODUCTION

1.1 PURPOSE

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 at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

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

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

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

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

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

1.2 SCOPE

This site profile provides information about U.S. Atomic Energy Commission (AEC), Energy Research and Development Administration (ERDA), and DOE operations at Ames Laboratory that pertains to radiation exposures for monitored or unmonitored workers. Section 2 provides a description of the site and operations that pertain to possible radiation exposures and discusses radiation source terms. Section 3 provides guidance for the determination of occupational medical dose. Section 4 provides guidance for the determination of dose to workers outside radiological facilities due to releases of radioactive materials to the environment. Section 5 provides guidance for the determination of intakes of radionuclides inside facilities. Section 6 provides guidance for the determination of external doses from measured doses or for periods for which records of measured doses are missing.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 7.

1.3 SPECIAL EXPOSURE COHORT

The Secretary of Health and Human Services has designated four classes of employees at Ames Laboratory for addition to the Special Exposure Cohort (SEC) authorized under EEOICPA.

NIOSH added classes to the SEC in 2006, 2007, and 2010 to cover three separate groups of employees based on work location and job description. While the classes added in 2006 and 2007 included specific workers performing specific tasks in designated buildings, the 2010 class determined that the information available about worker job description, work location, or movement about the site was insufficient to determine if an employee worked in the affected area(s). In 2011, NIOSH designated a fourth class (SEC-00185) that encompasses all previous Ames SEC periods and designates all Ames employees (including predecessor agencies, contractors, and subcontractors) as included. Additionally, SEC-00185 was revised to change the covered period's start date from January 1, 1942 to August 13, 1942, the start of the Manhattan Engineer District, known later as the Manhattan Project.

Although NIOSH cannot bound doses for certain areas and periods as described in the SECs, internal and external data that become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures) and applicable dose reconstruction methods that are defined in Sections 4, 5, and 6 of this site profile, will be used to complete partial dose reconstructions for workers who worked during the SEC periods but are not eligible for the SEC.

This site profile provides internal and external exposures that might coincide with work periods that fall within the SEC periods. There are varying types of exposures that can be applicable during the SEC periods to dose reconstructions for employees who do not qualify for the SEC(s). The periods in which doses can and cannot be reconstructed for a particular claim should be identified when writing the Dose Reconstruction Report. For dose reconstruction of claims with employment during the SEC period (see Section 1.3.4), all Ames employees are included and the dose reconstruction is a "partial reconstruction" by default. This designation should be included in the Dose Reconstruction Report.

1.3.1 August 13, 1942, through December 31, 1954

This SEC includes DOE employees or DOE contractor or subcontractor employees who worked at the Ames Laboratory in one or more of the following facilities or locations: Chemistry Annex 1 (also known as the old women's gymnasium and Little Ankeny), Chemistry Annex 2, Chemistry Building (also known as Gilman Hall), Research Building, or Metallurgical Building (now Harley Wilhelm Hall), from January 1, 1942, through December 31, 1954, for a number of workdays that total at least 250 workdays, or in combination with workdays within the parameters (excluding aggregate workday requirements) established for one or more classes of employees in the SEC, and who were monitored

or should have been monitored. As stated above and defined in Section 1.3.4, this SEC class has effectively been expanded to include all Ames employees, contractors, and subcontractors who worked in any area of the Ames Laboratory at Iowa State University, and the start date for this SEC was changed to August 13, 1942.

NIOSH evaluated the feasibility for completing dose reconstructions for employees at Ames Laboratory from August 13, 1942, through December 31, 1954, and found that the monitoring records, process descriptions, and source term data available are not sufficient to perform complete dose reconstructions for the SEC (SEC-00038) class of employees (NIOSH 2006). Table 1-1 summarizes the results of the feasibility findings for each exposure source for the period from August 13, 1942, through December 31, 1954.

Table 1-1. Feasibility findings for SEC-00038, August 13, 1942, through December 31, 1954. (NIOSH 2006, Table 7-1).

Source of exposure	Dose reconstruction is feasible	Dose reconstruction is NOT feasible
Internal dose:		
Uranium	X	
Thorium/plutonium		X
Thoron		X
External dose:		
Uranium beta/gamma	X	
Thorium/plutonium beta/gamma		X (except 1953 and 1954)
Neutron		X
Occupational medical X-ray	X	

1.3.2 January 1, 1955, through December 31, 1970

This SEC includes sheet metal workers, physical plant maintenance and associated support staff (includes all maintenance shop personnel), and supervisory staff who were monitored, or should have been monitored, for potential internal radiation exposures associated with the maintenance and renovation activities of the thorium production areas in Wilhelm Hall (a.k.a. the Metallurgy Building or "Old" Metallurgy Building) at the Ames Laboratory, for the period from January 1, 1955, through December 31, 1970, and who were employed for a number of workdays aggregating at least 250 workdays, either solely under this employment or in combination with workdays within the parameters (excluding aggregate workday parameters) established for other classes of employees included in the SEC. As stated above and defined in Section 1.3.4, this SEC class has effectively been expanded to include all Ames employees, contractors, and subcontractors who worked in any area of the Ames Laboratory at Iowa State University.

NIOSH evaluated the feasibility for completing dose reconstructions for employees at Ames Laboratory from January 1, 1955, through December 31, 1970. NIOSH found that the monitoring records, process descriptions, and source term data available are not sufficient to perform complete dose reconstructions for the SEC (SEC-00075) class of employees (NIOSH 2007a). Table 1-2 summarizes the results of the feasibility findings for each exposure source for the period from January 1, 1955, through December 31, 1970.

Table 1-2. Feasibility findings for SEC-00075, January 1, 1955, through December 31, 1970. (NIOSH 2007a, Table 7-1).

Source of exposure	Dose reconstruction is feasible	Dose reconstruction is NOT feasible
Internal		
Th-232 and progeny		X
Ambient environmental	X	
External		

Gamma	X	
Beta	X	
Neutron	N/A	
Ambient environmental	X	
Occupational medical X-ray	X	

1.3.3 January 1, 1955, through December 31, 1960

This SEC includes all employees of DOE, its predecessor agencies, and its contractors and subcontractors who worked in any area of DOE facilities on the Ames Laboratory Campus from January 1, 1955, through December 31, 1960, for a number of workdays aggregating at least 250 workdays, occurring either solely under this employment, or in a combination with workdays within the parameters established for one or more other classes of employees in the SEC.

NIOSH evaluated the feasibility for completing dose reconstructions for employees at Ames Laboratory from January 1, 1955, through December 31, 1960, and found that the monitoring records, process descriptions, and source term data available are not sufficient to perform complete dose reconstructions for the SEC (SEC-00166) class of employees (NIOSH 2010b). Table 1-3 summarizes the results of the feasibility findings for each exposure source for the period from January 1, 1955, through December 31, 1960.

Table 1-3. Feasibility findings for SEC-00166, January 1, 1955, through December 31, 1960. (NIOSH 2010b, Table 7-5).

Source of exposure	Dose reconstruction is feasible	Dose reconstruction is NOT feasible
Internal dose:		
Uranium and progeny	X	
Thorium and progeny	X	
Other radionuclides (Research Building)		X
External dose:		
Gamma	X	
Beta	X	
Neutron	X	
Occupational medical X-ray	X	

Although NIOSH found that it is not possible to reconstruct radiation doses completely for all of the evaluated class, it intends to use any internal and external monitoring data that might become available for an individual claim (and that can be interpreted using existing dose reconstruction processes or procedures). Therefore, partial dose reconstructions for individuals employed at Ames Laboratory from January 1, 1955 through December 31, 1960, but who do not qualify for inclusion in the SEC, can be performed using these data as appropriate.

1.3.4 Class Recommended by NIOSH for Addition to the SEC: August 13, 1942, through December 31, 1970

This SEC includes all employees of DOE, its predecessor agencies, and its contractors and subcontractors who worked in any area of DOE facilities on the Ames Laboratory at Iowa State University during the period from August 13, 1942, through December 31, 1970, for a number of workdays aggregating at least 250 workdays, occurring either solely under this employment, or in a combination with workdays within the parameters established for one or more other classes of employees in the SEC.

This SEC, SEC-00185 (NIOSH 2011), does not make any new information available for the feasibility of performing dose reconstructions between the dates of August 13, 1942, and December 31, 1970.

NIOSH previously designated the three classes associated with Sections 1.3.1, 1.3.2, and 1.3.3. NIOSH has determined that site-specific and claimant-specific data available for Ames Laboratory for this entire period are insufficient to enable it to determine that a specific work group was not potentially exposed to radioactive material releases or possible subsequent contamination. Based on this information, NIOSH has determined that the previously proposed SEC class definitions cannot be based on or limited to job titles or duties.

The guidance (and limitations) for performing dose reconstructions using the information in this site profile have not changed. However, the class definitions for the SECs in Sections 1.3.1 and 1.3.2 have effectively been expanded to include all Ames employees, contractors, and subcontractors who worked in any area of the Ames Laboratory at Iowa State University.

2.0 SITE DESCRIPTION

The Ames Laboratory site consists of a number of buildings at Iowa State University (ISU) in Ames, Iowa. The precursor to the Ames Laboratory was the Ames Project, which was established in 1942 in a contract between the Metallurgical Laboratory at the University of Chicago and Iowa State College (Fulmer 1947). Ames Laboratory was established by the AEC in May 1947 (Karsjen 2003). The Ames Project/Laboratory played a key role in the production of strategic nuclear materials for the Manhattan Project and the AEC.

Early in 1942, prior to the beginning of the Manhattan Project, the most pressing problem was the preparation of large amounts of pure uranium metal (Ames 1960a). Faculty members in the Chemistry Department at Iowa State College with expertise in rare earth metallurgy were called on to develop a method to purify uranium and reduce its cost of production (Ames 1960a). By November 1942, successful methods had been developed and approximately one-third of the uranium used in the Chicago pile was supplied by the Ames Project (Karsjen 2003). The Ames Project was asked to turn its process over to industry and, in the meantime, to produce as much pure uranium as possible. Between mid-1942 and August 1945, more than 1,000 tons of pure uranium metal was supplied to the Manhattan Project (Ames 1960a).

Once the potential need for thorium metal was recognized, the Ames Project began to develop methods for purifying thorium in 1943. By late 1944, a large-scale process for thorium metal production was developed; between 1950 and April 1953, when thorium production was turned over to industry, more than 65 tons of pure thorium metal and thorium compounds were produced by the Ames Laboratory (Ames 1960a).

In addition to the early uranium and thorium metal production operations, personnel at Ames Laboratory handled a number of other radionuclides and operated an 80-MeV synchrotron, a 5-MW research reactor, and several radiation-generating machines. Each of these radiation sources is described in the following sections.

2.1 FACILITIES

The original buildings at Iowa State College that were used for the Ames Project included Physical Chemistry Annex 1, the Chemistry Building (now Gillman Hall), and the Physics Building (now Physics Hall) (Ames 1967). In 1944, a new building, Physical Chemistry Annex 2, was constructed to house operations to recover uranium from scrap material (Fulmer 1947).

In November 1945, the buildings used by the Ames Project were designated as the Institute for Atomic Research. Four additional buildings were constructed: the Metallurgy Building (now Wilhelm Hall), the Research Building (now Spedding Hall), the Office and Laboratory Building that connected the Chemistry and Physics Buildings, and the Synchrotron Building (now the Spangler Geotechnical Laboratory) (Ames 1962a). The Metals Development Building was added in 1960 (Ames 1960a). In May 1947, the AEC established a major research facility at Ames, known as Ames Laboratory (Ames 1962a), which operated as an integral part of the Institute for Atomic Research. Construction of a 5-MW heavy-water-moderated research reactor began in 1962, and operations began in 1965. The Technical and Administrative Services Facility (TASF) was added to connect the Research Building (Spedding Hall) and the Chemistry Building (Gilman Hall); the TASF includes only offices. The locations of Ames Laboratory buildings on the ISU campus are shown in Figure 2-1. The locations of the Reactor facilities (now known as the Applied Science Complex) and the Synchrotron Building are shown in Figure 2-2.

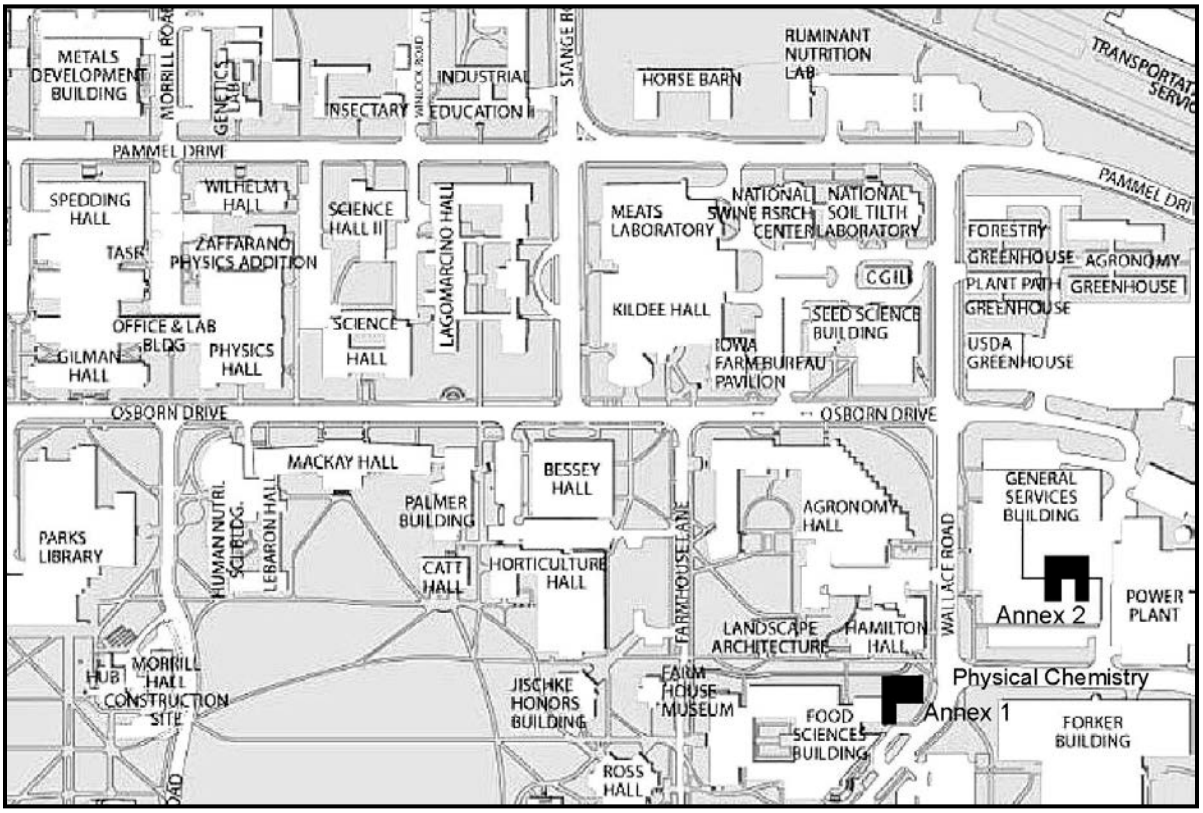


Figure 2-1. ISU campus map showing locations of Ames Laboratory buildings. The former sites of Physical Chemistry Annex 1 and Annex 2 are also shown. Source: ISU (2006).

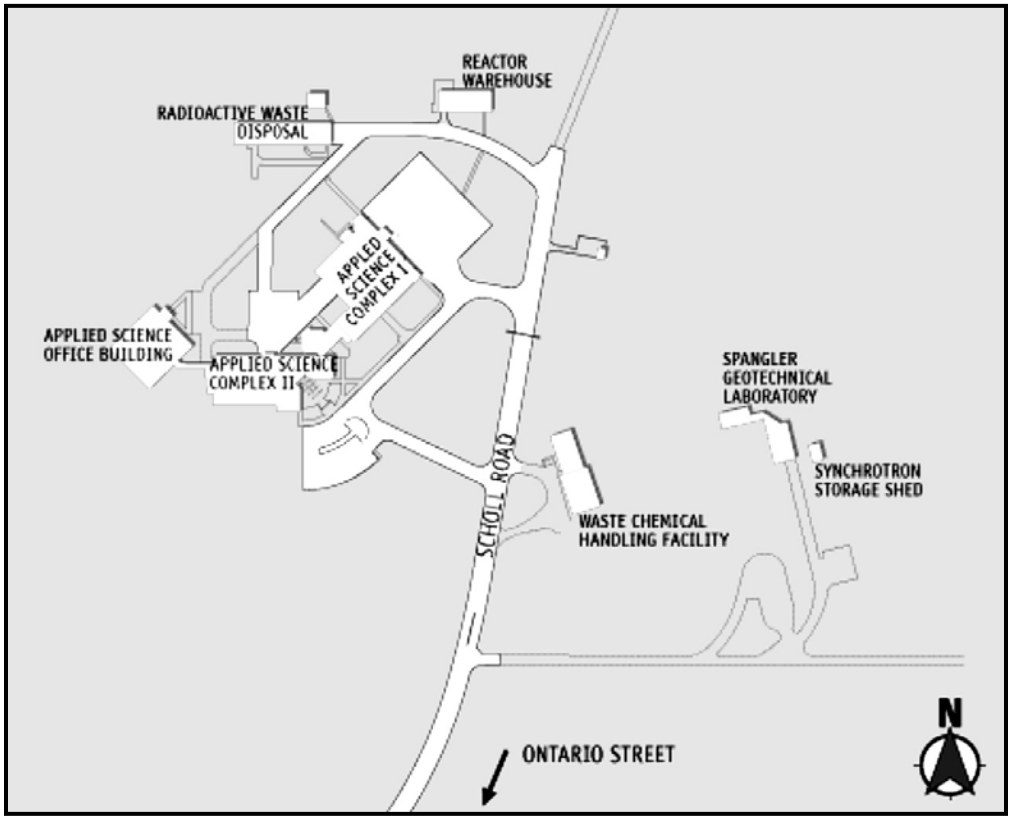


Figure 2-2. ISU campus map showing locations of reactor facilities and Synchrotron Building. Source: ISU (2006).

To avoid confusion between the historical building names and the current names, as shown in Figures 2-1 and 2-2, both names are used in the following sections with the historical name first and the current name in parentheses, where applicable.

2.1.1 Physical Chemistry Annex 1

The production of uranium metal was conducted in Physical Chemistry Annex 1, which was an old wooden structure east of the Dairy Industries building and west of Wallace Road. Uranium operations began there in mid-1942 and ended on August 5, 1945, when the uranium purification process was transferred to industry. More than 1,000 tons of pure uranium and more than 300 tons of uranium scrap were produced during this period (Karsjen 2003). In 1943, an open porch area was enclosed (to control dusty operations) and additions were constructed to accommodate increases in uranium production (Payne 1992). Beginning in 1943, the building was also used to produce thorium metal until the processing equipment was transferred to the new Metallurgy Building in 1949 or 1950 (Ames 1960a). Physical Chemistry Annex 1 was torn down in 1953 (Karsjen 2003). The building site was decontaminated, surveyed in May 1976, and designated acceptable for future construction (Voss 1979).

2.1.2 Chemistry Building (Gilman Hall)

The initial Ames Project work was conducted in the Chemistry Building in early 1942. The process for purifying uranium metal and the methods and equipment to increase production were developed in this building. Uranium production operations were moved to Physical Chemistry Annex 1 in mid-1942, while other uranium research continued in the Chemistry Building, including determination of uranium properties, studies of uranium corrosion, development of protective coatings for uranium, and development of uranium alloys and compounds (Fulmer 1947). Other research in the Chemistry Building involved development of pure thorium metal, thorium alloys and compounds, yttrium metal, cerium metal, and beryllium metal (Ames 1962a). Analytical work centered on plutonium chemistry and the radiochemistry of the separation of fission products from uranium and plutonium, which was conducted in the "hot laboratory" between 1942 and 1951 (Ames 1960a). The Chemistry Building was decontaminated and surveyed in May 1976 (Voss 1979).

2.1.3 Physics Building (Physics Hall)

Research and development (R&D) to support other work at Ames Laboratory was conducted in the Physics Building. Analytical equipment was developed, including three beta-ray spectrometers, a bent crystal X-ray spectrometer, a kevatron, X-ray and neutron diffraction spectrometers, scintillation and conduction crystal spectrometers, X-ray and electron diffraction machines, and an electron microscope (Ames 1951, 1962a). Nuclear fission was studied to identify the individual fission fragments, the energies involved, and the ionized state of the emitted particles (Ames 1951). Research was conducted to determine the stopping power and shielding properties of various solid materials (Ames 1951). Personnel of the Physics Department operated the 80-MeV synchrotron, which is described below (Ames 1962a).

2.1.4 Physical Chemistry Annex 2

Physical Chemistry Annex 2 was a brick fireproof structure built east of Wallace Road in early 1944 to house the recovery of uranium from scrap uranium metal turnings collected from other Manhattan Project sites. Operations in this building through December 1945 produced more than 300 tons of recovered uranium metal (Fulmer 1947). Operations ended in 1953 when the building was converted to a plumbing shop; it was razed in 1972 (Ames 1985). The area where the building stood was covered with concrete and served as a parking lot and loading zone. The area was surveyed in May 1979 (Voss 1979). Part of the General Services Building now covers part of the former building site (see Figure 2-1).

2.1.5 Metallurgy Building (Wilhelm Hall)

The Metallurgy Building, constructed by the AEC, was completed in October 1949 (Ames 1951). The building housed research directed toward the development of special metals and alloys used in nuclear energy projects. Zircaloy was initially developed at Ames Laboratory as part of a basic study of the zirconium-tin alloy phase diagram (Ames 1962a). The subject of reactor coolants was studied, as were heat-transfer properties of various metals and alloys (Ames 1951). Equipment available for research, development, and production in metallurgy included many types of furnaces; high-vacuum systems; pyrometric devices; fabricating and testing machines; metallographs; X-ray diffractometers; and ultrasonic, spectrographic, dilatometric, and other instruments for examination and study of metals and alloys. A glovebox line in the Metallurgy Building was used to study the behavior of plutonium in molten metal systems (Ames 1962a).

Thorium production and research activities were moved from Physical Chemistry Annex 1 to the Metallurgy Building in 1949 and work on thorium continued until 1953. Poor contamination control practices and poor ventilation contributed to contamination of the building. However, contamination levels have been reduced by mitigation, decontamination, remodeling, and renovation projects. Contamination still exists in many interspatial areas of the building and in some relatively inaccessible areas (Hokel et al. 1998).

2.1.6 Research Building (Spedding Hall)

The Research Building was constructed by the AEC and occupied in early 1951 (Ames 1951). Many metals, including rare earths, were investigated for mechanical, chemical, electrical, and other properties, and were studied by experimental techniques that probed the inner structures and forces of the materials (Ames 1962a). Research facilities in the building included a 150-kV accelerator that produced 14-MeV neutrons; a glovebox line for radiochemistry experiments; a hot canyon and hot cell with steel shielding, lead glass windows, and manipulators for work with highly radioactive materials; and an electron microprobe analyzer. The hot canyon was two stories high with the lower level in the basement adjacent to the hot cell. Research activities included electron beam welding; the study of the electronic structure of metals; and the separation, preparation, and measurement of properties of rare earth metals. The initial research on liquid metal coolants was done at Ames Laboratory in an engineering sodium test loop used in corrosion, fluid-flow, and heat transfer studies with liquid sodium (Ames 1967). Work in the hot cell continued until 1982.

2.1.7 Office and Laboratory Building

The Office and Laboratory Building connects the Chemistry and Physics Buildings and provides administrative offices of the Ames Laboratory, the special research laboratories used jointly by chemists and physicists, and a large physical sciences reading room (Ames 1951).

2.1.8 Synchrotron Building (Spangler Geotechnical Laboratory)

The Synchrotron Building was constructed in 1949 on a 200-acre tract northwest of the campus that was set aside for special use by the Institute for Atomic Research (Ames 1962a). The synchrotron room housed two electron accelerators that could project electrons up to 80 MeV onto a target, which produced high-energy gamma rays that interacted with nuclei to release neutrons, protons, and alpha particles (Ames 1967). The accelerators were operated from a Control Room where there was a safety gate that prohibited access to the synchrotron room when the beam was on (Ames 1967). In many cases, the products of these reactions were radioactive, and were used in research in nuclear physics and radiochemistry (Ames 1967). In addition, the synchrotron was used to probe nuclear structures and to provide radioisotopes for nuclear spectroscopy (Ames 1962a). Operations at the Synchrotron Building ended in June 1971; the equipment was decommissioned in the early 1990s.

2.1.9 Metals Development Building

Ames Laboratory facilities were expanded in 1960 to include the Metals Development Building. Its missions were to conduct process development research on larger-than-laboratory scale and to evaluate commercial feasibility of the processes developed. One major process was the production of very pure metals, and the production of small quantities of these metals to be used elsewhere as standards. The building contained a complete pilot plant with facilities for each step of the metal production process from ore treatment to metal fabrication or analysis (Ames 1960a). Equipment in the building included electron microscopes, an electron microprobe, metallography apparatus, liquid-liquid extraction apparatus, extrusion presses for producing rods and tubes, and rolling machines for making sheet metal (Ames 1967).

2.1.10 Ames Laboratory Research Reactor (Applied Science Complex)

Construction of a 5-MW, heavy-water-moderated research reactor began in 1961; the reactor was first operated in February 1965 (Ames 1967; Voigt 1981). The reactor and its support facilities were about 1.5 mi northwest of the ISU campus on a 200-acre site used by the Institute for Atomic Research (Ames 1967). The reactor fuel was 93% enriched ^{235}U contained in 24 fuel assemblies in a hexagonal arrangement in a core 30 in. across and 25 in. high (Voigt 1981). The reactor shielding was an irregular decahedral prism shape with a thermal column on one face and nine faces with beam tubes from which radiation beams (primarily neutrons) could be extracted and directed to experimental areas surrounding the reactor (Ames 1967; Voigt 1981). Other tubes and thimbles provided access to the reactor core for irradiation experiments. Research activities included radiation damage studies, determination of the crystalline structure of solids, determination of mechanical properties of reactor materials, and analysis of the decay products of nuclear fission. The experimental equipment featured an online isotope separator that received fission products directly from the operating reactor, separated them by weight, and analyzed them by isotope. Other research equipment included a neutron diffractometer used to determine the physical properties of solids and a hot cell for handling spent reactor fuel (Ames 1967). Operation of the reactor resulted in airborne tritium concentrations in occupied spaces of the building (Voigt 1981). A routine tritium bioassay program was part of the radiation safety program at the reactor (Voss 1971). Operation of the reactor ended in December 1977 and decontamination and decommissioning (D&D) of the facilities was completed in 1981. At the time operations ended, the heavy-water coolant contained approximately 1.7 Ci of tritium per liter (Voigt 1981).

2.2 OPERATIONS

Two major operations at Ames Laboratory resulted in radiation exposure to the staff – the production of large quantities of pure uranium (1942 to 1945) and thorium metal (1943 to 1953). A number of smaller operations contributed to staff exposure to radiation.

2.2.1 Uranium Metal Operations

The initial Ames process for production of uranium metal was based on the chemical reduction of uranium tetrafluoride (UF_4) by calcium metal. Finely ground UF_4 was mixed with granulated calcium metal and the mixture was poured into a refractory-lined container. A fuse wire buried in the charge was electrically heated to initiate the reaction, which continued until both uranium metal and calcium fluoride were in the molten state. The more dense uranium collected at the bottom of the container, where it was allowed to cool to room temperature, after which it was removed for casting. The uranium metal was cast by placing it in a graphite crucible, heating it in a vacuum, and allowing the liquid metal to flow into a graphite mold for specific shapes (Fulmer 1947). Although more complex, the uranium production process was improved by replacing the calcium reagent with magnesium metal (Fulmer 1947).

Most of the uranium metal production operations were conducted in Physical Chemistry Annex 1, which had poor contamination control and poor ventilation (Friedell 1942). Workers in this building were likely to have received intakes of uranium between 1942 and 1945 and thorium between 1943 and 1950 as well as external exposures to beta and gamma radiation. These exposures are estimated in Sections 5 and 6. Effluents from the building were not monitored and local environmental contamination by uranium compounds was likely. Uranium and thorium contamination of the ground surface around Annex 1 required removal and offsite disposal of the contaminated soil (Payne 1992). No other records could be found that addressed the disposal of contaminated waste or the control and monitoring of air and liquid effluents.

A substantial amount of uranium scrap material was produced, which was processed in Physical Chemistry Annex 2 from early 1944 to December 1945 (Fulmer 1947). Radiation exposures to workers in this building during this period were similar to the uranium exposures in Physical Chemistry Annex 1, and are estimated in Sections 5 and 6. Local environmental contamination around Annex 2 was assumed to be similar to that around Annex 1, and this is addressed in Section 4.1.

2.2.2 Thorium Metal Operations

The uranium metal production method was adjusted between August 1943 and August 1944 to produce thorium metal (Fulmer 1947). Thorium tetrafluoride (ThF_4) was mixed with calcium metal with a zinc chloride booster to produce a thorium-zinc alloy with a 96% yield of thorium metal. The alloy was heated under vacuum in a graphite crucible to distill off the zinc. Casting of thorium metal was difficult because of its high melting point and its reactive properties. Beryllium oxide crucibles had to be used; melting the thorium in a crucible the size and shape of the desired ingot proved to be a more reliable method of casting, although it often did not separate well from the slag and oxide. Castings were improved in late 1946 by pouring molten thorium into graphite molds (Fulmer 1947). Production of thorium metal and thorium compounds continued until April 1953, when thorium production operations were turned over to industry (Ames 1960a).

The purified feed material for the thorium production operation was prepared by dissolving thorium nitrate in nitric and oxalic acids, precipitating the thorium oxalate, drying the precipitate in trays, hydrofluorinating the precipitate to ThF_4, and crushing the ThF_4 to a fine powder (Fulmer 1947). This process was used throughout thorium production operations.

Thorium production operations, which were conducted in Physical Chemistry Annex 1 after uranium operations ended in 1945, continued until 1949 or 1950, when the operation and equipment were moved to the new Metallurgy Building (Wilhelm Hall) (Ames 1960a). Workers in Annex 1 were likely to have intakes of thorium and external exposures to beta and gamma radiation. These exposures are estimated in Sections 5 and 6. Effluents from the building were not monitored and local environmental contamination by thorium compounds was likely. Thorium production operations in the Metallurgy Building improved with better ventilation, but personnel exposures and environmental contamination continued to be unquantified until 1953 because of a lack of monitoring and inadequate records.

A summary of AEC activities at Ames Laboratory during the period from 1942 through 1954 is presented in Table 2-1.

Table 2-1. Timeline of AEC activities at Ames Laboratory, 1942 through 1954.

Operation	Dates[a]	Building	Activity
Uranium metal production	Feb 1942–Aug 1942	Chemistry (Gilman Hall)	Process development
Uranium metal production	Aug 1942–Dec 1942	Chemistry	Production
Uranium metal production	Sept 1942–Aug 1945	Annex 1	Production
Uranium scrap recovery	Late 1943–early 1944	Chemistry	Process development
Uranium scrap recovery	Early 1944–Dec 1953	Annex 2	Production

Uranium metal casting	Sept 1942–Aug 1945	Annex 1	Production
Thorium metal production	Aug 1943–Jun 1946	Annex 1	Process development
Thorium metal production	Jun 1946–late 1949	Annex 1	Production
Thorium metal production	Early 1950–Apr 1953	Metallurgy (Wilhelm Hall)	Production
Studies of plutonium properties	Jun 1943–Dec 1947	Chemistry & Metallurgy	Research
Plutonium/fission product separations	Summer 1943–Dec 1954	Chemistry & Research (Spedding Hall)	Research and hot cell work
Thorium metal casting	Jun 1946–late 1949	Annex 1	Production
Thorium metal casting	Early 1950–Apr 1953	Metallurgy	Production
Studies of uranium and thorium properties	Early 1942–Dec 1954	Chemistry	Research
Development of analytical procedures	Early 1943–Dec 1954	Chemistry	Research
Annex 1 demolition	1953	Annex 1	Demolition

a. Exact start-end dates vary depending on the reference. Listed dates are the most common or consensus.

2.2.3 Other Operations

Methods for routine analysis of fission products were developed at Ames Laboratory. These activities resulted in the discovery of the previously unidentified isotopes ^{33}P, ^{144}Pm, ^{125}Sb, and five isotopes of ruthenium and rhodium. Research on the parent-daughter relationship of $^{90}Sr/^{90}Y$ was conducted. Pioneering research in applications of alpha and beta spectroscopy and mass spectroscopy to identify specific radionuclides was part of the development of laboratory methods. A process for separating ^{233}U from thorium was developed.

2.3 SOURCE TERMS AND PERSONNEL MONITORING

Ames Laboratory staff members were exposed to a number of radiation sources, including environmental effluents, radioactive intakes, and external radiation dose.

Environmental effluents from the early Ames Laboratory buildings were unmonitored and uncontrolled. Radioactive effluents near these buildings could have exposed workers to unmonitored occupational environmental doses. An accidental release of thorium waste materials to the sanitary sewer system occurred in 1951 (Voss 1979). This release, which could have contributed to the occupational environmental dose received by workers who were involved in the incident response, is addressed in Section 4.

Uranium and thorium metal production involved several dusty operations that resulted in work-area contamination and potential worker inhalation and ingestion. The principal sources of surface contamination and airborne dust were the processes of grinding uranium fluoride into a fine powder, transferring the powder from the grinder, and mixing and loading the powder charge into the reduction crucibles. Because the uranium had been separated from radium and its decay products, ^{222}Rn was not a potential inhalation concern. The principal sources of thorium surface contamination and airborne dust were the processes of preparing and drying the fine powder and mixing and loading the powder charge into the reduction crucibles (Fulmer 1947).

There were frequent small explosions and fires associated with the uranium and thorium production operations. Payne (1992) cited as many as six small fires in a single day; these fires contributed to work-area contamination and potential airborne radioactive material exposures. No records were found to indicate that air sampling or contamination control was associated with these fires.

Personnel protection for potential dusty operations included the use of Laboratory-provided clothing and gloves, as well as restrictions on eating and smoking in areas where radioactive materials were handled. Although respiratory protection measures, such as gas masks and dust masks, were provided, their use was not enforced before about 1952. Showers were recommended for workers at

the end of each day, but records indicated that not all workers complied with this recommendation (Klevin 1952; Payne 1992).

Many other radioactive materials were handled in Ames Laboratory buildings before routine monitoring for radiation exposures began in about 1952. Because the bioassay program was minimal and records are sparse, intakes that might have occurred before 1953 were estimated. When reactor operations started in 1965, a routine tritium bioassay program was started, and tritium intakes can be reliably estimated. Occupational internal dose is addressed in Section 5.

The uranium and thorium metal production operations resulted in beta and gamma radiation exposures to workers. Beta radiation was the dominant external source of radiation associated with unshielded sources of uranium, such as uranium metals production, scrap recovery, and machining processes. The significance of beta emissions from thorium depends on the state of equilibrium with the ^{232}Th parent, which is a factor of the time elapsed since the thorium process feed material was separated (NIOSH 2006). Photon exposure rates as high as 22 mR/hr were reported for a thorium storage area, suggesting that this raw material for the thorium production process was not newly separated (Klevin 1952).

Only two film badge results were identified for 1944 with results in units of “average roentgens/8 hour day during week” (Tybout 1944). Before 1952, only pencil dosimeters were used in Ames Laboratory facilities and the records are sparse. The use of film badges began in late 1952; records from 1953 and 1954, along with workplace measurements, are used to estimate earlier radiation doses. Records of external radiation exposures received from 1955 to the present are used in Section 6 to evaluate occupational external dose.

3.0 OCCUPATIONAL MEDICAL DOSE

Occupational medical exposures are included in dose reconstruction only for medical examinations obtained at covered facilities (ORAUT 2011b). The X-ray equipment for occupational medical examinations of Ames Laboratory staff members was located at the Iowa State Student Health Center/College Hospital (Voss 1957a), which is not a covered facility under EEOICPA. Therefore, no occupational medical doses are included.

4.0 OCCUPATIONAL ENVIRONMENTAL DOSE – INTRODUCTION

Occupational environmental dose refers to the dose received by workers on the site but outside facilities (e.g., buildings). These doses can be internal or external depending on the characteristics of the individual radionuclides. Radionuclides at Ames Laboratory included uranium, plutonium, thorium, and small amounts of others used in the R&D program (Fulmer 1947). Tritium, argon, and krypton were released at the Ames Laboratory Research Reactor (ALRR). While most radionuclides when inhaled would give a dose to particular organs in the body, tritium gas would give a dose to the whole body (Voigt 1973). These radionuclides are addressed in the following sections.

Occupational environmental dose was not measured (direct radiation dosimeters) until 1953, when workers were badged (Martin 2006a,b), and it was not calculated from environmental media concentrations until 1962. Sources of potential environmental exposures (releases to the environment) were not measured until 1962 (Voss 1963).

Different activities were carried out during distinct periods of Ames Laboratory history. Occupational environmental doses are, therefore, addressed below for each of these periods and their activities. A significant release of radioactive materials to the environment from Ames Laboratory facilities is addressed in Section 4.5.

4.1 URANIUM/THORIUM PRODUCTION PERIOD, 1942 TO 1953

Uranium production occurred in Physical Chemistry Annex 1 (see Figure 2-1) from mid-1942 through August 5, 1945 (Karsjen 2003). Uranium scrap recovery occurred in Physical Chemistry Annex 2 (see Figure 2-1) from early 1944 through December 1945, and some operations continued until 1954. Thorium production occurred first in Physical Chemistry Annex 1 from 1944 through 1949, when the operation moved to the new Metallurgical Building (Wilhelm Hall; see Figure 2-1). The operation in the Metallurgical Building continued until April 1953. Like Annex 1 and Annex 2, there was no on- or offsite designation for this facility. Workers, students, and college personnel moved freely by and around the building. No measurements were made of the particulate or gaseous effluents from the buildings or of the radiation levels outside the buildings (Ames 1963).

Workers conducted their work inside the facilities and there were no specific work assignments outside the facilities; that is, input materials arrived at the facilities and were processed, and products were shipped from the facilities. Workers did not move between the facilities on campus and there was no transportation of materials among the facilities on campus (Ames 1962a).

There were no documents found that stated the room and hood ventilation stacks on the facilities had filters. Concentrations of uranium dust were measured in the operation rooms (Voss 1978). To estimate a bounding dose outside the facilities, it was assumed that losses of 0.1% of the uranium or thorium as dust in a facility were emitted continuously and dispersed from ground level in accordance with local and regional meteorological conditions (see Figure 4-1) (Voss 1981) and a standard Gaussian atmospheric dispersion computer model (Napier et al. 2004). When resuspension is included, the daily intake rate of uranium (modeled as ^{234}U, type M or S) was 5 pCi. This intake applies outside Physical Chemistry Annex 1 for August 1942 through August 1945 and outside Physical Chemistry Annex 2 for 1944 through 1954. In addition, daily intakes of ^{232}Th, ^{228}Th, and ^{228}Ra at 0.07 pCi each apply to Annex 1 for June 1946 through 1949 and to the Metallurgy Building for 1950 through April 1953. These are upper bound intakes so the distribution is constant (Napier 2006a).

4.2 SYNCHROTRON OPERATIONS PERIOD, 1949 TO 1971

Synchrotron operations occurred in what is now called the Spangler Geotechnical Laboratory from 1949 through June 1971 (see Figure 2-2). Unlike the uranium and thorium facilities, this facility was

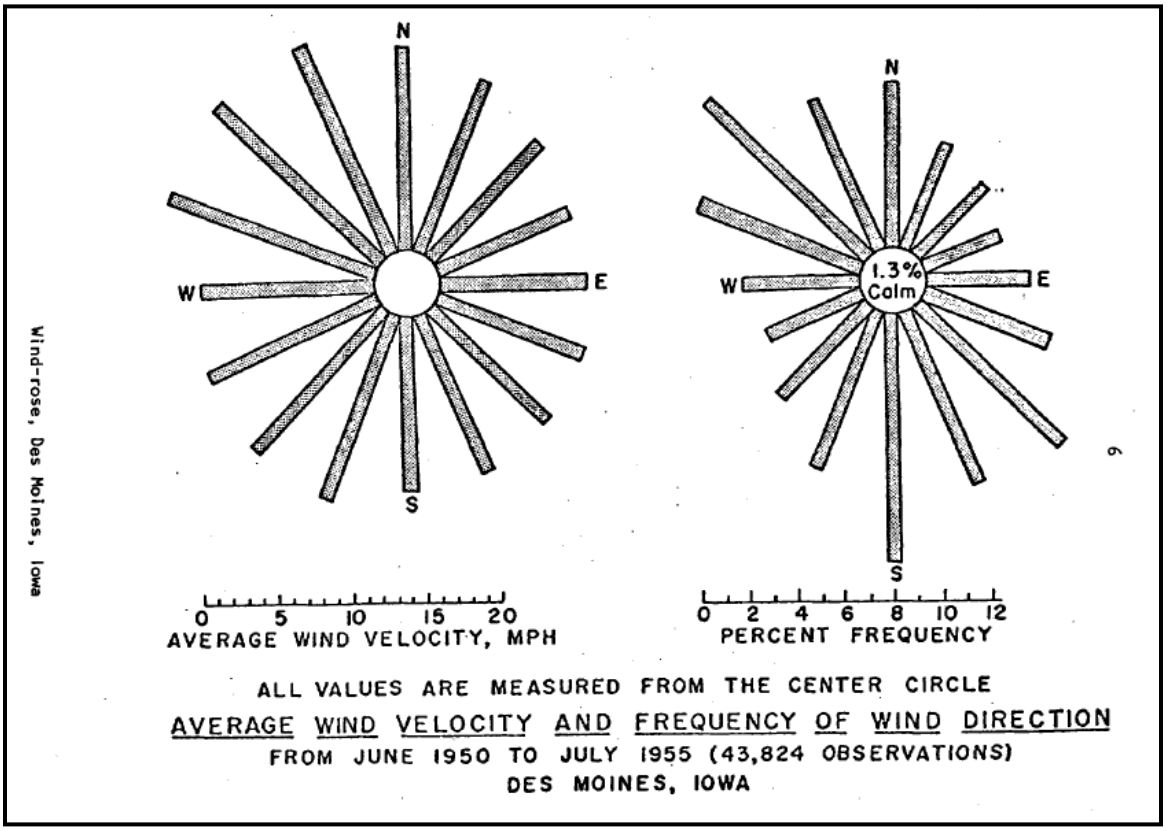


Figure 4-1. Wind rose for Des Moines, Iowa, 1950 to 1955. Source: Voss (1981).

fenced, providing a defined exclusion area outside the building. Dosimeters were not provided to anyone at the facility until late 1952 (Martin 2006c). There were small research amounts of radioactive materials that were generated during operations and negligible particulate or gaseous effluents were released from the building [1]. No routine measurements of direct gamma or neutron radiation were made outside the buildings, but a detailed survey of the facility, including fenceline gamma dose rates, was made on May 16, 1961, during a special synchrotron operation. The results of that survey are listed in Table 4-1 (Ames 1961a).

The fenceline gamma dose rates measured in the radiation survey (Ames 1961a) were worst case and were produced with the maximum beam current on a target and direction that would produce maximum dose rates at the fenceline [2]. This condition was most unusual in relation to typical research studies because it produced radiation levels in the normally occupied parts of the Synchrotron Building that were clearly hazardous to staff members (tens of milliroentgen per hour) (Ames 1961a). Dosimeter results for synchrotron personnel verified that the machine was not operated in this condition for significant periods (during the May 16, 1961, survey or at any other time) (Martin 2006c).

The synchrotron was operated part time by Physics Department faculty and graduate students. The typical research schedule would have been weekdays and evenings plus occasional weekends, which was estimated at a maximum of 3,000 hr/yr. Much of this time would have been occupied with experiment setup, maintenance, system startup, etc., so the maximum operating time would have been about 2,000 hr/yr [3]. Most of this operating time would have been dedicated to various material property studies that involved electron beams, beam currents, and targets that produced gamma fields outside the building that were at least a factor of 10 (and more likely a factor of 100) less than the survey dose rates in Ames (1961a) [4].

Table 4-1. Survey of fenceline gamma dose rates around the synchrotron facility with the beam directed west (Ames 1961a, p. 7).

Location	Dose rate (mrem/hr)[a]	Location	Dose rate (mrem/hr)[a]
1	0.75–1	14	1.8
2	3.5	15	1.8
3	7	16	1.8
4	6	17	1.6
5	5.5	18	1.5
6	7	19	1.5
7	4.75	20	1.75
8	3.5	21	1.6
9	2.6	22	1.6
10	1.5	23	1.5
11	2	24	0.75–1
12	2	25	4.5
13	1.8		

a. Background reading in the beam direction (see locations 3-6) before turning on the beam was 0.5 to 1.0 mrem/hr.

A topographical map that was part of the 1961 radiation survey (Ames 1961a) indicates the Synchrotron Building was surrounded by hills that would have protected nearby buildings from a direct beam. Therefore, the primary source of environmental exposures from this facility would have been skyshine from the synchrotron. The Waste Chemical Handling Facility (see Figure 2-2) was not built until 1980, so the area was not affected by synchrotron operations between 1949 and 1971. The ALRR (now the Applied Science Complex) is about 750 ft from the nearest part of the Synchrotron Building, and construction or operations at the two facilities overlapped, at a maximum, from 1962 to 1971. Skyshine from both heavy particles and photons decreases at rates equal to or greater than the reciprocal of the square of the distance ($1/r^2$) from accelerator facilities (NCRP 2003). If it is conservatively assumed that all of the radiation field measured in the 1961 survey was from skyshine, the dose rate at the ALRR would have been less than 0.13 mrem/hr during the worst-case operation and less than 0.013 mrem/hr during routine operations. On the main campus of the University, the dose rate would have decreased to less than about 0.00025 mrem/hr during routine operations.

It is favorable to claimants to use the environmental external dose from synchrotron operations at the ALRR for full-time exposure (2,000 hr/yr) for all locations at 25 mrem/yr for the period from 1949 to 1971.

4.3 ALRR OPERATIONS PERIOD, 1965 TO 1977

The ALRR started operations in February 1965 and continued through December 1977 (Ames 1967; Voigt 1981). The facility is surrounded by a fence about 700 ft from the reactor building that designates what is on and off the site (see Figures 2-2 and 4-2). From review of dosimetry records, it seems evident that all employees working inside the ALRR fence were provided dosimeters. However, not all dosimetry records are identified with names. Therefore, not all workers at the ALRR have recorded doses. In addition, environmental doses from gaseous effluents released from the operating reactor were not monitored. However, environmental doses to the public from airborne releases were calculated and reported (Voss 1975, 1976, 1977). The only air monitoring station in the vicinity of the reactor was on the roof of the reactor building, as shown in Figure 4-3.

Environmental Monitoring at Ames Laboratory: Calendar Year 1974 was the first annual report to provide gamma spectroscopy of environmental media samples (Voss 1975); subsequent annual reports (Voss 1976, 1977) provided similar results. The average release estimates from these reports

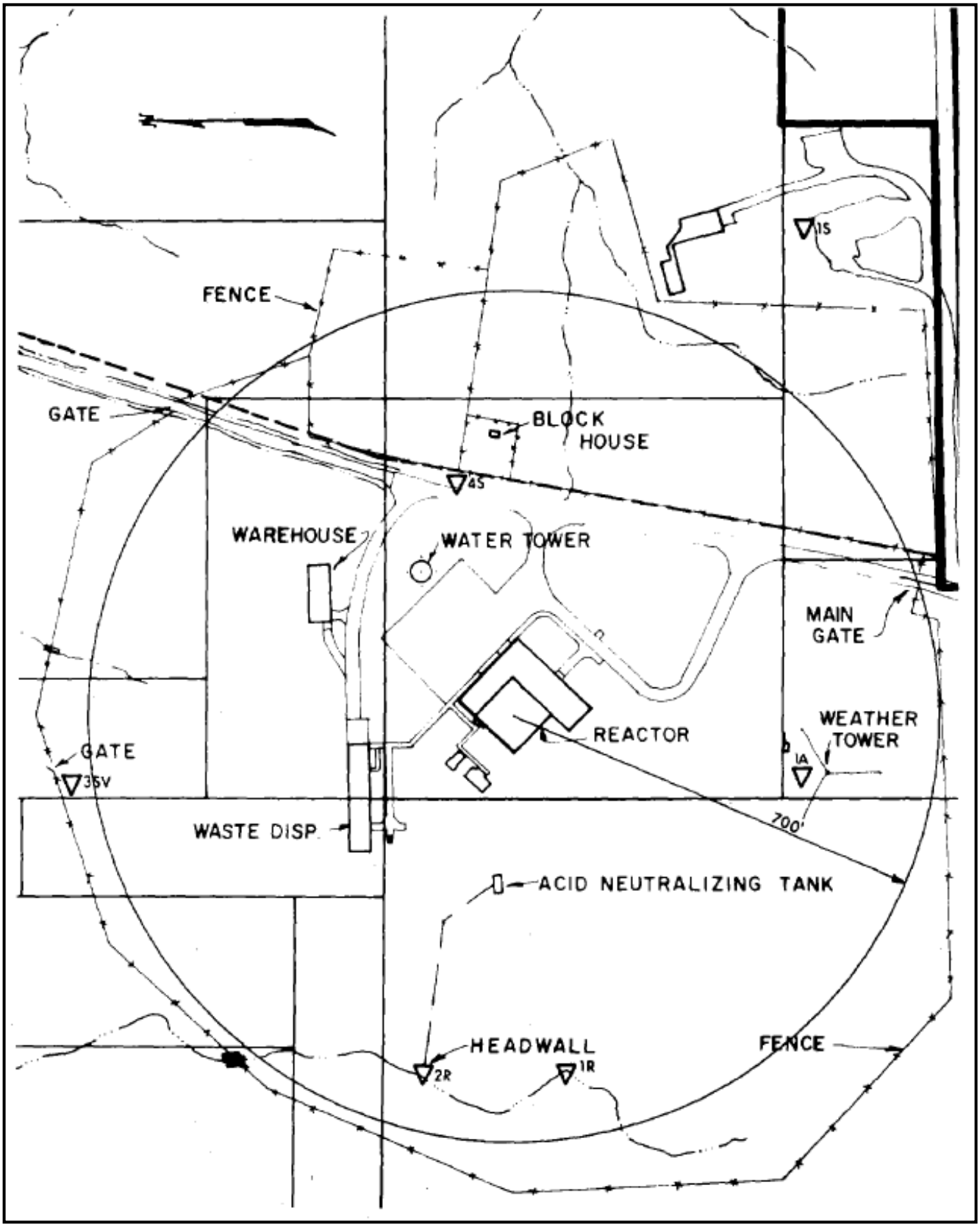


Figure 4-2. Ames Laboratory Research Reactor Site. Source: Ames (1974).

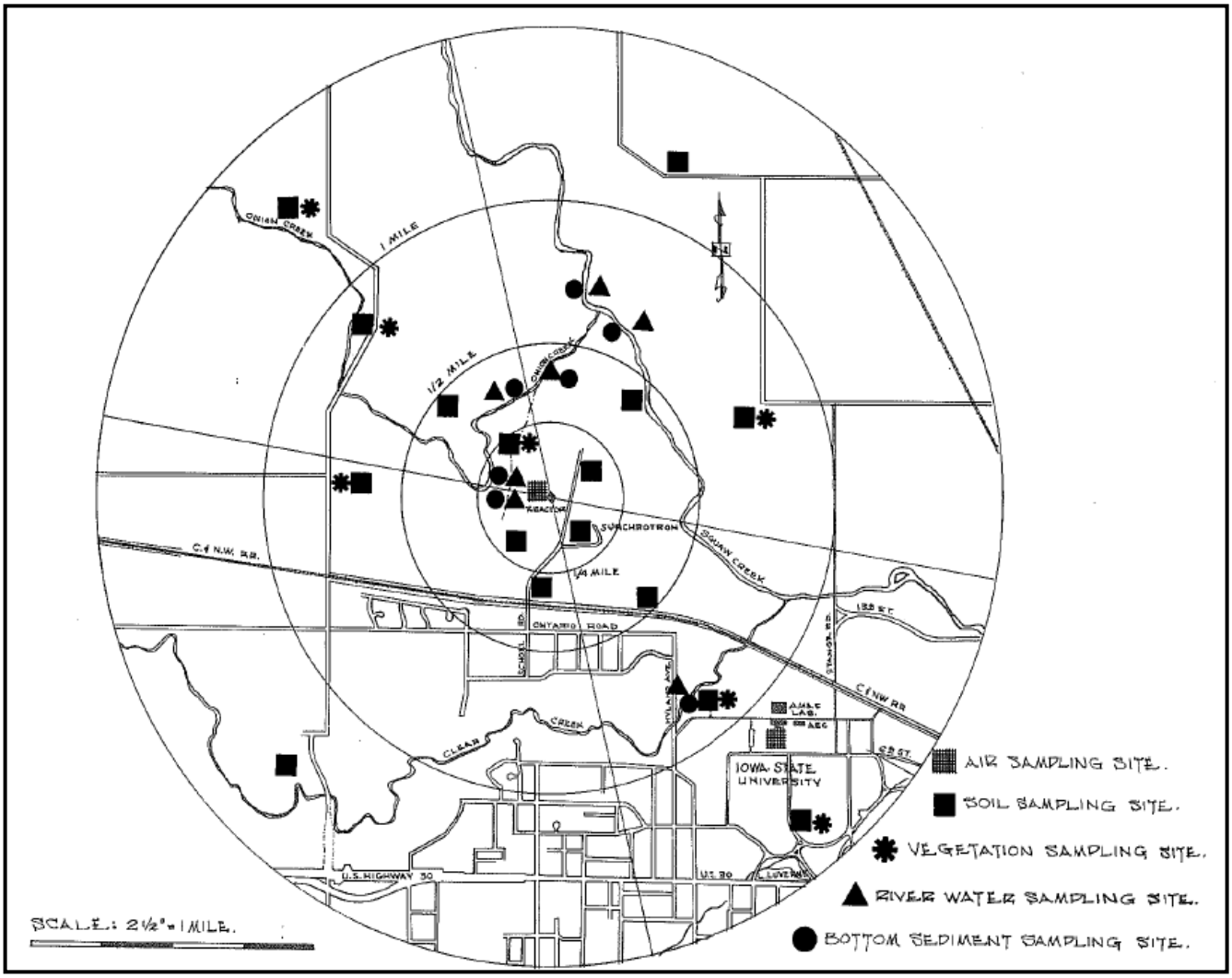


Figure 4-3. Environmental monitoring in the vicinity of the ALRR. Source: Voss (1975).

were used to determine the estimated environmental dose to offsite workers from gaseous releases from reactor operations. From the effluent data, it was shown that the contribution to radioactivity in air from ALRR operations consisted principally of ^{41}Ar and tritium (Voss 1975, 1976, 1977). An atmospheric dispersion model, which used annual average meteorological data for Ames and an exposure model (Napier et al. 2004; Napier 2006b), was used to determine external dose rates from the ^{41}Ar and inhalation intake estimates for the tritium.

At the fenceline location with the highest dose from gaseous effluents, the average annual dose to a person for the entire year (8,760 hours) was estimated to be 4.2 mrem from ^{41}Ar during the years of reactor operation. An offsite worker who worked full time at this location would not have been exposed for more than 2,080 hr/yr, which would result in a submersion dose of about 1 mrem/yr.

At the fenceline location with the highest concentration of tritium effluents, the average annual intake of tritium to a person for the entire year (8,760 hours) was estimated to be about 2.9 µCi/yr during the years of reactor operation. An offsite worker who worked full time at this location would not have been exposed for more than 2,080 hr/yr, which would result in an intake of tritium of about 0.7 µCi/yr or 2,700 pCi/d.

4.4 AMES LABORATORY RESEARCH AND DEVELOPMENT ACTIVITIES, 1942 TO PRESENT

R&D activities have been carried out at Ames Laboratory from 1942 to the present. It was the R&D capabilities of Iowa State College that attracted the attention of the Manhattan Engineer District in 1942 (Ames 1960a). Buildings owned by the government (AEC, ERDA, and now DOE) in addition to the uranium/thorium, synchrotron, and research reactor buildings discussed above, are on the main campus of ISU (see Figure 2-1).

The Ames Laboratory facilities consist of (Ames 1996):

- Metals Development Building
- Spedding Hall (formerly the Research Building)
- Wilhelm Hall (formerly the Metallurgy Building)
- TASF

Facilities owned by ISU but leased to the Government include (Ames 1996):

- Gillman Hall (formerly the Chemistry Building)
- Zaffarano Physics Addition
- Office and Laboratory Building

As can be seen in Figure 2-1, these R&D facilities are an integral part of the ISU campus and are, therefore, open to staff, students, and public traffic [5]. There were no restrictions on the movement of this traffic outside the facilities. In addition, there was no monitoring of radiation exposure or contamination of the personnel or environmental areas around and among these facilities. As research facilities, the quantities of radioactive materials involved in the R&D work were small compared to the production facilities.

There was insufficient information about releases from the R&D facilities to determine intakes directly. Principal sources would have been uranium, thorium, and fission products from the hot canyon/hot cell in the Research Building. Because of the smaller amounts of radioactive materials in the R&D facilities in comparison with those in the production facilities, the releases from the R&D facilities were assumed to be one one-hundredth of the releases from the production facilities [6]. That assumption resulted in daily environmental intakes of 5×10^{-2} and 7×10^{-4} pCi/d for uranium and thorium, respectively. A review of environmental intakes at somewhat related sites [Argonne National Laboratory–East (ANL-E), Brookhaven National Laboratory (BNL), and Lawrence Berkeley National Laboratory (LBNL)] was made. Neither ANL-E nor BNL listed any environmental intakes of uranium or ^{232}Th (ORAUT 2006a, 2010a). LBNL listed environmental intakes of unspecified gross alpha and gross beta emitters. From 1939 through 1999, the gross alpha intakes ranged from 6.6×10^{-3} to 2.7×10^{-1} pCi/d, and the gross beta intakes ranged from 1.5×10^{-1} to 63 pCi/d (ORAUT 2010b). The highest gross alpha intake occurred in 1996 and appears to be an anomaly; the second highest gross alpha intake was 5.9×10^{-2} pCi/d. Therefore, the proposed uranium intake is comparable to the second highest gross alpha intake from LBNL and, therefore, appears to be a reasonable upper bound. For environmental intakes for workers exposed around the R&D facilities, a daily intake of 5×10^{-2} pCi of ^{234}U and 7×10^{-4} pCi each of ^{232}Th, ^{228}Ra, and ^{228}Th was assumed [7]. Because the nature of the research materials is not known, absorption types F, M, or S for the uranium and M or S for the thorium/radium isotopes were assumed [8].

A hot laboratory was operated in the Chemistry Building but was replaced in 1951 by a “hot canyon/hot cell” in the Research Building. In the 1940s, the hot laboratory was used to study extraction of plutonium from irradiated uranium by means of ion exchange columns. A final report on

plutonium R&D was published in July 1946 (Fulmer 1947), at which time plutonium R&D and the personnel who conducted it were transferred to Los Alamos National Laboratory (LANL; Payne 1992).

> *A hot laboratory, capable of handling 5 curies through the adsorption process, was designed and built. The method proved successful in a number of runs. The uranium and plutonium were separated from one another and from the fission products using 1 kg samples of uranium, from the pile, which had an activity of 5 curies. (Fulmer 1947)*

Releases were estimated from this operation by making the following assumptions (see Bihl 2006 for details of the calculation):

- A total annual throughput of the laboratory of 50 Ci/yr
- An airborne fraction of 0.002 for boiling liquids
- Filtration efficiency for average particle sizes of 99.5% [high-efficiency particulate air (HEPA) filters were just being developed during this period]

These assumptions produce a total release of fission products of 5×10^8 pCi/yr.

Using the least dispersive approach recommended by the National Council on Radiation Protection and Measurements (NCRP) in Publication 123 (NCRP 1996), the ground-level annual average air concentration would have been 13.2 pCi/m^3. Using the inhalation rate of 2,400 m^3/yr and converting to a daily intake results in 87 pCi/d.

Using the recommended fission product ratios for 180-day cooled fuel from *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2007a), an 87-pCi/d intake of fission products is assigned to specific radionuclides as listed in Table 4-2.

Table 4-2. Annual environmental fission product intakes from the hot laboratory in the Chemistry Building (1943–1981).

Radionuclide	Relative fraction	Intake (pCi/d)
Ce-141	0.0221	1.92
Ce-144	0.2191	19.1
Cs-134	0.0054	0.470
Cs-137	0.0208	1.81
Eu-155	0.0014	0.122
Fe-55	0.0172	1.50
Nb-95	0.2492	21.7
Pm-147	0.0546	4.75
Ru-103	0.0321	2.79
Ru-106	0.0844	7.34
Sr-89	0.0558	4.85
Sr-90	0.0157	1.37
Y-91	0.0911	7.93
Zr-95	0.1311	11.4

Because maximizing assumptions were used for most of the parameters in this analysis, the distribution is an upper bound (constant). The absorption type for each radionuclide that is most favorable to claimants, as listed in International Commission on Radiological Protection (ICRP)

Publication 68 (ICRP 1995), should be used, with the exception that only type F should be used for strontium. The date of the first use of the hot laboratory was not found. Because it received fuel from the Chicago Pile, the earliest reasonable use would have been 1943 [9]. It was replaced by a hot cell with improved ventilation and filtration in the Research Building in 1951, which was removed from service and decontaminated in 1982. Releases from the hot cell are not documented but would have been smaller than those estimated above for the hot laboratory. It is favorable to claimants to apply the Table 4-2 intakes for each year from 1943 through 1981 [10].

4.5 SIGNIFICANT ENVIRONMENTAL EVENT

The only significant environmental event in the history of the Ames Laboratory was the release to the environment from operations that occurred from July 1951 through August 1952 (Hayes 1956). Metallic thorium was being produced from thorium nitrate tetrahydrate. During an early stage of the process, a filtrate with traces of thorium in the form of thorium nitrate and oxalate was released to the sewer that connected to the City of Ames sewer system. At the time disposal to the sewer was chosen, it was believed that this waste had a very low level of radioactivity. However, due to a change in feed material supplied to the Laboratory, considerable quantities of mesothorium (^{228}Ra) were being discharged in the filtrate. Mesothorium is one of the progeny of thorium decay; it decays by emitting gamma rays and beta and alpha particles. All sewage was processed in a complete-treatment sewage plant, which resulted in liquid effluent and dry sludge that was used for fertilizer. The liquid effluent contained negligible quantities of mesothorium. The dry product was spread on lawns at the sewer plant, airport, municipal parkway, and a cemetery. A thorough study of the incident disclosed that there was little hazard to the public or the employees of the sewer plant (Hayes 1956). Table 4-3 is a summary of environmental dose measurements at the four facilities in Ames, Iowa:

Table 4-3. Environmental dose measurements at the four Ames facilities (mrem/yr). Source: Voss 1979.

Lawn site	Maximum[a]	Average
Sewer plant	780	400
Cemetery	440	350
Airport	530	350
Parkway	530	350

a. Background in central Iowa is about 300 mrem/yr.

An aerial survey showed doses to be lower than those measured on the ground. Occupancy factors would reduce the doses in Table 4-3 by a large fraction, making them below the current public dose limit of 100 mrem/yr (10 CFR Part 20).

All the locations in Table 4-3 are outside the ISU campus and all are accessible by the public except portions of the sewer plant. Most Ames Laboratory workers were not exposed to the radioactive materials released during this event and were not involved in responding to the event. However, the Ames Laboratory health physics staff responded to the release event and were exposed to the materials while making radiation measurements and collecting environmental samples for analysis. It is reasonable to assume that radiation exposures to the health physics staff were monitored and offsite doses were included in their recorded occupational doses. Thus, no additional dose is recommended for occupational environmental dose for any workers.

4.6 ENVIRONMENTAL CONTAMINATION, 1953 TO PRESENT

Gamma spectral analysis of samples of environmental media was not initiated until November 1974 (Ames 1974). By that time, uranium and thorium production and scrap recovery had ceased, and Annex 1 had been demolished in 1953 (Karsjen 2003). Annex 2 operations ceased in 1953, and the building was razed in 1972 (Ames 1985). Synchrotron operations ceased in 1971, and the building

was decommissioned in 1990 [11]. ALRR operations ceased in 1977 and D&D was completed in 1981 (Voigt 1981). Only R&D activities continued after 1981.

Soil samples collected at five locations around the campus in 1974 showed only ^{137}Cs in very low concentrations, as listed in Table 4-4.

Table 4-4. Gamma analysis of soil samples (Voss 1975).

Site	Sample weight (g)	Cs-137 (pCi/g)[a]
1S	336	0.839
2S	332	1.100
4S	500	1.100
6S	341	0.991
7S	426	0.286
Background	336	0.294

a. Gamma detection limit for Cs-137 in soil samples was 0.01 pCi/g.

A summary of soil samples from 24 locations around the ISU campus in 1974 for alpha and beta particle activity is given in Table 4-5.

Table 4-5. Summary of 24 soil samples for beta and alpha activity (pCi/g). (Voss 1975).

Activity	Beta[a]	Alpha[b]
Average	10.56	0.61
High	13.01	0.96
Low	7.66	0.38
Background	11.9	0.76

a. Beta detection limit was 0.25 pCi/g.
b. Alpha detection limit was 0.10 pCi/g.

These analytical results are consistent with the low doses that Ames Laboratory has reported over the years and with the position of not monitoring workers in the environs of Laboratory facilities. They are the bases for demonstrating that environmental intakes from contamination other than uranium and thorium discussed above were negligible.

4.7 SUMMARY OF ENVIRONMENTAL EXTERNAL DOSES AND INTAKES

Tables 4-6 and 4-7 summarize the conclusions of the preceding sections on environmental external doses and intakes.

Table 4-6. Environmental external doses (to be applied only to unmonitored workers).

Location	Dates	Dose (mrem/yr)[a]	Distribution
Skyshine from Synchrotron Building	1949–June 1971	25	Constant
Ar-41 from ALRR	1965–1977	1	Constant
All other R&D Buildings (see list in Section 4.4)	All	Negligible	N/A

a. The energy range for all environmental external dose is assumed to be 100% 30–250 keV (NIOSH 2007b).

Table 4-7. Summary of environmental intakes.[a]

Dates	Radionuclide/absorption	Intake (pCi/d)	Distribution
August 1942–1953	U (assume U-234); type M or S	5	Constant
1954–present	U (assume U-234); type F, M, or S	0.05	Constant
June 1943–April 1953	Th-232, Ra-228, Th-228; all type M	0.07 each	Constant
1954–present	Th-232, Ra-228, Th-228; type M or S	0.0007 each	Constant
1965–1977	Tritium	2,700	Constant
1943–1981	Fission products per Table 4-2	Per Table 4-2	Constant

a. Apply the environmental intakes in this table if no occupational intakes are applied for the same radionuclide and the same period in accordance with the instructions in Table 5-8.

5.0 OCCUPATIONAL INTERNAL DOSE

The radionuclides of interest for internal dose at Ames Laboratory are uranium, thorium, tritium, and fission products. Based on the Evaluation Report for Petition SEC-00038 (NIOSH 2006) for the period from August 13, 1942, through December 31, 1954, it was determined that internal doses from the production and casting of thorium metal for DOE employees or contractor or subcontractor employees who worked in Chemistry Annex 1 (also known as the old women's gymnasium and Little Ankeny), Chemistry Annex 2, Chemistry Building (now Gilman Hall), Research Building, or Metallurgy Building (now Harley Wilhelm Hall), cannot be reconstructed. Therefore, only doses from thorium contamination left over after 1954 are addressed here. As stated in Section 1.3, this SEC class has been expanded to include all Ames employees, contractors, and subcontractors.

In addition, based on the Evaluation Report for Petition SEC-00075 (NIOSH 2007a) for the period from January 1, 1955, through December 31, 1970, it was determined that internal doses from maintenance and renovation activities of the thorium production areas in Wilhelm Hall (a.k.a. the Metallurgy Building or "Old" Metallurgy Building) cannot be reconstructed for sheet metal workers, physical plant maintenance, and associated support staff (includes all maintenance shop personnel of Ames Laboratory) and supervisory staff who were monitored or should have been monitored. Therefore, internal doses from these activities to this class of workers cannot be assessed using the approach described in Section 5.2. As stated in Section 1.3, this SEC class has been expanded to include all Ames employees, contractors, and subcontractors.

In addition, based on the Evaluation Report for Petition SEC-00166 (NIOSH 2010b) for the period from January 1, 1955, through December 31, 1960, it was determined that internal doses from radionuclides other than uranium in the Hot Canyon and Hot Cell (or the Cave) of the Research Building (Spedding Hall) cannot be determined for any workers. Further, this petition was expanded to include all workers at Ames Laboratory during this period due to the lack of information on work locations of employees.

5.1 URANIUM EXPOSURE

Few data on uranium exposure were found (Stone 1951); however, the SEC petition evaluation reports determined that doses could be estimated (NIOSH 2006, 2007a, 2010b). If uranium bioassay data are available for a worker, they should be used. However, because it is unlikely that uranium bioassay data will be available, default intakes were determined. Because even workplace sampling data were unavailable, this analysis used data from the documents: *Site Profiles for Atomic Weapons Employers* (AWE) *that Worked Uranium Metals* (Battelle 2011) and *The Industrial Hygiene of Uranium Refining* (Christofano and Harris 1960) to estimate doses to workers in the Chemistry Building, Physical Chemistry Annex 1, and Physical Chemistry Annex 2. These two documents are representative of the potential intakes that workers at Ames Laboratory might have had because the processes were often developed at the Laboratory and remained similar to processes used at AWE sites. The two processes that were used here to estimate doses are Metal Reduction (Christofano and Harris 1960) and Uranium Scrap Recovery and Casting (Battelle 2011).

5.1.1 Estimating Internal Doses from Uranium Metal

Historical aspects of the production of uranium metal at Ames Laboratory are discussed in Section 2. Although there was much historical information on the general methods for producing uranium metal (Fulmer 1947), there was very little specific information useful for determining intakes from uranium. If uranium bioassay data are available for a worker, they should be used to estimate the intake. However, it is likely that bioassay data will not be available. In that case, the information below has been developed from Battelle (2011) and Christofano and Harris (1960) as methods for assigning internal dose to workers.

The next two sections describe estimated doses for workers from August 13, 1942, through December 31, 1954. Additional information is provided to calculate dose to workers from radioactivity left over from the sites where uranium had been processed after 1954. Because there were numerous studies of various compounds of uranium during this period, all three absorption types (fast, moderate, and slow) were possible.

5.1.1.1 Uranium Inhalation

From 1942 through August 1945, individuals working in Physical Chemistry Annex 1 (also known as Little Ankeny) can be assigned doses from inhalation of uranium [12]. Similarly, individuals who worked in the Chemistry Building during this period or in Physical Chemistry Annex 2 from 1944 through December 1945 should be assigned doses from inhalation of uranium [13]. Job titles of researchers acknowledged for their work in uranium production (Fulmer 1947) include (1) chemist, (2) associate chemist, (3) junior chemist, (4) research assistant, (5) junior research assistant, (6) physicist, (7) analyst, (8) assistant physicist, (9) associate director, and (10) director. However, it is not entirely clear how much time these researchers spent in the area where production was performed.

Because it is not clear if there were clerical, janitorial, or nontechnical personnel and other types of researchers working in these buildings, and it is not known what precautions might have been taken for contamination control, it can be assumed that all individuals who worked in the buildings had some potential for exposure to uranium.

The data in Tables 5-1 and 5-2 were derived from data in Christofano and Harris (1960). In Christofano and Harris (1960), there is a description of the process for metal reduction that is similar to the process used at Ames Laboratory for production of uranium metal (Fulmer 1947). The primary difference appears to be that at Ames Laboratory the process used granulated calcium metal and at AWE sites the process used magnesium. Fulmer (1947) describes the process using magnesium, and it appears to be similar enough to be representative of the intakes at Ames Laboratory.

Table 5-1. Chemistry Building uranium intakes (pCi/d).

Period	Inhalation	Ingestion
Aug 1942–December 1953	8.5[a,b]	0.09[b]

a. No data were available for determination of intakes in the Chemistry Building; therefore, it was assumed that research activities would have one-hundredth the intake of production activities since uranium metal production was moved to the Physical Chemistry Annex 1.
b. Values are for workers assumed to work in research or production full time. For supervisors, assume one-quarter of the intake; for all other employees (clerical, janitorial, security, etc), assume one-tenth of the supervisor's intake.

Table 5-2. Physical Chemistry Annex 1 uranium intakes (pCi/d).

Period	Inhalation	Ingestion
August 1942–August 1945	853[a]	8.7[a]

a. Values are for workers assumed to work in research or production full time. For supervisors, assume one-quarter of the intake; for all other employees (clerical, janitorial, security, etc), assume one-tenth of the supervisor's intake.

Table 5-3. Physical Chemistry Annex 2 uranium intakes (pCi/d).

Period	Inhalation	Ingestion
January 1944–December 1950	6,061[a]	124[a]
January 1951–December 1953	5,556[a]	114[a]

a. Values are for workers assumed to work in research or production full time. For supervisors, assume one-quarter of the intake; for all other employees (clerical, janitorial, security, etc), assume one-tenth of the supervisor's intake.

Christofano and Harris (1960) provides data for multiple stages of the production operation for operators. Using the techniques outlined in *Default Assumptions and Methods for Atomic Weapons Employer Dose Reconstructions* (Battelle 2007), the values in the above tables were calculated from Table 8 of Christofano and Harris (1960). However, review of Ames Laboratory documentation did not reveal any information specific enough for determination of who would be responsible for what aspects of the process and for how long. Therefore, the value used for determining inhalation intakes, which was from Christofano and Harris (1960, Table 8), is for the Bomb Preparation operator, who is assumed to work a 2400-hour year. This value was the highest intake rate for the metal reduction process. This number is then scaled for the potential for intake (see Tables 5-1 and 5-2).

Battelle (2011) provides data for a process of scrap recovery that is similar to the scrap recovery process described in Fulmer (1947). Data from Battelle (2011, Table 7.8) are used in Table 5-3 for Ames Laboratory workers who worked in the Physical Chemistry Annex 2 building. The value used in Battelle (2011, Table 7.8) represents the most conservative intake for the scrap recovery operations, similar to those performed in Physical Chemistry Annex 2.

Research in the Chemistry Building began in January 1942. For workers involved only in research from January through July 1942 in the Chemistry Building, an exposure of one-tenth of that of the workers involved in the production operations is assumed, which corresponds to smaller quantities of uranium [14]. This period is prior to the beginning of the covered period for the EEOICPA statute, which is August 1942, the start of the Manhattan Engineer District, known later as the Manhattan Project. The process developed by this research was moved to the Physical Chemistry 1 building in July/August of 1942. There are very few details on the research activities in the Chemistry Building after July 1942; therefore, it was assumed that another one-tenth fraction should be applied (for a 1/100 reduction overall from operations). These intakes should be applied to researchers in the Chemistry Building through 1953, when production ended.

Individuals supervising the production processes were assumed to be exposed for one-fourth of the time of the production staff [15].

For workers not directly associated with uranium metal research or production, an exposure of one-tenth of that of the supervisors was assumed (see Tables 5-1, 5-2, and 5-3) [16].

The intakes in Tables 5-1, 5-2, and 5-3 were compared to the few actual bioassay results found for workers in approximately the same period. Chapter 7 in Stone (1951), "Uranium Excretion Studies," provided data from a series of uranium bioassays obtained from Ames Laboratory workers in 1944 and 1945. Of special interest was a series of samples from the supposedly highest exposed worker at Ames Laboratory and samples from the most highly exposed group of workers at the Laboratory (21 samples from 11 workers). An intake evaluation was performed on the results for the highest exposed worker assuming chronic intake from the start of that person's employment and absorption type M uranium (the document indicated the person was exposed to UF_4). The estimated intake was 1,200 µg/d or 820 pCi/d assuming natural uranium. This is consistent with the intake estimate in Table 5-2 for Physical Chemistry Annex 1 and quite a bit lower than that in Table 5-3 for Annex 2. The average bioassay result for the group of highest exposed workers was 75 µg/L. Assuming chronic intake for 1 year before the bioassay, the estimated intakes were:

- Absorption type F: 390 µg/d, 260 pCi/d
- Absorption type M: 1,670 µg/d, 1,100 pCi/d
- Absorption type S: 45,400 µg/d, 31,000 pCi/d.

If the geometric mean of the data is used, the estimated intakes are smaller; if the highest bioassay result of the set is used, the estimated intakes are 2.7 times greater. For type M, this range is still consistent with the intakes in Tables 5-1, 5-2, and 5-3. The type S intake estimates are greater than

those in the tables but it is unlikely anyone was actually exposed to just type S for long periods. Uranium tetrafluoride and metal were the principal forms of uranium in the production facilities.

Another small set of urinalysis data was found involving five or six samples from six workers from July 13 to September 4, 1944 (Tybout 1944). The data were hard to read and most results were listed as 0.00 mg/L. Using the data from the worker with the highest result and the most nonzero results and assuming chronic intake for 1 year of absorption type M uranium, the estimated intake was 850 µg/d (580 pCi/d). The estimated intake was slightly less if the intake period started in August 1942. These results provide a measure of confidence that the default intakes in Tables 5-1, 5-2, and 5-3 are favorable to claimants [17].

5.1.1.2 Uranium Ingestion

Ingestion intakes are determined using NIOSH (2004) and are listed in Tables 5-1, 5-2, and 5-3. To describe this process in brief, the daily ingestion rate from food contamination is given by $I = 0.0985A$, where I is the daily ingestion rate in picocuries per day and A is the average air concentration in picocuries per cubic meter. The ingestion rate I must be adjusted for the fact that the Integrated Modules for Bioassay Analysis (IMBA) computer program assumes chronic intakes, even during weekends, and that the number of hours worked in a year changed over time (Battelle 2007). The adjustments result in an IMBA chronic intake rate $I_{\text{IMBA}} = 3.373 \times 10^{-5} Ah$ where I_{IMBA} is the daily intake, A is the median air concentration using the specified units, and h is the number of hours in a work year. Making similar adjustments to Neton's equations for incidental hand-to-mouth ingestion (NIOSH 2004), the chronic IMBA daily intake rate is $I_{\text{IMBA}} = 3.425 \times 10^{-5} Ah$ where I_{IMBA} is the daily chronic intake rate, A is the median dust concentration, and h *is* the number of hours in a work year. The total ingestion rate is the sum of the food contamination and incidental hand-to-mouth ingestion rates and is $I_{\text{IMBA}} = 6.798 \times 10^{-5} Ah$ where the variables are the same as defined above. This intake applies to the period from 1942 through August 1945. The rationale used in inhalation intakes for reduced fractions for other workers is applied to ingestion intakes as well.

5.1.1.3 Resuspension During Periods with No Uranium Operations

There was a potential for internal exposure to resuspended material from AEC work during non-AEC operations soon after the actual operations. To estimate exposure from resuspended materials, this analysis assumed that surfaces in the building became contaminated by deposition of uranium dust during operations similar to metal working operations. The daily intake rates for the Chemistry Building can be estimated by reducing the generic exposure estimate from ORAUT (2006e) by a factor of 10 to account for the conclusion of the work, the standard laboratory precautions that were in place, and the production time at about 50% of full time in the Chemistry Building. The maximum intake from inhalation for the researchers in metal production operations (Physical Chemistry Annex 1) and scrap recovery (Annex 2) were used to estimate intakes for these facilities. The method used in Battelle (2011) was applied to the maximum intake described for each process. Table 5-4 lists the intake in picocuries per day for each building for the specified periods. These intakes apply to anyone in the building, regardless of job description. NIOSH (2004) indicates that the ingestion rate, in terms of disintegrations per minute for an 8-hour workday, can be estimated by multiplying the air concentration in disintegrations per minute per cubic meter by a factor of 0.2. The ingestion intakes are also provided in Table 5-4.

Table 5-4. Resuspension during periods with no uranium operations (pCi/d).

Period	Chemistry Building inhalation/ingestion	Physical Chemistry Annex 1 inhalation/ingestion	Physical Chemistry Annex 2 inhalation/ingestion
September 1945–December 1953	N/A	17.5/1.6	N/A
January 1954–May 1976	4.1/0.68	N/A	124.7/11.2 (only through 1972), then 0[a]

a. Physical Chemistry Annex 2 was razed in 1972.

5.1.1.4 Uranium Excretion Plots for Use with Bioassay Data

Figures 5-1, 5-2, and 5-3 are plots for expected urinary excretion assuming chronic intakes at the highest intake values (scrap recovery) for both inhalation and ingestion, for chronic intake periods of 365, 730, and 1,826 days, respectively. If bioassay results are available for a worker, they can be compared to the excretion curves in the figures if efficiency techniques are being used; for instance, for a noncompensable efficiency case based on the default intakes, the Energy Employee's actual bioassay should be less than the applicable curve in the figures [18]. Types F, M, and S are defined as fast, medium, and slow, respectively, in the figures.

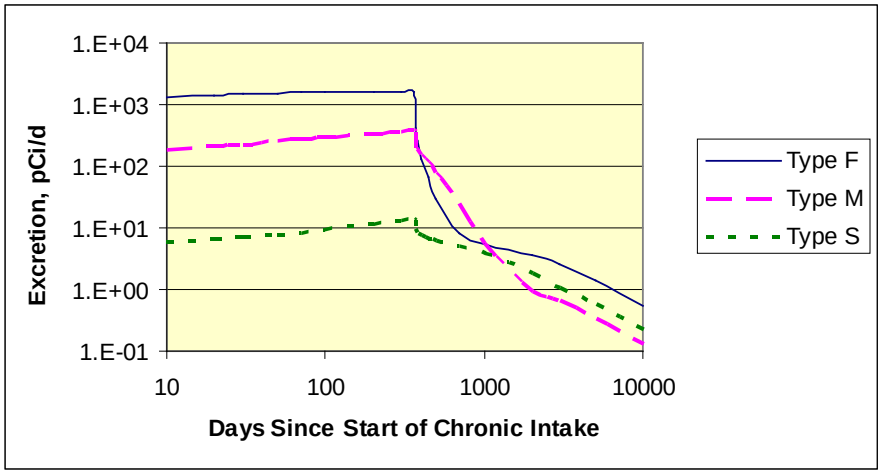


Figure 5-1. Urinary excretion of uranium expected from 365-day chronic intake at 6,061 pCi/d inhalation and 124 pCi/d ingestion.

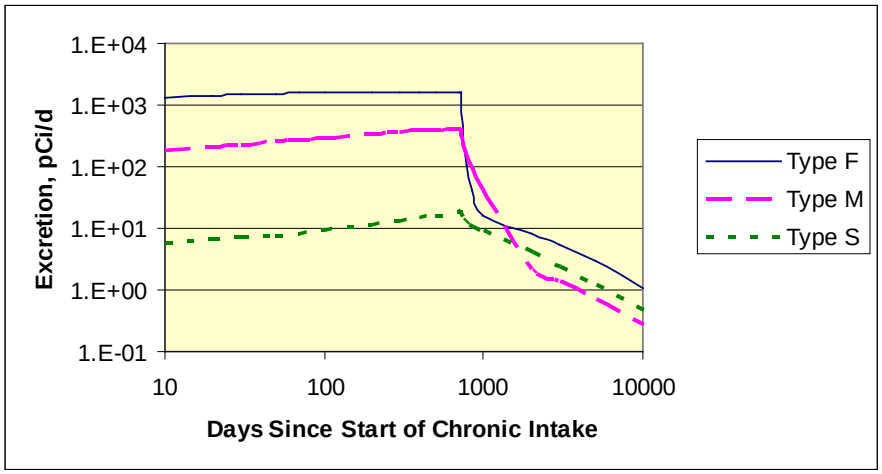


Figure 5-2. Urinary excretion of uranium expected from 730-day chronic intake at 6,061 pCi/d inhalation and 124 pCi/d ingestion.

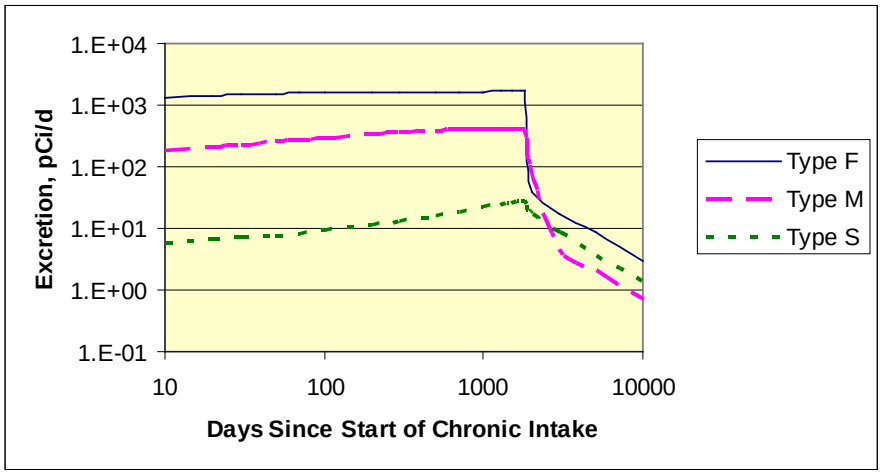


Figure 5-3. Urinary excretion of uranium expected from 1,826-day chronic intake at 6,061 pCi/d inhalation and 124 pCi/d ingestion.

5.2 THORIUM EXPOSURE FROM THORIUM CONTAMINATION AFTER 1954

As discussed in Section 1.3, NIOSH determined that it was not feasible to complete dose reconstruction for internal dose for thorium, plutonium, and thoron for 1942 through 1954 (NIOSH 2006) and, therefore, internal dose is not addressed here for those radionuclides during those years.

Research using small quantities (micrograms) of plutonium in the 1940s involved methods of extraction of plutonium from up to 5 Ci of fission products in irradiated fuel, which would not have contained much plutonium (Fulmer 1947). There is no indication of research that involved sufficient plutonium such that exposure to plutonium after 1954 would have been significant.

However, Metallurgy Building (Wilhelm Hall) was the location of the main activities with thorium and is still in use. As discussed in Section 1.3, it is not feasible to determine internal doses for sheet metal workers, physical plant maintenance and associated support staff (includes all maintenance shop personnel of Ames Laboratory), and supervisory staff who performed maintenance and renovation activities in Wilhelm Hall (Old Metallurgy Building) during the period of January 1, 1955, through December 31, 1970. This method can be used, however, to provide an estimate of internal exposures for others who performed work in this building for non-SEC claims.

In March 1952, a study was performed in Wilhelm Hall to identify health and safety issues that were occurring during refining and thorium metal production. The results of the study are in *Ames Research Laboratory Occupational Exposure to Thorium and Beryllium* (Klevin 1952). Starting in 1984 and continuing through the early 1990s, surveys were conducted in Wilhelm Hall to determine locations of contamination left from the early production years. This information is in Hokel et al. (1998). The following discussion using data from Hokel et al. and Klevin provides an estimate for intakes by workers in Wilhelm Hall from 1955 to the present. A summary of the air concentrations and daily inhalation and ingestion intakes is provided in Table 5-5.

All the data in Hokel et al. (1998) were reviewed and considered. Much of the data was related to locations that were hard to access and considered not to be an inhalation issue. There was one set of data that had floor surveys made in 1988, but these locations all had fixed activity, indicating that this was from beta radiation (see Appendix 6 of Hokel et al.). For this reason, the recommendation for assessing dose from thorium from 1954 through the present was provided. The accessible areas of the building, including rooms, air ducts, hallways, stairwells, transformer rooms, etc., were surveyed starting in 1984. The data in Hokel et al. present an overview of the survey results. In 1996, some measurements were made using an Alpha Continuous Air Monitor in the sub-basement

pipe tunnels and in large vertical void spaces in the stairwells. All of these results were less than background for thorium, thoron, or radon (Hokel et al. 1998). Therefore, the results used for this estimation are from a pipe tunnel survey that showed removable contamination on smears ranging from background (3 dpm) to 1,224 dpm. These numbers are high for generally accessible areas and with the possibility of the contamination becoming resuspended. Although other survey numbers in the report with removable contamination are higher, they are in locations that are inaccessible (in a drain line), hard to reach (inside a drawer), or small (edge of a sink), or the contamination was fixed.

Table 5-5. Air concentrations and daily intakes of ^{232}Th at Wilhelm Hall after 1954.

Year	t (yr)	Air concentration (dpm/m^3)	Chronic inhalation intake (pCi/d)	Chronic ingestion intake (pCi/d)
1955[a]	0.5	448	1330	40.4
1956[a]	1.5	392	1160	35.3
1957[a]	2.5	343	1020	30.9
1958[a]	3.5	300	887	27.0
1959[a]	4.5	262	776	23.6
1960[a]	5.5	229	679	20.6
1961[a]	6.5	200	594	18.1
1962[a]	7.5	175	519	15.8
1963[a]	8.5	153	454	13.8
1964[a]	9.5	134	397	12.1
1965[a]	10.5	117	347	10.6
1966[a]	11.5	103	304	9.24
1967[a]	12.5	89.7	266	8.08
1968[a]	13.5	78.4	232	7.07
1969[a]	14.5	68.6	203	6.18
1970[a]	15.5	60.0	178	5.40
1971	16.5	52.5	155	4.73
1972	17.5	45.9	136	4.13
1973	18.5	40.1	119	3.61
1974	19.5	35.1	104	3.16
1975	20.5	30.7	90.9	2.76
1976	21.5	26.8	79.5	2.42
1977	22.5	23.5	69.5	2.11
1978	23.5	20.5	60.8	1.85
1979	24.5	18.0	53.2	1.62
1980	25.5	15.7	46.5	1.41
1981	26.5	13.7	40.7	1.24
1982	27.5	12.0	35.6	1.08
1983	28.5	10.5	31.1	0.946
1984	29.5	9.18	27.2	0.827
1985	30.5	8.03	23.8	0.724
1986	31.5	7.02	20.8	0.633
1987	32.5	6.14	18.2	0.553
1988	33.5	5.37	15.9	0.484
1989	34.5	4.70	13.9	0.423
1990	35.5	4.11	12.2	0.370
1991	36.5	3.59	10.6	0.324
1992	37.5	3.14	9.31	0.283
1993	38.5	2.75	8.14	0.248
1994	39.5	2.40	7.12	0.217
1995	40.5	2.10	6.23	0.189
1996	41.5	1.84	5.45	0.166
1997	42.5	1.61	4.76	0.145

Year	t (yr)	Air concentration (dpm/m^3)	Chronic inhalation intake (pCi/d)	Chronic ingestion intake (pCi/d)
1998	43.5	1.41	4.17	0.127
1999	44.5	1.23	3.64	0.111
2000	45.5	1.08	3.19	0.0969
2001	46.5	0.941	2.79	0.0847
2002	47.5	0.823	2.44	0.0741
2003	48.5	0.719	2.13	0.0648
2004	49.5	0.629	1.86	0.0567
2005	50.5	0.550	1.63	0.0496
2006	51.5	0.481	1.43	0.0434
2007	52.5	0.421	1.25	0.0379
2008	53.5	0.368	1.09	0.0332
2009	54.5	0.322	0.953	0.0290
2010	55.5	0.282	0.834	0.0254

a. Dose reconstructors should not use data from 1955 to 1970 to determine internal doses for employees affected by SEC-00075 (NIOSH 2007a).

Hokel et al. (1998) stated that the thorium in Wilhelm Hall is in equilibrium.

For estimating dose to individuals working in Wilhelm Hall from 1955 to the present, an intake estimate favorable to claimants would be as follows:

A removable surface concentration of 2,000 dpm/100 cm^2 of thorium in equilibrium with its progeny was assumed [19]. This means the ^{232}Th activity was approximately 200 dpm/100 cm^2. Applying a resuspension factor of 10^{-4} [20]:

$$\text{Air concentration} = (200\ dpm/100\ cm^2)(10^{-4}/m)(100 \times 100\ cm^2/m^2) = 2\ dpm/m^3$$

The value of 2 dpm/m^3 on November 15, 1995, when the survey of the pipe tunnel occurred, represents an upper bound.

To estimate residual air concentrations for other years, a single exponential decay of “available” activity over time in a facility was used. This applies to a facility in which the contamination is mostly undisturbed, but undergoes some weathering or occasional cleaning, maybe minor moving of equipment or work benches followed by cleaning, etc. This method of internal dose estimation does not apply to workers in the buildings or areas designated in SEC-00075 (NIOSH 2007a).

Klevin (1952) did a thorough survey during operations, including breathing-zone task-specific air concentrations, time-weighted air concentrations for various workers, and general room air concentrations. Because survey information in the building just after cessation of the thorium operations was not available, the Klevin data were used to represent air concentrations in 1955 with the caveats provided below:

1. The Klevin air concentrations for specific tasks during production and the time-weighted averages are not relevant for Wilhelm Hall after 1954. The general room concentrations would probably be a bit high, but reasonably representative when applied to long-term chronic intakes for people not actually performing tasks with thorium. These can be assumed to represent air concentrations in 1955.

2. The general air concentrations included one room that was 10 times higher than all the others and included the lunch room. The highest room was clearly related to operations so

that datum was excluded, as was the lunch room, which was assumed not to be contaminated (the result was zero). With these results removed, the geometric mean of the distribution was 7.9 dpm/m^3 and the 95th percentile was 479 dpm/m^3.

Because the 2 dpm/m^3 from Hokel et al. (1998) represented an upper bound, this value is compared with the 479 dpm/m^3 value. With 479 as A_o and 2 as A(t) and assuming t is from January 1, 1955, to November 15, 1995, for a total of 14,929 days,

$$A(t) = A_o\exp(-\lambda t). \quad \lambda = 3.7 \times 10^{-4}/d \text{ or } 0.134/yr.$$

A(t) was solved for the midpoint of each year from 1995 to the present. The air concentration was converted to intake using

$$\text{intake (pCi/d)} = (\text{air conc. dpm/m}^3)(1 \text{ pCi}/2.22 \text{ dpm})(1.2 \text{ m}^3/\text{hr})(2{,}000 \text{ hr/yr})/365\text{d/yr})$$

The estimates in Table 5-5 for ^{232}Th assume an occupancy factor of 1 (100% for a working year). This is an upper bound (constant) estimate for thorium intakes for individuals who worked in accessible areas in Wilhelm Hall from January 1955 to the present.

The dose reconstructor needs to include equal intakes of ^{228}Ra and ^{228}Th to account for the progeny radionuclides [21]. The associated ingestion intakes would be:

(0.2)(air concentration in dpm/m^3) ÷ 2.2 dpm/pCi = X pCi/d each for ^{232}Th, ^{228}Ra, and ^{228}Th.

The annual ingestion intakes are listed in Table 5-5.

Workers occasionally entered areas with higher concentrations under Radiation Work Permits to perform maintenance or remodeling. However, the spotty contamination in these areas was about 10 to 100 times greater than the value used in the calculation above [22]. Based on the rarity of exposure and the radiation protection measures used (including respiratory protection), the impacts from these possible acute intakes would have been adequately accounted for by the chronic intake scenario provided in Table 5-5 [23]. However, this assumption is not valid for employees who worked in the buildings or areas described in SEC-00075 (NIOSH 2007a).

5.3 TRITIUM EXPOSURE

The ALRR operated from February 1965 through December 1977. The reactor was then decommissioned, which was completed in 1981. The ALRR was a 5-MW, heavy-water-moderated research reactor. See Section 2.1.10 for more information on reactor operations. Because of the operation of the reactor, tritium is the primary radionuclide that must be considered for internal dose. No other radionuclides were considered internal dose hazards from the operation of the reactor. A routine bioassay program was established in 1965; bioassay data for workers were collected and recorded through at least 1981 while the reactor was being decontaminated and decommissioned (Voss 1971).

5.3.1 Tritium Exposure for Monitored Workers, 1965 to 1981

A number of reports addressed tritium dose assessment. Highlights of those reports are given here and referenced for further study by the dose reconstructor.

Program for Monitoring Personnel for Tritium, Sources of Tritium at the Ames Laboratory (Voss 1971) describes potential sources of tritium exposure to Laboratory personnel.

In addition to the reactor, Ames Laboratory had a Texas Nuclear Model 9900 Neutron Generator, which was a potential source of exposure to tritium. The neutron generator used tritium targets with activities ranging from 5 to 10 Ci. The tritium, which was slowly released from the target, was documented as a potential airborne contaminant. Other possible sources of tritium were spent targets from the neutron generator, tritium gas in cylinders, and tritiated compounds that would have been present occasionally (Voss 1971).

Voss (1971) is summarized here to help understand dose records that might be in the worker's files. It stated that workers and other personnel who were part of the tritium monitoring program included:

1. Reactor operators (monthly samples).

2. Reactor operators involved in specific operations that were likely to cause tritium exposures (sampled during and after the operation).

3. Other reactor personnel and other personnel assigned to work at the reactor (provide samples on a semiannual basis for routine checks on tritium uptake).

4. Personnel described in Item 3 who were involved in operations in which tritium exposure conditions existed (sampled during and after the operation).

5. Other personnel under conditions of potential tritium exposure (monitored on the basis of their work with sources of tritium).

6. All personnel who showed tritium concentration greater than 1,500 dpm/min/mL of urine were resampled weekly until the detected activity level was below 1,500 dpm/min/mL of urine.

7. Samples were taken at the termination of employment of persons assigned to work areas where monitoring for tritium was required.

8. As a check and control, termination samples were taken from other personnel (it is not clear if it was all personnel or just some personnel).

Voss (1971) stated that liquid scintillation counting was used for tritium bioassay analysis.

Assumptions described in Voss (1971) for radiation dose determination were:

1. The concentration of tritium was the same in all body water. Urine samples were used as representative of all body water in terms of tritium activity levels. The radiation dose was for the whole body.

2. The radiation dose was due to tritium as tritiated water; other compounds of tritium and other organ configurations are not included. Except in unusual circumstances, tritiated water was the expected source of exposure. Thus, body water was the critical organ.

3. For calculation and derivation consistency, the mass of body water was taken as 43 kg (standard man value).

4. The effective half-life for dose determination could vary. The actual half-life from the data points was used up to a half-life of 15 days. If the half-life could not be determined from the data, a value of 12 days was used. Data points that showed a half-life greater than 15 days were treated as additional separate exposures.

5. The quality factor was 1, the effective absorbed energy per disintegration was 0.006 MeV, and the half-life of tritium was 12.26 years (4.48×10^3 days).

Other considerations for assignment of dose that were discussed in Voss (1971) are:

1. The dose assignment system had two separate conditions of radiation dose delivery to the body: (a) The dose delivered by the uptake of tritium from time t = 0 until complete decay-removal (t = ∞) (additional uptakes might occur during the decay removal of the preceding uptakes), and (b) the radiation dose that would be delivered by maintenance of a fixed level of tritium in the body.

2. For anyone with a bioassay sample greater than 1,500 dpm/min/mL in urine, sampling continued until the urine sample level was less than 1,500 dpm/min/mL.

3. For any person on the routine and incident monitoring program, a basic dose assignment was made. This basic dose was that associated with the maintenance of a level of 750 dpm/min/mL of urine, which was equal to 38 mrem/yr whole body based on the methodology used at the time.

4. For single or multiple exposures, the dose assessment was made on the basis of the effective half-life applicable to the individual's data. The dose contribution from the incident exposure that would appear in the basic dose assignment was subtracted from the basic dose assignment.

5. The final reported dose was that due from an incident exposure plus the residual dose from the basic dose assignment. In summary, a person on the tritium monitoring program was assigned a basic radiation dose of 38 mrem/yr (prorated for a fraction of a year) to which was added the adjusted radiation dose contribution from all other exposures received during the year.

A number of other documents discussed tritium dose assessment, but they are similar to the information provided above. "Personnel Monitoring Program" (Ames 1974) provides the Tritium Dose Calculation Procedures used at the time.

Personnel in the tritium monitoring program should have bioassay and dose records. Tritium bioassay records were found for 1965 through 1981, although those for 1965 through 1968 appear to be incomplete.

Figures 5-4 and 5-5 show examples of what a bioassay record in a worker's file might look like. The data that can be found in this record include:

1. The year being reported.

2. The badge number of the individual (Note: It appears that an individual would retain the same badge number during employment but, once they terminated, that badge number would be reassigned to a new employee.).

3. The name of the individual.

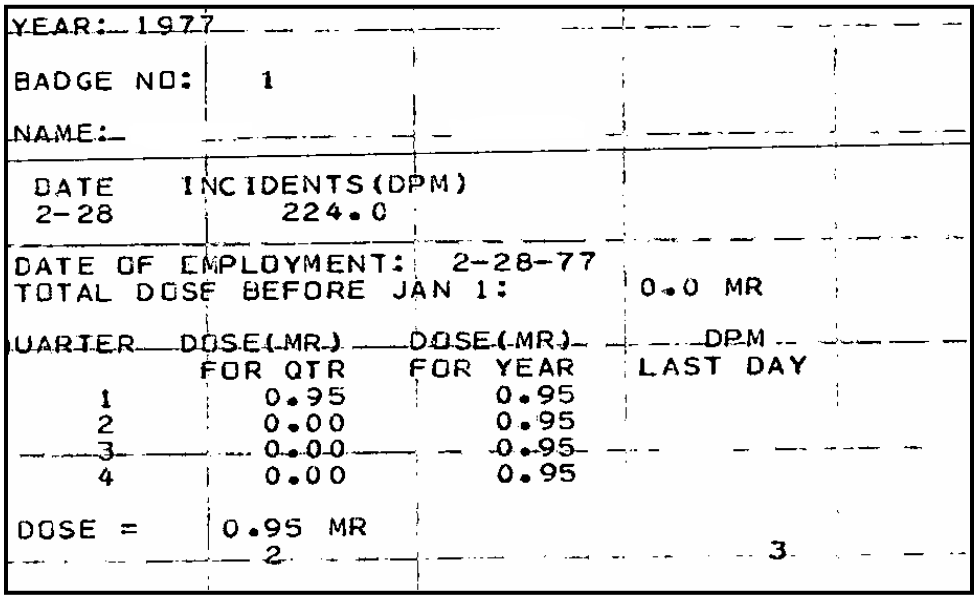
4. Dates during the reporting period for which monitoring for incidents was performed; the result was in "DPM." It is assumed that incidents were operations that were performed and sampling was based on potential for intake.

5. Date that individual was employed.

6. The individual's dose before January 1 of the year for the record.

7. If the person has terminated, the termination date.

8. The four quarters for the year. The example in Figure 5-4 is for the first quarter and shows the dose for quarter 1. For each subsequent quarter, the dose would be added and the total for the year calculated. It can be assumed that "0.00" means the worker was not monitored, and not necessarily that there was no dose.



YEAR: 1977

BADGE NO: 1

NAME:

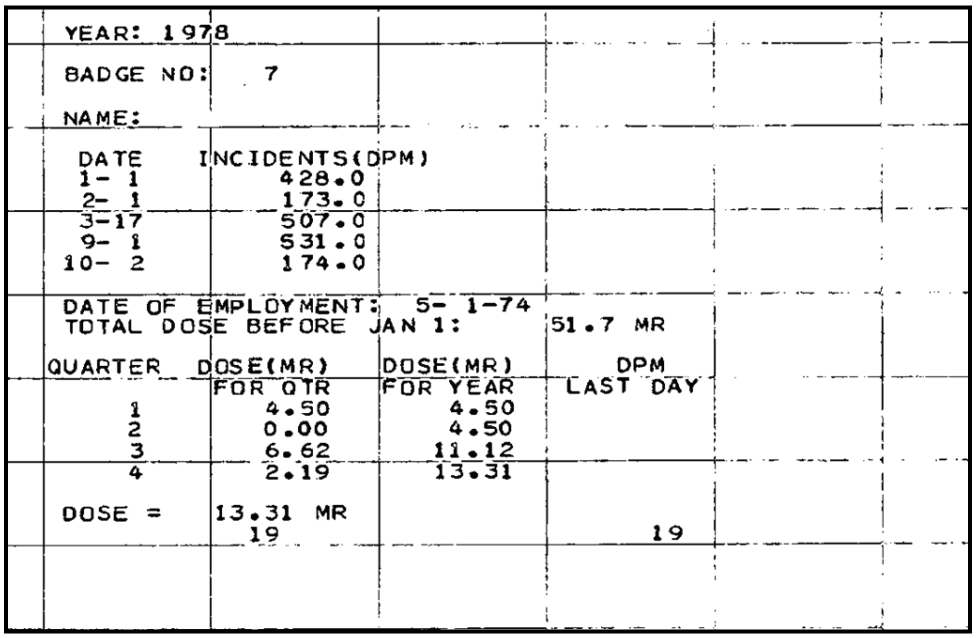
DATE	INCIDENTS(DPM)
2-28	224.0

DATE OF EMPLOYMENT: 2-28-77
TOTAL DOSE BEFORE JAN 1: 0.0 MR

QUARTER	DOSE(MR) FOR QTR	DOSE(MR) FOR YEAR	DPM LAST DAY
1	0.95	0.95	
2	0.00	0.95	
3	0.00	0.95	
4	0.00	0.95	

DOSE = 0.95 MR / 2 3

Figure 5-4. Bioassay record example 1.



YEAR: 1978

BADGE NO: 7

NAME:

DATE	INCIDENTS(DPM)
1- 1	428.0
2- 1	173.0
3-17	507.0
9- 1	531.0
10- 2	174.0

DATE OF EMPLOYMENT: 5- 1-74
TOTAL DOSE BEFORE JAN 1: 51.7 MR

QUARTER	DOSE(MR) FOR QTR	DOSE(MR) FOR YEAR	DPM LAST DAY
1	4.50	4.50	
2	0.00	4.50	
3	6.62	11.12	
4	2.19	13.31	

DOSE = 13.31 MR / 19 19

Figure 5-5. Bioassay record example 2.

Documents from the early 1970s (Voss 1971; Ames 1974) indicate that a person in the tritium monitoring program would have been assigned a default radiation dose of 38 mrem/yr (or it would have been prorated for the time monitored during the year); however, a review of the available dose

records indicated that the policy to assign a 38-mrem/yr dose to everyone on the tritium program was not always followed.

No information was found that indicated the analysis method or the minimum detectable activity (MDA); therefore, it is recommended that 0.1 µCi/L (220 dpm/mL) be used as the MDA. This is a reasonable assumption compared with MDAs at other sites during this period and is consistent with the way the data were recorded [24].

5.3.2 Tritium Dose to Workers, 1982 to Present

An interoffice memorandum (Voss 1987) dated May 8, 1987, addresses radiation dose at the Applied Science Center (inhalation exposure from tritium as tritiated water vapor in air resulting from deposition during reactor operations). The memorandum addresses the methodology for estimation of dose to an individual in the reactor room assuming 100% occupancy without forced ventilation, based on the standard man inhalation rates in BRH (1970). There is also an estimate for a maximum dose of 5.8 mrem/yr to an individual in the pump room. In a letter dated May 27, 1987, R. G. Struss, Associate Director for Operations, Ames Laboratory, discussed another estimate for airborne tritium concentrations that indicated an exposure potential of 8.6 mrem/yr in the reactor pump room for a 40-hour workweek (Struss 1987).

The dose reconstructor can consider assigning a dose of 8.6 mrem/yr for personnel working in the Applied Science Center in either the reactor room or the pump room if there are no bioassay data available.

5.3.3 Tritium Dose to Unmonitored Workers, 1965 Through 1981

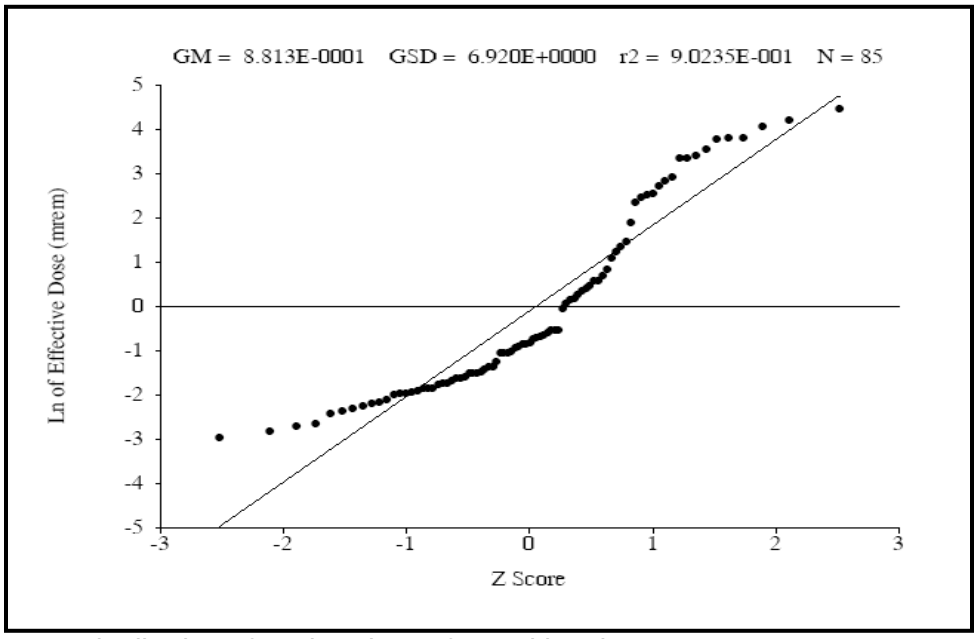
A coworker analysis of worker tritium bioassays was conducted on bioassay samples from 1965 through 1981 to provide default doses for unmonitored workers. Results of the coworker analysis are listed in Table 5-6, columns 2 and 3. If there is an indication that an Energy Employee worked in the ALRR during 1965 through 1981 but was not monitored for tritium, the annual doses listed in Table 5-6, column 4, should be assigned. These doses apply to all organs. Assume a lognormal distribution with the geometric standard deviations (GSDs) as listed in Table 5-6, column 5.

Table 5-6. Results of tritium coworker analysis [25].

	Coworker results		Assigned values	
Year	Annual dose (mrem)	GSD[a]	Annual dose (mrem)	GSD[a]
1965	0.44	1.6	0	N/A
1966	0.54	1.5	1	3
1967	0.46	1.9	0	N/A
1968	0.23	2.3	0	N/A
1969	1.9	7.9	1.9	7.9
1970	0.88	6.9	1	6.9
1971	1.3	15	1.3	15
1972	1.3	12	1.3	12
1973	1.2	18	1.2	18
1974	1.9	8.1	1.9	8.1
1975	1.3	17	1.3	17
1976	1.0	9.5	1.0	9.5
1977	1.6	25	1.6	25
1978	1.7	11	1.7	11
1979	1.0	7.5	1.0	7.5
1980	1.0	9.8	1.0	9.8
1981	1.1	4.7	1.1	4.7

a. Use a GSD of 3.0 if the calculated value is less than 3 in Table 5-6.

The tritium coworker doses were determined using methods described in *Analysis of Coworker Bioassay Data for Internal Dose Assignment* (ORAUT 2005a) and *Tritium Calculated and Missed Dose Estimates* (ORAUT 2004). Each annual dose was the geometric mean of the lognormal distribution fitted to the distribution of individual workers' annual doses. Figure 5-6 shows an example distribution and fit for 1970. Annual doses less than 1 mrem are insignificant and can be ignored when assigning doses; a minimum GSD of 3.0 was assigned if the calculated value was less than 3.



5-6. Distribution of worker doses from tritium in 1970.

5.4 FISSION PRODUCT INTAKES

5.4.1 Fission Product Intakes from Early Fuel Research

As described in Section 4.4, a hot laboratory was operated in the Chemistry Building; it was replaced in 1951 by a hot canyon/hot cell in the Research Building. In the 1940s, the hot laboratory was used to study extraction of plutonium from irradiated uranium by means of ion exchange columns.

> *A hot laboratory, capable of handling 5 curies through the adsorption process, was designed and built. The method proved successful in a number of runs. The uranium and plutonium were separated from one another and from the fission products using 1 kg samples of uranium from the pile, which had an activity of 5 curies. (Fulmer 1947)*

Intakes by workers in the hot laboratory were estimated using the approach for calculating intakes described in NUREG-1400, *Air Sampling in the Workplace* (Hickey et al. 1993). The equation from NUREG-1400 is:

$$I = Q \times 10^{-6} \times R \times C \times D$$

where

I = intake
Q = source term for 1 year = assumed to be 50 Ci/yr
R = release fraction = 0.01 for a liquid (the material comes in as a solid, but would be liquefied for research activities)
C = confinement factor = 0.1 assuming material was handled in some containment

D = dispersibility factor = 10 for heating or chemical reactions

$$I = 50 \times 10^{-6} \times 0.01 \times 0.1 \times 10 = 5 \times 10^{-7} \text{ Ci/yr} = 5 \times 10^{5} \text{ pCi/yr}$$

$$= 5 \times 10^{5} \text{ pCi/yr} / 365 \text{ d/yr} = 1{,}370 \text{ pCi/d}$$

As described in Section 4, using the recommended fission product ratios for 180-day cooled fuel from *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2007a), the 1,370-pCi/d intake of fission products is assigned to specific radionuclides, as listed in Table 5-7.

Table 5-7. Default annual fission product intakes for researchers from hot laboratory in the Chemistry Building (1943–1951).

Radionuclide	Relative fraction	Intake (pCi/d)
Ce-141	0.0221	30.3
Ce-144	0.2191	300
Cs-134	0.0054	7.4
Cs-137	0.0208	28.5
Eu-155	0.0014	1.9
Fe-55	0.0172	23.6
Nb-95	0.2492	341
Pm-147	0.0546	74.8
Ru-103	0.0321	44
Ru-106	0.0844	115.6
Sr-89	0.0558	76.4
Sr-90	0.0157	21.5
Y-91	0.0911	124.8
Zr-95	0.1311	179.6

Because maximizing assumptions were used for most of the parameters in this analysis, the distribution is an upper bound (constant). The absorption type for each radionuclide that is most favorable to claimants, as listed in ICRP Publication 68 (ICRP 1995), should be used, with the exception that only type F should be used for strontium [26]. The date of first use of the hot laboratory was not found. Because it received fuel from the Chicago Pile, the earliest reasonable use would have been 1943. The hot laboratory was replaced by a hot cell in the Research Building in 1951. Therefore, these intakes should be applied for each year from 1943 through 1951. As discussed above, SEC-00166 (NIOSH 2010b) states that intakes from the hot cell cannot be determined for radionuclides other than uranium. Due to the smaller amounts of material and the improved ventilation and containment of the Research Building over its predecessor discussed in Sections 2 and 4, respectively, workers in the Research Building who did not work with the hot cell or laboratories associated with the hot cell should be assigned environmental internal and external doses.

5.4.2 Fission Product Intakes from Research Reactor Operations and Decontamination and Decommissioning

There was no evidence found during review of any of the Ames Laboratory documents that bioassay was performed or considered necessary for radionuclides other than tritium for the ALRR. Review of *Decommissioning of the Ames Laboratory Research Reactor* (Voigt 1981) indicated that there was a negligible amount of fission products in the building after shutdown. This indicates that there was probably a negligible amount of fission products released during reactor operations; therefore, accounting for internal dose from reactor operations is not necessary.

During D&D operations, radiation protection appeared to be acceptable for the time (Voigt 1981) and external exposure and internal exposure for tritium were tracked. There is no evidence that bioassay for other radionuclides was performed. However, during D&D of the reactor, which included dismantlement, cutting, grinding, etc., it can be expected that some intakes from activation products occurred.

A reasonable estimate of intakes from D&D of the reactor can be made by using the gross beta air concentration limit from 1977 (1×10^{-9} µCi/mL) (ERDA 1977) based on the most conservative beta emitter (^{90}Sr).

For workers involved in the actual decontamination or demolition, we assumed:

$$(1 \times 10^{-9} \text{ µCi/mL}) (1 \times 10^{6} \text{ pCi/µCi}) (1 \times 10^{6} \text{ mL/m}^3) = 1{,}000 \text{ pCi/m}^3$$

Assuming an occupancy factor of 0.5 (1,000 hr/yr exposure) and a breathing rate of 1.2 m^3/hr.

$$(1{,}000 \text{ pCi/m}^3) (1.2 \text{ m}^3\text{/hr}) (1{,}000 \text{ hr/yr}) \div 365 \text{ d/yr} = 3{,}300 \text{ pCi/d}.$$

Assign this intake to the D&D workers for the appropriate years between 1978 and 1981. For supervisors and others not directly involved in the work, assign one-fourth of the intake.

5.5 SUMMARY OF INTERNAL DOSE RECOMMENDATIONS

Default occupational intakes for workers without the applicable bioassay data are summarized in Table 5-8.

Table 5-8. Default intakes for Ames Laboratory (if no bioassay results are available) [27].

Job category or task/Building	Dose calculation parameters					IREP input parameters	
	Period	Material	Mode	Absorption type or f_1	pCi/d	Distribution type	Parameter 1
Researcher/Chemistry Building[a]	Aug 1942–Dec 1953	Natural uranium	Chronic inhalation	F, M, or S	8.5	Constant	Dose
Researcher/Chemistry Building[a]	Aug 1942–Dec 1953	Natural uranium	Chronic ingestion	0.02 with F,M; 0.002 with S	0.09	Constant	Dose
All employees in Chemistry Building	Jan 1954–May 1976	Natural uranium	Chronic Inhalation	F, M, or S	4.1	Constant	Dose
All employees in Chemistry Building	Jan 1954–May 1976	Natural uranium	Chronic Ingestion	0.02 with F,M; 0.002 with S	0.68	Constant	Dose
Researcher in hot lab/Chemistry Building	1943–1951	Fission products	Chronic inhalation	Use most favorable to the claimant; for Sr-90 use F	See Table 5-7	Constant	Dose
Researcher, production technician, anyone involved daily with uranium in Annex 1[a]	Aug 1942–Aug 1945	Natural uranium	Chronic inhalation	F, M, or S	853	Constant	Dose
Researcher, production technician, anyone involved daily with uranium in Annex 1[a]	Aug 1942–Aug 1945	Natural uranium	Chronic ingestion	0.02 with F,M; 0.002 with S	8.7	Constant	Dose
All employees in Annex 1	Sep 1945–Dec 1953	Natural uranium	Chronic inhalation	F, M, or S	17.5	Constant	Dose
All employees in Annex 1	Sept 1945–Dec 1953	Natural uranium	Chronic ingestion	0.02 with F,M; 0.002 with S	1.6	Constant	Dose
Researcher, production technician, anyone involved daily with uranium in Annex 2[a]	Jan 1944–Dec 1953	Natural uranium	Chronic inhalation	F, M, or S	6,061 through 1950, 5,556 from 1951 to 1953	Constant	Dose
Researcher, production technician, anyone involved daily with uranium in Annex 2[a]	Jan 1944–Dec 1953	Natural uranium	Chronic ingestion	0.02 with F,M; 0.002 with S	124 through 1950; 114 from 1951 to 1953	Constant	Dose
All employees in Annex 2	Jan 1954–1972	Natural uranium	Chronic inhalation	F, M, or S	124.7	Constant	Dose
All employees in Annex 2	Jan 1954–1972	Natural uranium	Chronic ingestion	0.02 with F,M; 0.002 with S	11.2	Constant	Dose
Anyone routinely in Wilhelm Hall (Metallurgy Building)	1955–present	Th-232	Chronic inhalation	M or S	See Table 5-5	Constant	Dose
Anyone routinely in Wilhelm Hall	1955–present	Th-232	Chronic ingestion	5E-4 with M; 2E-4 with S	See Table 5-5	Constant	Dose
Anyone routinely in Wilhelm Hall	1955–present	Ra-228	Chronic inhalation	M	See Table 5-5	Constant	Dose
Anyone routinely in Wilhelm Hall	1955–present	Ra-228	Chronic ingestion	0.2	See Table 5-5	Constant	Dose

Job category or task/Building	Dose calculation parameters					IREP input parameters	
	Period	Material	Mode	Absorption type or f_1	pCi/d	Distribution type	Parameter 1
Anyone routinely in Wilhelm Hall	1955–present	Th-228	Chronic inhalation	M or S	See Table 5-5	Constant	Dose
Anyone routinely in Wilhelm Hall	1955–present	Th-228	Chronic ingestion	5E-4 with M; 2E-4 with S	See Table 5-5	Constant	Dose
Reactor workers and D&D workers	1965–1981	Tritium	Total of all modes	N/A	N/A	Lognormal, see Table 5-6 for GSD	Dose; see Table 5-6
Workers involved with D&D of reactor, including former reactor workers[a]	1978–1981	Co-60 or Zn-65	Chronic inhalation	Type S	3,300	Constant	Dose
Workers involved with D&D of reactor, including former reactor workers[a]	1978–1981	Co-60 or Zn-65	Chronic ingestion	Type S	660	Constant	Dose
Anyone routinely in Applied Science Center	1982–present	Tritium	Total of all modes	N/A	N/A	Constant	8.6 mrem/yr

a. Values are for workers assumed to work in research or production full time. For supervisors, assume 0.25 of the intake; for all other employees (clerical, janitorial, security, etc) assume one-tenth of the supervisor's intake (0.025 of the intake in the table).

6.0 OCCUPATIONAL EXTERNAL DOSE

Workers at Ames Laboratory received external radiation doses between 1942 and 1952 that were largely unmonitored [28]. Pocket chambers were available that might have been used to monitor external doses, but very few records could be found [29]. External exposures since 1953 have been monitored with film badges and thermoluminescent dosimeters (TLDs), and the records are essentially complete. However, there were extensive periods (1965 through 1981) when individual names were not recorded with dosimeter numbers and many records cannot be clearly associated with individual workers [30]. For cases in which the individual's name is clearly associated with dosimetry records and the records are essentially complete, these individuals are considered monitored workers. For all other cases in which the records are unidentified or partially complete, the individuals are considered unmonitored workers for the missing or incomplete periods. External doses received since 1982 have been reliably recorded for each individual. Details of the external dosimetry program are addressed in this section.

6.1 INTRODUCTION

Ames Laboratory responses to DOL requests for claimant records were limited to raw data from dosimetry files. Not all of these files had been summarized for annual total doses for individuals and the files for some individuals were incomplete, as noted above. To assist dose reconstructors in determining annual doses for claimants, the raw data were entered into Excel® spreadsheets and summarized to give annual totals for identified individuals; the spreadsheet data were analyzed to give 50th- and 95th-percentile values that could be applied to unidentified (unmonitored) individuals (Martin 2006c).

Dose reconstructors should have access to dosimetry records from Ames Laboratory for each claimant, but the records might be incomplete and difficult to use. The spreadsheets assembled by Martin (2006c) should be used if there is any doubt about completeness.

As described in Section 1.3, some portions of external dose cannot be adequately estimated for certain workers from August 13, 1942, through December 31, 1954 (NIOSH 2006).

6.2 EXTERNAL RADIATION DOSIMETERS AND RECORDS

An AEC survey conducted at Ames Laboratory from March 18 to 21, 1952, found the personnel monitoring program to be less than adequate. A number of recommendations were made in the survey report to promote improvements in radiation protection, including film badge service and regular radiation monitoring (Hokel et al. 1998). Before this time, only pocket chambers (pencil dosimeters) were used to monitor radiation exposure, and records were incomplete (Voss 1949). Regular film badge service was started in February 1953; beta/gamma film badges were exchanged on a biweekly frequency with the results tabulated monthly and summarized annually (Ames 1954a). Dosimetry services evolved and improved over the following 50 years; the characteristics of the dosimeters used are summarized in Table 6-1.

6.2.1 Historical Administrative Practices

Some of the early administrative practices at Ames Laboratory related to dosimetry recordkeeping were unusual and inconsistent with current practice. The recordkeeping practices that correspond to each step in the evolution of dosimetry services (Table 6-1) are addressed in this section.

Pocket chambers (pencil dosimeters) were used at Ames Laboratory at various times, possibly as early as 1942, through 1952 when film badge service was begun. However, the records of pocket chamber results that were found apply only to a brief period in February 1949 and only for a few

Table 6-1. Dosimeter type, period of use, exchange frequency, MRD, and MDL.

Dosimeter type, provider	Period	Exchange frequency[a]	MRD (mrem)[b]			MDL (mrem)[c]		
			Skin	β/γ deep	Neutron	Skin	Deep	Neutron
Pocket Chambers, Ames Laboratory	1942–1952[d]	Daily		5			5	
β/γ film, Ames Laboratory in-house system	Feb 1953–May 1953	Biweekly		10		40	40	
	June 1953–Dec 1953			25		40	40	
	Jan 1954–Feb 1957		25	25		40	40	
	Mar 1957–Dec 1961		10	10		40	40	
β/γ/NTA film, BNL	Apr 1954–June 1957	Biweekly	15	15	10	40	40	(e)
β/γ/NTA film, NCA	July 1957–June 1963	Biweekly	10	10	10	40	40	(e)
β/γ/NTA film, Atomic Film Badge Corporation	July 1963–Mar 1965	Monthly	10	10	10	40	40	(e)
Pocket Chambers, Ames Laboratory	Apr 1965–June 1965	Daily		5			5	
β/γ/NTA film, Health Physics Services	July 1965–Oct 1979	Monthly	10	5	28	40	40	(e)
β/γ film, Health Physics Services	Nov 1979–Sept 1981	Monthly	10	5		40	40	
β/γ TLDs, Landauer	June 1980–Dec 1981	Monthly & quarterly	40	10		30	30	
β/γ TLDs, Landauer	Jan 1982–Dec 1994	Quarterly	40	10		30	30	
β/γ TLDs, Siemens	Jan 1995–June 1996	Quarterly	10	10		30	30	
β/γ TLDs, ICN Dosimetry Service	July 1996–Sept 1998	Quarterly	10	10		30	30	
β/γ TLDs, Landauer	Oct 1998–Dec 2004	Quarterly	40	10		30	30	
β/γ TLDs, Global Dosimetry	Jan 2005–present	Quarterly	10	10		30	30	

a. The exchange frequency was established from dosimetry reports.
b. MRD = minimum recordable dose; based on minimum doses recorded on dosimetry reports.
c. Estimated minimum detection level (MDL) typical of film dosimeter capabilities (Wilson 1960, 1987; NIOSH 1993; NRC 1989; Wilson et al. 1990).
d. Exposures from August 13, 1942, through December 31, 1952, that resulted from thorium, its progeny, and plutonium cannot be reconstructed for workers who worked in buildings or areas included in SEC-00038 (NIOSH 2006). However, external dose from uranium can be estimated from 1942 to 1953 (Section 6.3.1).
e. For years of NTA film use, between 1954 and 1979, the adjusted neutron dose is calculated using correction factor of 2.

individuals (Voss 1949). Thus, it is only possible to estimate external doses before 1953. External doses at Ames Laboratory before 1953 are discussed in Section 6.3.1.

An upper bound of the total external exposures cannot be made for SEC-00038 workers because external doses from beta and gamma radiation resulting from exposure to thorium and its daughters or plutonium cannot be reconstructed due to a lack of information on the percentage of thorium daughter in-growth (up to 1954, when thorium operations ended) (NIOSH 2006). However, external dose from potential exposure to uranium from 1942 to 1953 can be estimated.

An in-house film badge system was established at Ames Laboratory in the fall of 1952 in response to recommendations from the AEC. Weekly film badges were provided to a few individuals who worked at the synchrotron from September 1952 through March 1953, but records are incomplete (Ames 1954a). Regular film badge service began in February 1953 for Laboratory staff. Beta/gamma film badges were exchanged every 2 weeks and results were summarized monthly. However, between late September 1953 and early January 1954, there were three 4-week and one 3-week exchange periods (Ames 1954a). The in-house film badge system had variable exchange periods during 1954, 1957, 1958, 1959, and 1961. There was a 4-week exchange period in January, a 3-week period in February, and a 4-week period in December 1954; otherwise, the exchange periods were 2 weeks

each (Ames 1954b, 1955). The exchange frequency during 1955 and 1956 was every 2 weeks, with no exceptions (Voss 1956a,b). There was a 4-week exchange period in May and June 1957; all other exchange periods in 1957 were 2 weeks (Ames 1958). There was a 3-week exchange period in December 1958 and January 1959; all other exchange periods in 1958 and 1959 were 2 weeks (Ames 1959, 1960b). The exchange frequency during 1960 was every 2 weeks without exception (Ames 1961b). During 1961, there were nine 3-week exchange periods and 12 2-week periods (Ames 1962b).

Initially, the minimum recordable dose (MRD) was 10 mR, and no distinction was made between beta and gamma doses (recorded nonzero doses were assumed to be gamma doses). As of June 1953, the MRD was reported as 25 mR. In 1954, the MRD was specified as 25 mrep and 25 mR for beta and gamma, respectively. In March 1957, the MRD decreased to 10 mR for both beta and gamma doses; this MRD was unchanged through 1961 (Ames 1957b, 1958, 1959, 1960b, 1961b, 1962b).

Between October 24 and November 21, 1953, unusually high beta dose readings (about 700 mR) were “caused by the film being left unprotected near X-ray radiation” (Ames 1954a). Corrections that are favorable to claimants for this exposure that was not received by personnel are recommended in Martin (2006c).

In the fourth quarter of 1953 and continuing through 1961, beta and gamma doses were reported separately, but were added to give a total dose (Ames 1954a). Between 1954 and 1961, neutron doses were added to the beta and gamma doses to give a total dose (Ames 1954a,b, 1955, 1956a,b, 1958, 1959, 1960b, 1961b, 1962b).

In the annual summaries for 1953 through 1956 (and for January and February 1957), an assumed dose of 25-mR gamma was assigned for each month in which the dosimeter reading was zero or less than the MRD (Voss 1954a,b, 1955, 1956a,b,). From March 1957 through 1961, when the dosimeter readings were zero or less than the MRD, an assumed dose of 10-mR gamma was assigned for each dosimeter exchange period (Ames 1958, 1959, 1960b, 1961b, 1962b). This practice of assigning doses equal to the MRD is roughly equivalent to the standard method for correction of missed dose for monitored workers (NIOSH 2007b). Therefore, no additional correction to the gamma dose is required for monitored workers for the period from 1953 through 1961.

Beginning in April 1954 and continuing through June 1957, a beta/gamma/neutron film badge service was provided by BNL for the synchrotron staff (Ames 1954b, 1955). The film badges were exchanged every 2 weeks and results were included in the annual dose summaries. The MRD was 15 mrep and 15 mR for beta and gamma, respectively. Neutron dose was reported as the number of recoil proton tracks in the open window, and the number of tracks was multiplied by 10 to give the neutron dose in millirem (Ames 1954b, 1955, 1962b).

Between July 1957 and June 1963, beta/gamma/neutron film badge service was provided by the Nucleonic Corporation of America (NCA) for the synchrotron staff (Ames 1958). The film badges were exchanged every 2 weeks and results were included in the annual dose summaries. The MRDs were 10 mrad and 10 mrem for beta and gamma, respectively. Neutron dose was reported as the number of recoil proton tracks in the open window, and the number of tracks was multiplied by 10 to give the neutron dose in millirem (Ames 1958).

The data discussed above for the period from 1953 through 1961 were compiled on a spreadsheet and the inconsistencies were corrected or minimized (Martin 2006c). For example, if beta, gamma, and neutron doses were reported separately but added together for a total dose, the separated beta, gamma, and neutron doses were entered in the spreadsheet to facilitate analysis. For cases in which data were questionable, assumptions were made that would be favorable to claimants [31].

Dosimetry recordkeeping from January 1962 through June 1963 was less than adequate. The only records found for 1962 were some calibration data from April and the annual summary data (Ames 1963). It was assumed that the in-house film badge system continued through December 1962 with badges exchanged on a biweekly frequency; however, no records could be found to confirm this assumption. Monthly film badge service was provided for all Ames Laboratory personnel with the potential for radiation exposure by the Atomic Film Badge Corporation from July 1963 through March 1965. No records were found for the first half of 1963. The annual summary for 1963 reflects results from July through December 1963 only; therefore, all Ames Laboratory workers are assumed to have been unmonitored for the first half of 1963 (Ames 1964). The annual summary for 1964 accurately reflected the sum of the monthly dosimeter readings during 1964 (Voss 1965).

Dosimetry records for January through March 1965 appeared to be reliable; however, the Atomic Film Badge Corporation defaulted on its contract and went out of business in April 1965 (Mattmueller 1965). Ames Laboratory staff members were monitored with pocket chambers on a daily basis and records were compiled in-house from April through June 1965 (Ames 1966). Film badge service was provided by Health Physics Services from July 1965 through September 1981. The monthly service included beta, gamma, and neutron film badges for reactor and synchrotron staff members and beta/gamma film badges for the remaining Laboratory personnel (Ames 1966). The MRDs were 10 mrad for beta, 5 mR for gamma, and 28 mrem for neutrons (Ames 1966).

TLD services for beta and gamma dosimetry were provided by Landauer from June 1980 through June 1995. Similar services have been provided in subsequent years by Siemens, ICN Dosimetry Service, and Landauer. The MRD for TLD services was 10 mrem for beta and gamma, except Landauer specified an MRD of 40 mrem for hard beta (greater than 1.5 MeV) (Landauer 1982; Siemens 1996; ICN 1997). Dosimetry records compiled since 1981 are considered to be complete and reliable. Records for monitored workers include an identification number, name, Social Security Number, and recorded doses for the current period, calendar quarter, and calendar year. The TLD services used by Ames Laboratory have been accredited by the National Voluntary Laboratory Accreditation Program (NVLAP) since 1985 (Landauer 1985).

6.2.2 Dosimetry Technology

Ames Laboratory used film badge dosimeters from 1952 through 1981; TLDs have been used since 1980. The initial beta/gamma film badge system was operated in-house from 1952 to 1962. The film was Kodak type K and the developer was Kodak D-19. However, no records could be found that described the film holder, although it seems likely that one of several commercially available holders would have been procured, and that such a holder would have had standard design features as described in AEC (1955).

The initial neutron dosimetry service used at Ames Laboratory, which began in 1954 and included beta and gamma dosimetry, was provided by BNL. The film badge holder was the basic Oak Ridge National Laboratory (ORNL) multielement dosimeter; the neutron film was nuclear track emulsion type A (NTA) (ORAUT 2006b). Similar film badge holders and film were provided by NCA, Atomic Film Badge Corporation, and Health Physics Services for the periods listed in Table 6-1.

With the termination of operations at the synchrotron in 1971 and the removal of fuel from the ALRR in October 1979, neutron dosimetry services were no longer needed after 1979 [32]. From 1980 to the present, beta/gamma TLD service has been provided by Siemens, ICN Dosimetry Service, and Landauer for the periods listed in Table 6-1. These services have been accredited by NVLAP since 1985 (Landauer 1985).

6.2.2.1 Beta/Photon Dosimeters

Figure 6-1 shows the response of a film badge to photon radiation of different energies; it also shows the *Hp(10)* response. The figure shows two responses for film badges: one for a sensitive DuPont 502 emulsion in a two-element badge (Pardue, Goldstein, and Wollan 1944) and one for a sensitive DuPont 555 emulsion in the multielement badge (Thornton, Davis, and Gupton 1961). The response of the sensitive Eastman Type 2 film in a multielement film badge is similar to that of the sensitive DuPont 555 emulsion. The film badges show an over-response at photon energies around 100 keV, due primarily to relatively (compared with tissue) high atomic numbers [silver (47) and bromine (35)] in the film emulsions. The two-element film badges under-respond to lower energy photons; the multielement film badge typically over-responds to photons between 50 and 150 keV.

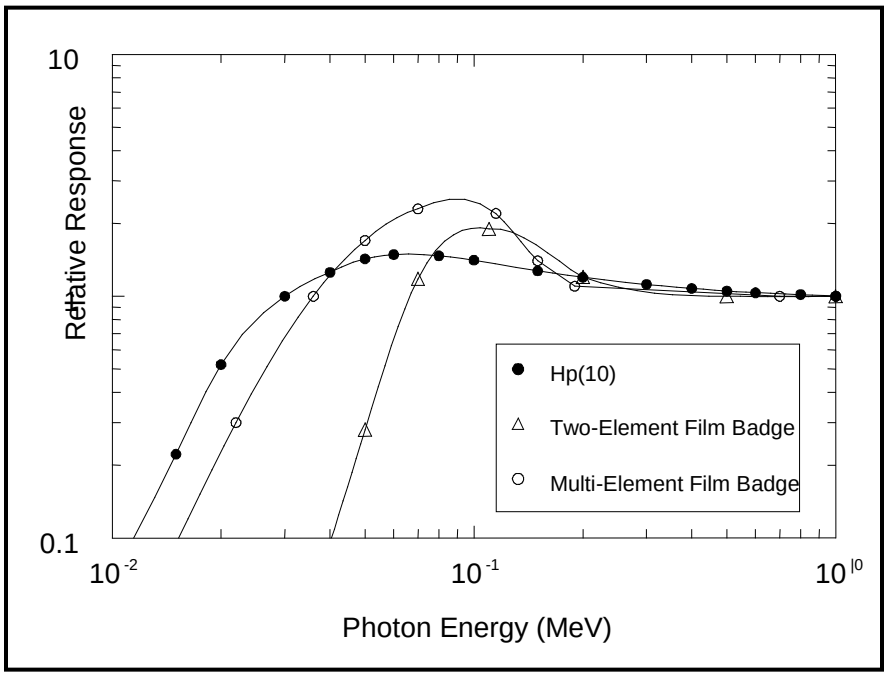


Figure 6-1. Comparison of *Hp(10)* for photons with energy responses for sensitive DuPont 502 emulsion in two-element film badge (Pardue, Goldstein, and Wollan 1944) and sensitive DuPont 555 emulsion in multielement film badge (Thornton, Davis, and Gupton 1961).

6.2.2.2 Neutron Dosimeters

NTA film was used by the dosimetry services provided to Ames Laboratory from 1954 through 1979. In general, the response of the NTA film decreases with decreasing neutron energies greater than a minimum threshold energy estimated to be between 500 and 800 keV (IAEA 1990; ORAUT 2006b). An unknown fraction of the total neutron dose was received from neutrons with energies less than about 800 keV.

6.2.3 Dosimeter Calibration Procedures

Calibration procedures for beta/photon dosimeters were followed consistently from 1953 to 1981, with the exception of 1962 and the first half of 1963 [33]. The suppliers of neutron dosimeters were generally relied on for neutron calibrations.

6.2.3.1 Beta/Photon Dosimeters

A set of calibration control film badges was exposed and processed with each biweekly set of films used in the Ames Laboratory in-house film badge system from 1953 through 1961 (Ames 1974). The beta calibration was accomplished by the exposure of films for varying times to a milled uranium plate to produce beta doses between 10 and 1,000 mR. The gamma calibration was done by the exposure of films to doses of gamma radiation from a 10-mg ^{226}Ra source that ranged between 10 and 1,000 mR (Ames 1954b, 1955). The interpretation of film density into exposure units was made from curves drawn for the calibration films (Ames 1974).

Between January 1962 and June 1963, there was evidence of calibrations done only during April 1962 (Ames 1963). From June 1963 to March 1965, when the Atomic Film Badge Corporation provided film badge services, monthly gamma calibrations were done by the exposure of films to doses of gamma radiation from a 10-mg ^{226}Ra source that ranged from 100 to 10,000 mR (Ames 1964, 1965). There was no evidence of beta or neutron calibrations during this period.

From July 1965 through September 1981, when film badge services were provided by Health Physics Services, a set of calibration control film badges were exposed to beta and gamma sources with each biweekly set of personnel film badges (Ames 1966, 1971, 1974; Ames/HPS 1978, 1979, 1981). The calibration procedures were the same as those used during the period from 1953 through 1961.

Between 1980 and 1984, when TLD services were provided by Landauer, there is no evidence of any dosimeter calibrations by Ames Laboratory. After 1985, NVLAP accreditation was relied on for dosimeter calibrations.

6.2.3.2 Neutron Dosimeters

From April 1954 through June 1957, when film badge services were provided by BNL, a set of four calibration badges was included in each monthly set of films (Voss Ames 1954b, 1955, 1956a,b). The calibration badges were exposed to known doses of beta, gamma, and neutron radiation. The beta exposures were to a milled uranium plate, the gamma exposures were to a 10-mg ^{226}Ra source, but records of the source of neutron exposures were not found (Voss Ames 1954b, 1955, 1956a,b).

From July 1957 through June 1963, when NCA provided film badge services, a set of area monitor film badges was included with each month's set of personnel dosimeters (Ames 1958, 1959, 1960b, 1961b, 1962b, 1963, 1964). The monthly readings on area monitor film badges were used to check on the validity of personnel dosimeter readings, but actual calibrations were not performed by Ames Laboratory. NCA calibrated its NTA film badges by exposing films for 40 hours to a fast neutron flux of 18 n/cm^2/s (2.6×10^6 n/cm^2/40 hr), which produced approximately 10 recoil proton tracks in 25 fields or an exposure equivalent to 100 mrem (Ames 1961b). From June 1963 through March 1965, when the Atomic Film Badge Corporation provided film badge services, there was no evidence of neutron calibrations.

Between July 1965 and September 1981, when Health Physics Services provided film badge services, a set of calibration control film badges were exposed to beta and gamma sources with each biweekly set of films (Ames 1974). However, no evidence could be found that neutron exposures were included in this calibration procedure.

6.2.4 Workplace Radiation Fields

Workplace radiation fields at Ames Laboratory rose from the production of uranium and thorium metal, the operation of the synchrotron and ALRR, and research activities in laboratories. With few exceptions, the following sections show that, for external dose reconstruction, all beta radiation fields

were greater than 15 keV, all photon radiation fields were between 30 and 250 keV, and all neutron fields were between 0.1 and 2 MeV [34]. Assuming that 100% of the radiation fields are within these ranges is a simplifying conservative assumption that is generally favorable to the claimant.

6.2.4.1 Beta Radiation

The major sources of beta radiation at Ames Laboratory between 1942 and 1954 were the large quantities of uranium and thorium metal produced. The uranium handled is assumed to have been in equilibrium with its ^{234}Th (24.1-day half-life) and ^{234m}Pa decay products (DOE 2001, p. 2-17). From an external dose standpoint, the most significant beta radiation emitted from uranium metal is that from ^{234m}Pa, with a maximum energy of 2.29 MeV (Shleien, Slaback, and Birky 1998). The beta dose rate (with 7 mg/cm^2 filtration) is 233 and 179 mrad/hr at the surface of a uranium slab and UF_4, respectively (BRH 1970). Thorium, if recently separated from its ^{228}Ra and subsequent decay products, would have very little beta emission. However, if ThF_4 was stored for several years before being processed to thorium metal, beta radiation from ^{228}Ra and ^{228}Ac would be measurable (Shleien, Slaback, and Birky 1998).

A large number of beta-emitting radioisotopes were handled in research work at Ames Laboratory throughout its history [35]. However, the activities of these sources were relatively small; the sources had a much smaller potential for beta radiation exposure to workers in comparison with the beta dose from uranium metal. With the exception of tritium, the energy of all beta sources was assumed to be greater than 15 keV; beta doses would have been readily measured in the open-window portion of the film badges used at the Laboratory. Tritium exposures that occurred at the ALRR were monitored by urine bioassay, which is addressed in Section 5 of this site profile.

6.2.4.2 Photon Radiation

Bremsstrahlung radiation doses from uranium metal ingots were calculated by Anderson and Hertel (2005). Contact photon dose rates of a few milliroentgens per hour were calculated for typical plates, billets, and cylindrical ingots of the type produced at Ames Laboratory. Another source of photon radiation in the early years was from stored thorium in the form of ThF_4. A gamma dose rate of 22 mR/hr in a thorium storage area was reported by Klevin (1952). Exposures to photon radiation were possible in the experimental areas at the synchrotron and ALRR, and near the many gamma-emitting radioisotopes that were handled in research work. A hot laboratory was located in the Chemistry Building and a hot canyon/hot cell was located in the Research Building for work with more intense gamma sources. Photon sources had energies in the 30-to-250-keV and greater-than-250-keV ranges. X-ray diffraction machines could have operated at energies less than 30 kV, and measured exposures probably would have been reported as beta dose. Limited data do not permit an accurate estimate of the fraction of photon exposures expected in these energy ranges. Assuming that 100% of the photons were in the 30-to-250-keV range is a simplifying conservative assumption that is generally favorable to claimants [36]. Photon radiation in the workplace could have been readily measured at Ames Laboratory with available dosimeter technology since 1953.

6.2.4.3 Neutron Radiation

The work involving plutonium and the research associated with thorium and uranium in the laboratories had the highest potential for exposure to neutrons. Neutron doses cannot be reconstructed for the SEC-00038 class of employees from 1942 through 1954 (NIOSH 2006). As stated in Section 1.3, this SEC class has been expanded to include all Ames employees, contractors, and subcontractors.

The primary sources of neutrons at Ames Laboratory were the synchrotron, which operated from 1949 through 1971, and the ALRR, which operated from 1965 through 1977. In both facilities, fast,

intermediate, and thermal neutrons would have been present in the workplace [37]. Unfortunately, no records could be found that characterized the neutron spectrum in either facility, and there were no records of neutron surveys around experimental areas or penetrations in the shielding. The only documented evidence of neutrons in the workplace is the many nonzero fast neutron doses measured with NTA film worn by personnel at each facility (Martin 2006c). Assuming that 100% of the neutrons were in the 0.1-to-2-MeV energy range is a simplifying conservative assumption that is generally favorable to claimants.

6.2.5 Dosimeter Response to Radiation Fields

The personnel dosimeters used at Ames Laboratory were properly selected for the radiation fields encountered in the workplace. The response of each of the dosimeters used is described in the following sections.

6.2.5.1 Beta/Photon Film Dosimeter Response

Ames Laboratory used film badges for beta and photon dosimetry from 1952 to 1981. An in-house film badge system was used from 1952 to 1962; film badges were provided by four commercial services from 1954 to 1981 [38]. The dosimeters provided an open window with little filtration, a lower energy window that allowed beta particles and lower energy photons to enter a film area with a plastic filter, and a film area with a metal (usually aluminum) filter (Ames 1974). The open window enabled measurement of beta particles and lower energy photons. The plastic filter enabled measurement of intermediate energy photons, and the metal filter enabled measurement of higher energy photons (1-cm depth).

The beta/photon dosimeters were calibrated regularly by methods consistent with accepted practice, but there was no evidence of any formal dosimeter response testing by an independent entity. The recorded beta and gamma doses appear to be consistent with that expected from the source terms and personnel monitoring described in Section 2.4. The recorded beta and gamma doses can be considered reliable for dose reconstruction [39].

6.2.5.2 Beta/Photon TLD Response

Ames Laboratory has used TLDs for beta and photon dosimetry since 1980. The TLDs, their analyses, and recordkeeping have been provided by reliable commercial services. The dosimetry services have been accredited by NVLAP since 1985 (Landauer 1985).

6.2.5.3 Neutron Dosimeter Response

The neutron doses received by synchrotron staff members measured by NTA film appeared to be consistent with similar accelerator operations at other sites. The neutron doses received by ALRR personnel were generally higher than those experienced at other sites (e.g., the High Flux Beam Reactor at BNL) (ORAUT 2010a). The higher neutron doses at ALRR might have been due to less than adequate management of neutron shielding in the experimental areas [40]. Fast, intermediate, and thermal neutrons were present in both Ames Laboratory facilities, but no records were found that characterized the neutron energy spectrum at either facility. Thus, there is an unknown fraction of the total neutron dose (due to neutrons with energies below about 800 keV) that was not measured by NTA film.

To correct the measured neutron doses for the unmeasured fraction of neutrons with energies below 800 keV, data from similar facilities were examined. The 200-MeV electron synchrotron at the Stanford Linear Accelerator Center (SLAC) was determined to be similar to the 80-MeV synchrotron at Ames Laboratory in relation to neutron exposures (ORAUT 2007a). The Materials Test Reactor

(MTR) at the Idaho National Laboratory (INL) was determined to be similar to the ALRR with respect to neutron exposures (ORAUT 2010c). Neutron dosimeters with NTA film were used at both facilities and neutron spectral data are available for estimation of the correction factor for missed neutron dose.

Accelerator operations at SLAC started in 1966; a multielement dosimeter was used until 1971 for beta/gamma and neutron (NTA film) dosimetry. ORAUT (2006c) concluded that all SLAC neutron doses measured with NTA film should be multiplied by 1.53 ±0.14 to account for the unmeasured neutrons with energies below 800 keV. The SLAC and Ames Laboratory synchrotrons were similar but their neutron spectra were not identical [41]. To allow for differences in the neutron spectra and other variables, a neutron dose correction factor of 2 is recommended for Ames synchrotron workers [42].

The MTR at INL, which operated from 1952 to 1970, was similar to the ALRR in that it was fueled with enriched uranium and it had a power level of 30 MW (ORAUT 2010c). The MTR was water-cooled and light-water-moderated, whereas the ALRR was heavy-water-moderated. Both reactors had ports where fast neutron beams could be extracted from the core and directed to accessible experimental areas. Both were used to study reactor fuels and structural materials. Neutron doses at both reactors were monitored with multielement dosimeters including NTA film. The neutron spectrum in experimental areas at the MTR was measured with Bonner spheres in 1961 (Hankins 1961). The data from these measurements were reanalyzed in ORAUT (2010a). Measurements made at 22 locations at the MTR were analyzed in relation to the response of NTA film. It was determined, on average for the 22 locations, that 52% of the total neutron dose would have been detected by NTA film and 48% would have been below the 800-keV threshold and undetected (ORAUT 2006c). Because the MTR and the ALRR had very similar neutron spectra, a neutron dose correction factor of 2 is recommended for ALRR workers.

6.2.5.4 Neutron Dose Weighting Factor

At Ames Laboratory, neutron dosimeter measurements were based on fluence-to-dose conversion factors and quality factors similar to those from ICRP Publication 21 (ICRP 1973) and NCRP Report 38 (NCRP 1971). It is necessary to adjust the neutron dose to account for the change in neutron quality factors between historical and current scientific guidance, as discussed in NIOSH (2006). Table 6-2, which is from ORAUT (2006d), shows the correction factor to use.

Table 6-2. Neutron dose energies, percentages, and associated ICRP (1991) correction factors.

Process description	Neutron energy (MeV)	Default dose fraction[a] (%)	ICRP (1991)/NCRP (1971) correction factor
Neutron exposures associated with synchrotron and research reactor activities	0.1–2	100	1.91

a. The assumption that all neutron energies are between 0.1 and 2 MeV is favorable to the claimant.

6.3 RECOMMENDATIONS FOR AMES LABORATORY WORKER EXTERNAL DOSE RECONSTRUCTION

Dose reconstruction for Ames Laboratory workers is based on the information above, which requires assessment of additional dose to be added to the measured photon dose from three primary causes:

- Adjustments to measured photon dose for dosimeter uncertainty
- Adjustments to measured neutron dose using a correction factor to account for neutrons with energies less than 800 keV that were not measured by NTA film
- Multiplication of the adjusted neutron dose by an ICRP (1991) neutron weighting factor adjustment of 1.91 for neutron energies between 0.1 and 2 MeV

6.3.1 Unmonitored External Dose

At Ames Laboratory, the concept of *unmonitored worker* will have to be expanded to include *monitored but records not found* [43]. Unmonitored doses were determined in two periods by two different methods. Previous work on exposures to workers at AWE uranium facilities (Battelle 2011, Christofano and Harris 1960, and ORAUT 2010d) was used to calculate unmonitored doses before 1953, while coworker dose based on a review of Ames Laboratory dosimetry data was used to construct coworker doses beginning in 1952.

For unmonitored doses in 1952 and 1953 (for which both methods provide unmonitored doses), the dose reconstructor should choose the method that best represents the employee's exposure based on work location and job description.

6.3.1.1 Unmonitored Doses Before 1953

Before the use of film badges, which began in 1953, external doses at Ames Laboratory were monitored using pocket dosimeters, but not generally recorded. Minimal records were found of these pocket dosimeter results (Voss 1949). To estimate reasonable external doses from the uranium processes before 1953, the methods described in three documents (Battelle 2011, Christofano and Harris 1960, and ORAUT 2010d) were used. These documents are representative of the potential external exposures encountered at Ames Laboratory because the processes developed there were similar to processes at AWE sites.

Two processes were considered to be the most similar to the processes in use at Ames Laboratory; Scrap Recovery in Battelle (2011, Section 2.1) is similar to processes in use in Physical Chemistry Annex 2, and Metal Reduction in Christofano and Harris (1960) is similar to processes in Physical Chemistry Annex 1 and the Chemistry Building.

The operation and processes of these three facilities are described in Section 2. However, there is not enough information to develop specific exposure potential parameters to the occupations in these facilities from 1942 through 1953. It is safe to assume that all workers in these facilities received exposures directly from the material being used and from airborne and surface contamination in the facilities. Battelle (2011) and Christofano and Harris (1960) provide estimates of exposure for typical scenarios that were similar in many AWE uranium facilities. These estimates are based on several assumptions that were designed to determine reasonable exposures for workers in these categories: operators (production workers), general laborers, supervisors, and clerical. For this site profile, security and janitorial personnel have been placed in the clerical category. Ames Laboratory had a number of personnel with occupations listed as researcher, junior researcher, research technician, etc. It is assumed that these research workers were exposed to doses that were one-tenth those of the production workers [44].

Exposure levels in these documents were converted to annual doses. These doses are provided for the Chemistry Building, Physical Chemistry Annex 1, and Physical Chemistry Annex 2 in Tables 6-4, 6-5, and 6-6 below. External doses from submersion in airborne contamination were less than 0.001 rem/yr and, therefore, were insignificant. Doses from surface contamination on the floor were less than 0.001 rem/yr in Battelle (2011) and, therefore, these doses were not included in Annex 2 doses (Table 6-6). In addition, doses in Battelle (2011) were slightly less after 1950 than in the years before. Based on information available about Annex 2, there is no indication that production in that facility decreased after 1950. Therefore, the higher doses were applied for all years of operation.

The annual doses in the tables below can be prorated as necessary based on the work period of the employee. Doses from the tables should be applied with the parameters from Table 6-3 for deep and shallow doses.

Table 6-3. Parameters for application of unmonitored external doses from 1942 through 1953.

Exposure category	Overestimate energy distribution	Exposure type	GSD
Whole body	Photons 100% 30 to 250 keV	Lognormal	5
Skin	Electrons >15 keV	Lognormal	5
Exposure from contaminated floor	Photons 100% 30 to 250 keV	Lognormal	4.5

Tables 6-4, 6-5, and 6-6 provide the annual doses that can be applied to workers at Ames Laboratory before 1953.

Table 6-4. Annual external exposures from uranium for workers in Chemistry Building (mrem/yr).

Period	Job title	Whole body[a]	Skin of hands or forearms[a]	Skin other than hand or forearm[a]	Exposure from contaminated floor[a]
August 1942–December 1945	Production worker/operator	N/A[b]	N/A[b]	N/A[b]	N/A[b]
	Researcher	930	2,900	1,600	41
	Supervisor	2,300	7,200	4,100	100
	Clerical, security, janitorial	230	720	410	10
	General laborer	4,600	14,000	8,100	210
January 1946–December 1953[c]	Production worker/operator	N/A[b]	N/A[b]	N/A[b]	N/A[b]
	Researcher	93	290	160	4.1
	Supervisor	230	720	410	10
	Clerical, security, janitorial	23	72	41	1
	General laborer	460	1,400	810	21

a. This value can be prorated based on the fraction of year worked by the AWE employee.
b. In general, there were no operators in this building; consider that they might have worked in Physical Chemistry Annex 1 or 2 instead, and apply doses from the tables below.
c. After production period ended in 1945, research is assumed to have one-tenth the exposure potential.

Table 6-5. Annual external exposures from uranium for workers in Physical Chemistry Annex 1 (mrem/yr).

Period	Job title	Whole body[a]	Skin of hands or forearms[a]	Skin other than hand or forearm[a]	Dose from contaminated floor[a]
August 1942–August 1945	Production worker/operator	9,300	29,000	16,000	410
	Researcher	930	2,900	1,600	41
	Supervisor	2,300	7,200	4,100	100
	Clerical, security, janitorial	230	720	410	10
	General laborer	4,600	14,000	8,100	210

a. This value can be prorated based on the fraction of year worked by the AWE employee.

Table 6-6. Annual external exposures from uranium for workers in Physical Chemistry Annex 2 (mrem/yr).

Period	Job title	Whole body[a]	Skin of hands or forearms[a]	Skin other than hand or forearm[a]
January 1944–December 1953	Production worker/operator	2,500	280,000	25,000
	Researcher	250	28,000	2,500

	Supervisor	220	14,000	1,100
	Clerical, security, janitorial	22	0	0
	General laborer	450	138,000	12,000

a. This value can be prorated based on the fraction of year worked by the AWE employee.

6.3.1.2 Coworker Doses – 1952 to the Present

Extensive dosimetry records have been found for Ames Laboratory workers; however, many of the records for 1965 to 1981 do not identify the person receiving the radiation dose. If a worker was monitored but cannot be identified in the dosimetry records, that individual must be considered unmonitored and assigned a dose in each year for which no clearly identified records exist.

A coworker data study was used for this site profile to permit dose reconstructors to complete certain cases for which external monitoring data were unavailable or incomplete. Coworkers are workers at a site (potentially grouped by work location, job description, or other appropriate category) whose measured doses are considered representative of those received by one or more workers with no individual monitoring data (ORAUT 2011a).

The general approach to applying coworker data for cases with little or no individual external monitoring data is to assign the 50th- or 95th-percentile dose with the intent that the assigned doses represent, but do not underestimate, the doses that would have been assigned had the worker been monitored (50th percentile) or if the monitored worker was clearly identified in the dosimetry records (95th percentile) (ORAUT 2011a).

Some workers might have never been monitored during their employment at Ames Laboratory. Workers with job titles such as security patrolman, craftsman, janitor, secretary, or clerk who did not work routinely in radiological areas were probably not monitored. In general, the 50th-percentile dose can be used to bound doses for those workers when professional judgment indicates the worker was likely to have been exposed to intermittent low levels of external radiation (ORAUT 2011a).

Some workers with job titles such as scientist, chemist, metallurgist, engineer, technician, or machinist were probably monitored, but some or all of their dosimetry records might be missing because of the lack of clearly identified dosimetry records between 1965 and 1981. If any part of a worker's dosimetry record is missing (unidentified), coworker data should be applied in the years for which records are missing (ORAUT 2011a). The 95th-percentile dose should be assigned to those workers who could have been regularly exposed.

The coworker data study for Ames Laboratory included all available dosimetry records from 1952 through 1981 (Martin 2006c). All dose results were analyzed, including zeros and blank values, to determine the 50th- and 95th-percentile doses for each year for beta, gamma, and neutron exposures (McCartney 2006). The results of the analysis are summarized in Table 6-7. The missed dose recommended for monitored workers in Table 6-8 was added to the 50th- and 95th-percentile values in Table 6-7. Specifically, half of the maximum annual missed doses were added to the reported annual doses, except the reported positive doses, in which case the maximum missed dose was reduced by the dose corresponding to one badge exchange (because it is not possible that all individual badge results were zero if a positive annual dose was reported) (ORAUT 2011a). For beta doses, the maximum missed dose was not added to ensure missed dose is not assigned twice when evaluating dose to shallow dose organs (ORAUT 2005b). These external doses cannot be applied for the SEC-00038 class of employees (NIOSH 2006), which covers the period from August 13, 1942, through December 31, 1954. As stated in Section 1.3, this SEC class has been expanded to include all Ames employees, contractors, and subcontractors.

Table 6-7. Assigned dose for unmonitored workers (mrem/yr) (McCartney 2006).

Year	Beta		Gamma		Neutron		Raw data files References
	50th percentile	95th percentile	50th percentile	95th percentile	50th percentile	95th percentile	
1952	N/A[a]	N/A[a]	130	268[b]	163[c]	163[c]	Ames (1954a)
1953	0	0	520	590[b]	650[c]	650[c]	Ames (1954a)
1954	0	0	520	737[b]	650	650	Ames (1954b, 1955)
1955	0	0	520	520[b]	650	650	Ames (1956a)
1956	0	0	520	520[b]	650	650	Ames (1956b)
1957	0	0	520	520[b]	650	650	Ames (1957)
1958	0[a]	0[a]	520	520[b]	650	650	Ames (1959)
1959	0[a]	0[a]	520	520[b]	650	650	Ames (1960b)
1960	0	0	520	520[b]	650	650	Ames (1961b)
1961	0[a]	0[a]	520	520[b]	650	650	Ames (1962b)
1962	0	0	520	520[d]	650	650[d]	Ames (1963)
1963	0	0	240	265[e]	650	650	Ames (1964)
1964	0	0	240	313[e]	650	650	Ames (1965)
1965	0	0	240	408[e]	650	820[e]	Ames (1966)
1966	0	86[e]	240	565[e]	806[f]	988[e]	HPS (1966)
1967	0	246[e]	240	745[e]	932[f]	1,128[e]	HPS (1967)
1968	0	87[e]	240	769[e]	876[f]	1,086[e]	HPS (1968)
1969	0	175[e]	240	1,182[e]	1,030[f]	1,296[e]	HPS (1969)
1970	0	0	240	671[e]	680[f]	932[e]	HPS (1970)
1971	0	0	240	809[e]	722[f]	950[e]	HPS (1971)
1972	0	0	240	512[e]	652[f]	988[e]	HPS (1972)
1973	0	0	240	701[e]	650	932[e]	Ames (1973)
1974	0	0	240	721[e]	650	876[e]	HPS (1974)
1975	0	0	240	1,298[e]	650	870[e]	Ames/HPS (1975)
1976	0	0	240	497[e]	650	958[e]	Ames/HPS (1976)
1977	0	0	240	752[e]	650	904[e]	Ames/HPS (1977)
1978	0	0	240	722[e]	650	652[e]	Ames/HPS (1978)
1979	0	0	240	819[e]	650	650	Ames/HPS (1979)
1980	0	0	240	1,595[e]	N/A[c]	N/A[c]	Ames/HPS (1980)
1981	0	0	240	240	N/A[c]	N/A[c]	HPS (1981)
1982	0	0	85	555	N/A[c]	N/A[c]	Landauer (1982)
1983	325	510	105	134	N/A[c]	N/A[c]	Landauer (1983)
1984	0	0	60	85	N/A[c]	N/A[c]	Landauer (1984)
1985	0	0	60	125	N/A[c]	N/A[c]	Landauer (1985)
1986	0	0	60	60	N/A[c]	N/A[c]	Landauer (1986)
1987	0	0	60	114	N/A[c]	N/A[c]	Landauer (1987)
1988	0	310	72	149	N/A[c]	N/A[c]	Landauer (1988)
1989	0	0	60	81	N/A[c]	N/A[c]	Landauer (1989)
1990	0	9	60	91	N/A[c]	N/A[c]	Landauer (1990)
1991	0	51	60	85	N/A[c]	N/A[c]	Landauer (1991)
1992	0	30	60	71	N/A[c]	N/A[c]	Landauer (1992)
1993	0	22	60	118	N/A[c]	N/A[c]	Landauer (1993)
1994	0	20	62	138	N/A[c]	N/A[c]	Landauer (1994)
1995	0	70	60	68	N/A[c]	N/A[c]	Siemens (1995)
1996	0	0	60	65	N/A[c]	N/A[c]	ICN (1996)
1997	0	0	60	60	N/A[c]	N/A[c]	Hokel (1998)
1998	0	0	205	245	N/A[c]	N/A[c]	Landauer (1998)
1999	0	115	60	160	N/A[c]	N/A[c]	Landauer (1999)
2000	0	150	60	110	N/A[c]	N/A[c]	Landauer (2000)
2001	0[a]	0[a]	60	76	N/A[c]	N/A[c]	Simpson (2001); Beckel (2001a,b, 2002)

Year	Beta		Gamma		Neutron		Raw data files References
	50th percentile	95th percentile	50th percentile	95th percentile	50th percentile	95th percentile	
2002	0	49	60	68	N/A[c]	N/A[c]	Landauer (2002)
2003	0	150	201	394	N/A[c]	N/A[c]	Landauer (2003)
2004	0[a]	0[a]	63	118	N/A[c]	N/A[c]	Beckel (2004a,b, 2005)
2005	0	204	60	123	N/A[c]	N/A[c]	McGuigan (2006)

a. No beta doses were measured or reported in 1952, 1958, 1959, 1961, 1981, 1982, 2001, or 2004.
b. As discussed in Section 6.2.1, the practice of assigning doses equal to the MRD is roughly equivalent to the standard method for correction for missed dose for monitored workers (NIOSH 2007b). Therefore, no additional correction to gamma doses is required for monitored workers for the period from 1952 through 1961. However, from 1955 through 1961, this method resulted in adjusted doses that were less than the missed dose assigned to unmonitored workers (50th-percentile value), so the higher missed dose was assigned.
c. Neutron dosimeters were not provided in 1952 or 1953 and were not needed after 1979.
d. The dosimeter exchange frequency in 1962 is uncertain; therefore, the missed dose for a biweekly frequency was assumed to be favorable to claimants.
e. The 95th-percentile value was assumed to be received all in one month; missed dose was added for the other 11 months of the year.
f. The 50th-percentile value was assumed to be received all in one month; missed dose was added for the other 11 months of the year.

Table 6-8. Potential missed dose for monitored workers.

Dosimeter	Period	Exchange frequency[a]	MDL (mrem)			Missed annual mean dose (mrem)		
			Skin	Deep	Neutron	Skin	Deep	Neutron
Film badge-β,γ,n	1952–1956	Biweekly	40[b]	40[b]	50[b]	0[c]	0[c]	0[c]
	1957–1962	Biweekly	40[b]	40[b]	50[b]			
Film badge-β,γ,n	1963–1981	Monthly	40	40	50	240	240	300
TLD-β and γ	1981–present	Quarterly	30	30	N/A[d]	60	60	

a. Exchange frequencies were established from dosimetry reports.
b. Estimated MDL typical of film dosimeter capabilities (Wilson 1960, 1987; NIOSH 1993; NRC 1989; Wilson et al 1990).
c. As discussed in Section 6.2.1, doses equal to the MRD were added to the recorded dose when zero readings occurred. Therefore, no additional correction is required for monitored workers for the period from 1952 through 1961.
d. Neutron dosimeters were not assigned after 1979, due to the removal of fuel from the ALRR in October 1979.

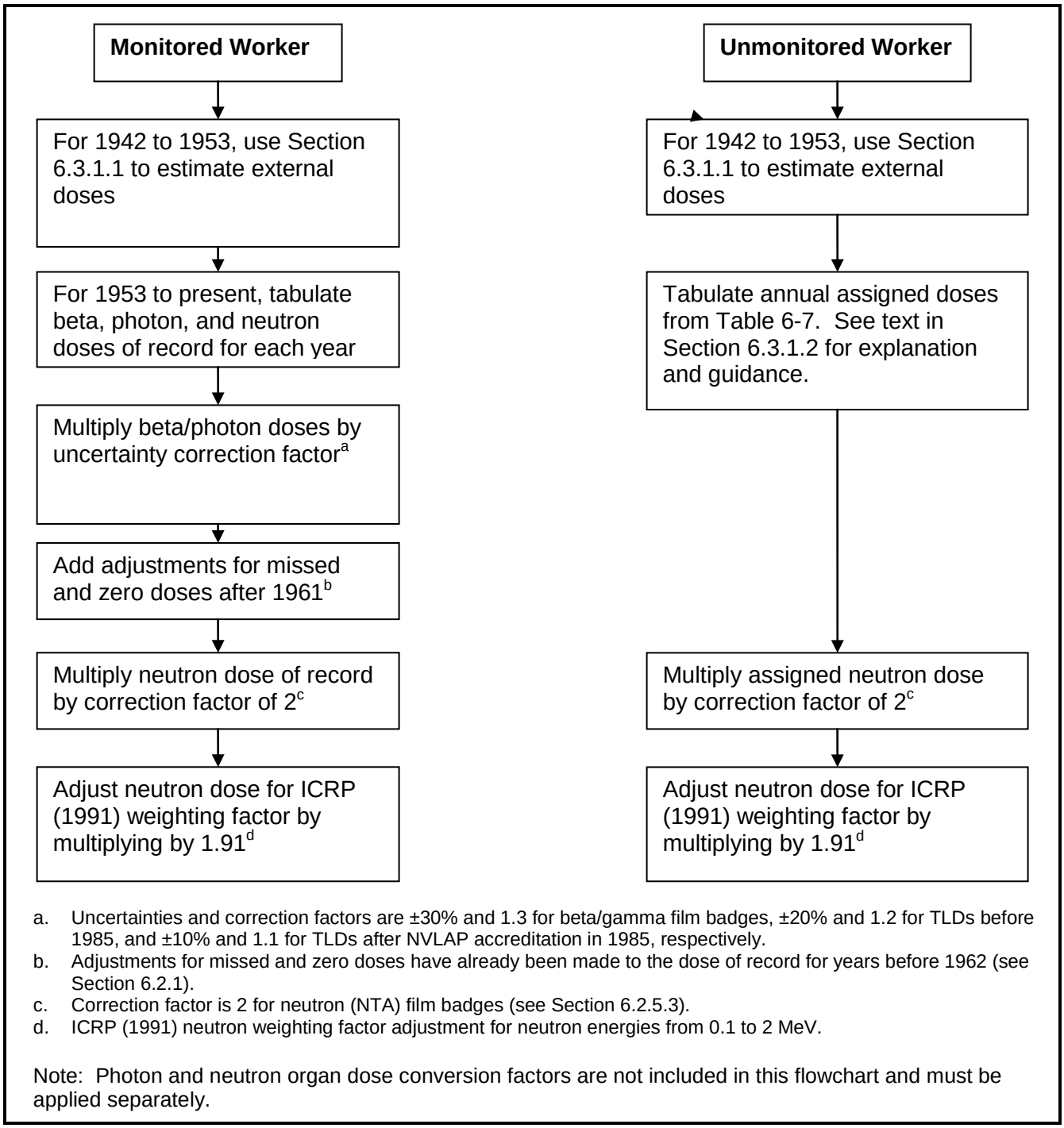
6.3.2 Missed External Dose for Monitored Workers

If external dose data are found in a worker's file, dose reconstructors should assign a missed photon dose based on the MDL/2 method and the number of exchange periods (NIOSH 2007b) listed in Table 6-8 for the dosimetry systems. These missed doses are included in the values in Table 6-7.

A flow chart is provided in Figure 6-2 to guide dose reconstructors in the application of the above tables and recommendations. The Monitored Worker side of the figure should be used if there are record files for the entire period of employment. If data for any exposure period are missing, the Unmonitored Worker side of the figure should be used for the missing period only. If there are no dose record files for a worker, the Unmonitored Worker side of the figure should be used for the entire period of employment.

6.3.3 Uncertainty in Photon Doses

For the usual analysis of measured film badge doses, MDLs in the literature range from about 30 to 50 mrem for beta/photon irradiation (Wilson et al. 1990). It is possible to read a photon dose of 100 mrem to within ±15 mrem if the exposure involved photons with energies between several keV



a. Uncertainties and correction factors are ±30% and 1.3 for beta/gamma film badges, ±20% and 1.2 for TLDs before 1985, and ±10% and 1.1 for TLDs after NVLAP accreditation in 1985, respectively.
b. Adjustments for missed and zero doses have already been made to the dose of record for years before 1962 (see Section 6.2.1).
c. Correction factor is 2 for neutron (NTA) film badges (see Section 6.2.5.3).
d. ICRP (1991) neutron weighting factor adjustment for neutron energies from 0.1 to 2 MeV.

Note: Photon and neutron organ dose conversion factors are not included in this flowchart and must be applied separately.

Figure 6-2. Flowchart for monitored and unmonitored workers. Source: authors.

and several MeV (Morgan 1961). The estimated standard error in recorded film badge doses from photons of any energy is ±30% (ORAUT 2006d). The estimated uncertainty in doses recorded by TLDs is ±20% from 1982 through 1984 and ±10% since 1985 with NVLAP accreditation (Landauer 1985).

6.3.4 Skin Dose

In years before 1981, the beta and skin dose records included beta doses only; that is, only nonpenetrating beta dose is recorded. In 1981 and subsequent years, the beta and skin dose has

been calculated as the sum of the whole-body penetrating dose and the nonpenetrating dose (Landauer 1985). For cases in which no nonpenetrating dose was recorded, the skin dose is assumed to be equal to the whole-body penetrating dose. Guidance on determining skin dose can be obtained from *Interpretation of Dosimetry Data for Assignment of Shallow Dose* (ORAUT 2005b).

6.3.5 Extremity Dose

Assignment to and use of extremity dosimeters by Ames Laboratory workers has been inconsistent. In some years, there are no reported extremity dose results; in many years, there are only a few results, which suggests a less-than-rigorous extremity dosimetry program given the types and quantities of radioactive materials handled. Results are so sparse that it must be concluded that extremity doses were essentially unmonitored.

A standard practice in operational health physics is to use a factor of 10 between whole-body and extremity exposures (PNNL 2004). That is, if the measured contact dose rate from a source is 10 times (or more) the measured dose rate at the location of the whole-body dosimeter, extremity dosimeters should be assigned for the work. In the case of missing extremity dose data (or unmonitored extremities), the whole-body dose can be multiplied by 10 and the result assigned as a conservative extremity dose. If the cancer site involves the hands, forearms, feet, or legs below the knees, the conservative calculated extremity dose can be used by dose reconstructors.

When a more accurate evaluation of extremity dose based on whole-body dosimetry readings is needed, the dose reconstructor should refer to the Division of Compensation Analysis and Support, DCAS-TIB-0010, *Technical Information Bulletin: Best Estimate External Dose Reconstruction for Glovebox Workers,* (NIOSH 2010c) for information regarding adjustments to whole-body doses at locations associated with the extremities.

6.3.6 Radiation Dose Fraction

Section 6.2.4 addresses the recommended energy ranges and fractions for Ames Laboratory dose according to facilities, processes, or activities as required by the Interactive RadioEpidemiological Program (IREP).

6.4 ORGAN DOSE

Once the *Hp(10)* adjusted doses have been calculated for each year, the values are used to calculate organ doses of interest using the external dose reconstruction implementation guidelines (NIOSH 2007b). Consistent with Project direction, dose reconstructors should assume anterior-posterior (AP; front to back) geometry for the irradiation geometry and for conversion to organ dose. For photons measured with film badges (before 1982), use the conversion factor from exposure to organ dose. Beginning in 1982, when all measurements were made using TLDs, deep dose equivalent should be used. For neutron doses, use the conversion factors from deep dose equivalent to organ dose for AP irradiation from NIOSH (2007b).

7.0 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 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.

[1] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
This statement is based on personal experience with a 52-in. cyclotron at the University of Colorado, where a bioassay program, routine contamination surveys, and effluent monitoring were conducted. Bioassay results were always indistinguishable from background. Particulate contamination frequently had a very short half-life and detectable levels were almost always less than 10 CFR Part 835, Appendix D values. Airborne effluents also had short half-lives and were indistinguishable from background. This information is consistent with the Site Profile for SLAC (ORAUT 2007a), which found that the environmental dose from accelerator effluents was less than 1 mrem/yr and negligible.

[2] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
Common practice in accelerator health physics is to conduct a periodic fenceline survey of maximum radiation dose rates. The purpose of such surveys is to determine the maximum offsite impact of accelerator operations and identify special controls or limitations that might be needed for safe operation. To produce maximum fenceline dose rates, the maximum beam energy and beam current were applied to a target selected for maximum radiation production. These operating conditions were unusual and could be detrimental to the accelerator equipment, so the maximum dose rate conditions existed only long enough to obtain the fenceline survey results. Normally, the accelerator operated at reduced beam currents and with targets that produced much lower radiation dose rates on and off site. This was verified by examining the dosimetry records of the synchrotron staff, which showed no unusually high exposures during the May 1961 survey or at any other time.

Ames (1961a) states that "background reading in this area (synchrotron) prior to beam being turned on was 0.5 to 1.0 mr/hr." Apparently, the background reading was taken near the synchrotron where the radiation level was slightly elevated (due to activation products) above the normal natural background level. However, the exact location of the background reading cannot be determined. The fenceline readings were taken some distance from the background reading location and should have not been influenced by the elevated background near the accelerator. In any event, the elevated background dose rates were not subtracted from the fenceline dose rates reported in Table 4-1.

[3] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
Experience at other accelerator facilities indicates that generally about one-third of the total staff time is spent on nonoperational activities, such as experiment setup, equipment startup, maintenance, repairs, etc. Thus, about two-thirds of the total time is available for operations.

[4] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The radiation conditions during the fenceline radiation survey on May 16, 1961, produced dose rates at occupiable locations inside the Synchrotron Building that were greater than 1 mR/hr. Full-time occupancy in such a location would give an annual dose of greater than 2,000 mrem. A review of dosimetry records for synchrotron personnel indicated that no one received such a dose during 1961 or at any other time. The average annual dose was less than 200 mrem. The average annual dose outside the Synchrotron Building would have been much lower.

[5] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The buildings of Ames Laboratory are on the main campus of Iowa State University and are not physically separated in any way from other university buildings (i.e., no fence or other boundaries). Staff, students, and the public are free to walk around the exterior walls of the buildings.

[6] Shipler, Dillard B. ORAU Team. Principal Health Physicist. February 2007.
More than 1,000 tons of uranium metal were processed at Ames Laboratory in about 3 years. The University's chemistry laboratories were used for R&D on small quantities, rather than production quantities. The estimate that the source term in the laboratories was 1/100th of the amount handled in production facilities was made and was shown to be comparable to releases from laboratories at the University of California at Berkeley, thereby supporting the estimate as reasonable.

[7] Shipler, Dillard B. ORAU Team. Principal Health Physicist. February 2007.
The source of the intake numbers was explained in the previous attribution. The assumption was made that the thorium was in equilibrium with its progeny, which is favorable to claimants. The intake-to-organ-dose conversion factors require that ^{232}Th, ^{228}Ra, and ^{228}Th be entered as separate intakes.

[8] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
Because the chemical forms and, therefore, the absorption types of the uranium and thorium were not known, especially for the R&D activities, all the default values in ICRP Publication 68 (ICRP 1995) were considered possible.

[9] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
The first sustained chain reaction occurred in the Chicago Pile on December 2, 1942. The pile was dismantled in the spring of 1943, so the first irradiated fuel would not have reached Ames Laboratory until after that, probably not until 1944, but assuming 1943 was considered favorable to claimants.

[10] Bihl, Donald E. ORAU Team. Principal Health Physicist, February 2007.
The hot cell was engineered to prevent releases into the workplace and would have had better filtration in the exhaust system than a general chemistry lab built in the 1940s. The exact nature of the hot laboratory used in the 1940s was not discovered in the records. Presumably, it would have involved some methods for recognizing and controlling the spread of contamination with some filtration on the exhausts, but it was unlikely that the controls of the hot laboratory were as good as those for the hot cell.

[11] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The decommissioning date of 1990 was determined in a personal communications with (Name redacted), Radiation Safety Officer at Ames Laboratory, July 2006.

[12] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Review of references and discussions in Sections 2.1 and 2.2 indicated the location of operations with uranium.

[13] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Review of references and discussions in Sections 2.1 and 2.2 indicated the location of operations with uranium.

[14] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Because there was no information on the amount of time in a day that a worker would be exposed to material, an assumption had to be made. The assumptions in the section on uranium are the same as the assumptions used in Battelle (2011).

[15] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Because there was no information on the amount of time in a day that a worker would be exposed to material, an assumption had to be made. The assumptions in the section on uranium are the same as the assumptions used in Battelle (2011).

[16] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Because there was no information on the amount of time in a day that a worker would be exposed to material, an assumption had to be made. The assumptions in the section on uranium are the same as the assumptions used in Battelle (2011).

[17] Bihl, Donald E. ORAU Team. Principal Health Physicist. April 2007.
The calculations in the preceding three paragraphs were conducted by Mr. Bihl using IMBA from data in the referenced documents. Although hard to read, the urinalysis values used for the most exposed worker from Tybout (1944) were 0.00, 0.06, 0.00, 0.20, 0.00, and 0.10 mg/L adjusted to mg/d. The 0.00 values were entered as <0.056 mg/d, assuming the detection level was approximately 0.04 mg/L.

[18] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
As long as the Energy Employee's actual excretion is less than the excretion plotted in the figures, the intake is also less than the default intakes. For a noncompensable case, if the true intake is less than the default intake, the default intake can be used for efficiency.

[19] Bihl, Donald E., and Hickey, Eva Eckert. ORAU Team. Principal Health Physicists. February 2007.
The air sample results were indistinguishable from background, so were not much help. Of the surface survey results in areas somewhat accessible to people, the highest result was 1,224 dpm; however, there was no record of the surface area smeared. Surface activity is usually reported in units of activity per 100 cm^2, but the authors could not be sure of that. We had to "assume" the survey values were for 100 cm^2. Because of that uncertainty, we arbitrarily raised the 1,224-dpm value to 2,000 dpm/100 cm^2 to represent an upper bound clearly.

[20] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
Indoors resuspension factors range from 10^{-6} to 10^{-3}. The International Atomic Energy Agency (Clayton 1970) recommends an indoor resuspension factor of 5×10^{-5}. The factor 10^{-4} was chosen for this application to represent an upper bound for chronic intake from surfaces not immediately accessible and, therefore, not often disturbed.

[21] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
Hokel et al. (1998) indicated the thorium was assumed to be in equilibrium with its progeny. The intake dose conversion factors account for buildup/decay of ^{232}Th progeny in the body after the intake, but the activity of the progeny already in the air at intake has to be treated as separate intake radionuclides. Because of their short half-lives, ^{228}Ac is included in dose calculations for ^{228}Ra and all the progeny from ^{228}Th are included in the dose calculations for ^{228}Th, so only ^{228}Ra and ^{228}Th need to be entered as intakes separate from ^{232}Th.

[22] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Review of Hokel et al. (1998) suggested there were areas with higher contamination, such as crawlspaces and vertical void spaces in rooms that were rarely, if ever, occupied. The statement in Section 5.2 that contamination was 10 to 100 times greater generalizes the discussion in Hokel et al. on this subject.

[23] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
In Hokel et al. (1998), there were statements that the areas that were rarely accessed that are still contaminated are carefully controlled, and that workers wear protective equipment and are monitored by health physics technicians; therefore, the 6-pCi/d chronic intake is an upper bound.

[24] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
MDAs or recording levels varied at the major AEC sites in the 1960s and 1970s from 1 µCi/L at the Hanford Site, Savannah River Site (SRS), and LANL, to 0.02 µCi/L at ORNL and Lawrence Livermore National Laboratory, to 0.0002 µCi/L at INL. The Hanford and SRS values were recording levels with actual laboratory MDAs generally being lower. Review of the way data were recorded at Ames Laboratory indicated that 0.1 µCi/L was the likely MDA or recording level.

[25] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
The table was created by Mr. Bihl using the lognormal tritium dose plots created by Thomas R. La Bone, as explained in the text.

[26] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
Strontium is type F with the exception of the titanate form, which is rare. There is no reason to suspect strontium titanate was in use at Ames Laboratory.

[27] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Table 5-8 is a summary of the intakes described in this section. The data are taken from (1) Tables 5-1, 5-2, and 5-3 and the discussions in Sections 5.1.1.1 on uranium inhalation and Section 5.1.1.2 on uranium ingestion; (2) Table 5-4 and the discussion in Section 5.1.1.3 on resuspension of uranium during periods of no operation; (3) the discussion in Section 5.2 on thorium exposure from leftover thorium contamination; (4) doses from tritium as discussed in Section 5.3; and (5) Table 5-7 doses from fission product intakes as discussed in Section 5.4.

[28] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The uranium and thorium metal production processes at Ames Laboratory established that workers were exposed to external radiation doses between 1942 and 1952. The only records of radiation dosimetry measurements during this period are Tybout (1944) and Voss (1949), which are very incomplete. Either the radiation doses were not measured or essentially no records were kept. In either case, the workers during this period were unmonitored.

[29] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
Voss (1949) recorded radiation exposures measured by pocket chambers for 15 individuals during a 2-week period in February 1949. These were the only pocket chamber dosimeter records found for the 1942-to-1952 period. The record suggests that pocket chambers were available during this period, but either they were not used or, if they were used, records of the dosimeter readings were not kept.

[30] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
During the 1965-to-1981 period, the film badge dosimetry records reflect badge numbers only. Rosters of names and badge numbers for bioassay records were found and it was possible to

correlate some badge numbers and individual names. However, there were many badge numbers for which a name could not be assigned.

[31] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
As the spreadsheets were assembled, some addition and subtraction errors in the dosimetry records were noted. In all cases, the result that gave the highest dose was assumed and recorded in the spreadsheets.

[32] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
After 1979, neutrons were no longer included in the source term at Ames Laboratory and neutron dosimeters were no longer needed.

[33] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
As stated in Section 6.2.1, recordkeeping from January 1962 through June 1963 was less than adequate. The only records found for 1962 were some calibration data from April 1962 and the annual summary data for 1962 (Ames 1963). It was assumed that the in-house film badge system continued through December 1962, with film badges exchanged on a biweekly frequency; however, no records could be found to confirm this assumption. Monthly film badge service was provided for all Ames Laboratory personnel with the potential for radiation exposure by the Atomic Film Badge Corporation from July 1963 through March 1965. No records were found for the first half of 1963.

[34] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
With the wide variety of radiation sources used at Ames Laboratory, it is possible that beta, photon, and neutron fields with energies outside the assumed ranges existed. However, the predominant radiation fields were within the assumed energy ranges and, if it is assumed that 100% of the radiation fields were within these ranges, it is a conservative assumption that is generally favorable to claimants.

[35] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The early history of research at Ames Laboratory was described by Fulmer (1947); later history was described in Ames (1985). These documents discussed numerous research projects that involved many beta-emitting radioisotopes. The quantities used in research were much smaller than the quantities involved in production operations.

[36] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The descriptions of research and production operations at Ames Laboratory (Fulmer 1947; Ames 1985) established the presence of photons in the less-than-30-keV, 30-to-250-keV, and greater-than-250-keV energy ranges. However, the predominant photon energy range was from 30 to 250 keV. If it is assumed that 100% of the photon radiation fields were within this range, it is a conservative assumption that is generally favorable to claimants.

[37] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
Descriptions of the synchrotron (Ames 1967) and the research reactor (Voigt 1981) established that neutrons of varying energies were present in these facilities. However, no records were found that characterized the neutron spectrum in either facility. If it is assumed that 100% of the neutron radiation fields were with the 0.1-to-2-MeV energy range, it is a conservative assumption that is generally favorable to claimants.

[38] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
Commercial film badge service was supplied by BNL from April 1954 to June 1957, by Nucleonic Corporation of America from July 1957 to June 1963, by Atomic Film Badge

Corporation from July 1963 to March 1965, and by Health Physics Services from June 1965 to September 1981 (see Table 6-1).

[39] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The film badge dosimeters used at Ames Laboratory were state-of-the-art commercial products that were properly designed and calibrated to give reliable dose results. The TLDs used since 1980 were commercial products that were NVLAP-accredited and calibrated to give reliable dose results.

[40] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The neutron doses received by ALRR personnel were somewhat higher when compared with similar personnel at the High Flux Beam Reactor at BNL (ORAUT 2010a). The ALRR records did not include any gamma or neutron radiation surveys around the experimental areas, nor was there any evidence of neutron spectral measurements. The lack of survey data suggests that management of neutron shielding might have been less than adequate.

[41] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The maximum beam energy at the Ames Laboratory synchrotron was 80 MeV; the maximum beam energy at the SLAC synchrotron was 200 MeV. Both accelerators produced neutrons, but the spectrum at SLAC included higher energy neutrons, thus the neutron radiation fields were not identical.

[42] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The SLAC Site Profile (ORAUT 2007b) used a neutron dose correction factor of 1.53 ±0.14 to account for unmeasured neutrons with energies below 800 keV. Although the neutron spectra of the Ames Laboratory and SLAC synchrotrons were not identical, they were similar enough to justify the assumption of a neutron dose correction factor of 2 for the Ames synchrotron.

[43] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
During the 1965-to-1981 period, the film badge dosimetry records reflect badge numbers only. Rosters of names and badge numbers for bioassay records were found and it was possible to correlate some badge numbers and individual names. However, there were many badge numbers for which a name could not be assigned. These records indicate that some workers were monitored, but they have to be considered unmonitored because the records for specific individuals cannot be retrieved.

[44] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Because there was no information on the amount of time in a day that a worker would be exposed to material, an assumption had to be made. The assumptions in the section on uranium are the same as the assumptions used in Battelle (2011).

REFERENCES

AEC (U.S. Atomic Energy Commission), 1955, "Intercomparison of Film Badge Interpretations," *Isotopics*, volume 2, number 5, pp. 8–23. [SRDB Ref ID: 3640]

Ames (Ames Laboratory), 1949, Daily Meter Record, Pocket Chamber Readings, Iowa State College, Ames, Iowa. [SRDB Ref ID: 25819]

Ames (Ames Laboratory), 1951, *Ames Laboratory*, United States Atomic Energy Commission, Ames, Iowa, August. [SRDB Ref ID: 18861]

Ames (Ames Laboratory), 1954a, Film Badge Records for 1953, Iowa State College, Ames, Iowa. [SRDB Ref IDs: 18821, 18822, 18833]

Ames (Ames Laboratory), 1954b, Film Dosimetry on Hands, Iowa State College, Ames, Iowa. [SRDB Ref IDs: 25816]

Ames (Ames Laboratory), 1955, Film Badge Records for 1954, Iowa State College, Ames, Iowa. [SRDB Ref IDs: 18834, 18835]

Ames (Ames Laboratory), 1956a, Film Badge Records for 1955, Iowa State College, Ames, Iowa. [SRDB Ref IDs: 18835, 18841]

Ames (Ames Laboratory), 1956b, Film Badge Records for 1956, Iowa State College, Ames, Iowa. [SRDB Ref IDs: 25169, 25171]

Ames (Ames Laboratory), 1958, Film Badge Records for 1957, Iowa State College, Ames, Iowa. [SRDB Ref IDs: 25214, 25215, 25216]

Ames (Ames Laboratory), 1959, Film Badge Records for 1958, Iowa State University, Ames, Iowa. [SRDB Ref IDs: 25459, 25460]

Ames (Ames Laboratory), 1960a, *Historical Development of the Ames Laboratory*, Historical Supplement, U.S. Atomic Energy Commission, Iowa State University, Ames, Iowa. [SRDB Ref ID: 18832, pp. 11-18]

Ames (Ames Laboratory), 1960b, Film Badge Records for 1959, Iowa State University, Ames, Iowa. [SRDB Ref IDs: 25461, 25898]

Ames (Ames Laboratory), 1961a, "Synchrotron Building Radiation Survey," U.S. Atomic Energy Commission, Iowa State University, Ames, Iowa, May 16. [SRDB Ref ID: 25766]

Ames (Ames Laboratory), 1961b, Film Badge Records for 1960, Iowa State University, Ames, Iowa. [SRDB Ref IDs: 25907, 25898]

Ames (Ames Laboratory), 1962a, *Iowa State University, Institute for Atomic Research, including the Ames Laboratory of the Atomic Energy Commission (AEC)*, Ames, Iowa. [SRDB Ref ID: 18856]

Ames (Ames Laboratory), 1962b, Film Badge Records for 1961, Iowa State University, Ames, Iowa. [SRDB Ref ID: 25913]

Ames (Ames Laboratory), 1963, Film Badge Records for 1962, Iowa State University, Ames, Iowa. [SRDB Ref ID: 25909, 25837]

Ames (Ames Laboratory), 1964, Film Badge Records for 1963, Iowa State University, Ames, Iowa. [SRDB Ref IDs: 25914, 25837]

Ames (Ames Laboratory), 1965, Film Badge Records for 1964, Iowa State University, Ames, Iowa. [SRDB Ref IDs: 25916, 25837]

Ames (Ames Laboratory), 1966, Film Badge Records for 1965, Iowa State University, Ames, Iowa. [SRDB Ref ID: 25919]

Ames (Ames Laboratory), 1967, *25th Year Open House: 1942-1967, April 7-8, 1967*, Iowa State University, Ames, Iowa. [SRDB Ref ID: 18830]

Ames (Ames Laboratory), 1973, Film Badge Records for 1973, Iowa State University, Ames, Iowa. [SRDB Ref ID: 25839]

Ames (Ames Laboratory), 1974, "Personnel Monitoring Program," Iowa State University, Ames, Iowa. [SRDB Ref ID: 25767]

Ames (Ames Laboratory), 1985, *The Building of the Ames Laboratory*, Iowa State University, Ames, Iowa. [SRDB Ref ID: 18860]

Ames (Ames Laboratory), 1996, *Site Environmental Report, Calendar Year 1995*, IS-5123, Iowa State University, Ames, Iowa. [SRDB Ref ID: 25225]

Ames/HPS (Ames Laboratory/Health Physics Services, Inc.), 1975, Film Badge Records for 1975, Iowa State University, Ames, Iowa. [SRDB Ref ID: 25860]

Ames/HPS (Ames Laboratory/Health Physics Services, Inc.), 1976, Film Badge Records for 1976, Iowa State University, Ames, Iowa. [SRDB Ref ID: 26613]

Ames/HPS (Ames Laboratory/Health Physics Services, Inc.), 1977, Film Badge Records for 1977, Iowa State University, Ames, Iowa. [SRDB Ref ID: 25861]

Ames/HPS (Ames Laboratory/Health Physics Services, Inc.), 1978, Film Badge Records for 1978, Iowa State University, Ames, Iowa. [SRDB Ref ID: 26615]

Ames/HPS (Ames Laboratory/Health Physics Services, Inc.), 1979, Film Badge Records for 1979, Iowa State University, Ames, Iowa. [SRDB Ref ID: 26619]

Ames/HPS (Ames Laboratory/Health Physics Services, Inc.), 1980, Film Badge Records for 1980, Iowa State University, Ames, Iowa. [SRDB Ref ID: 26619]

Anderson, J. L., and N. E. Hertel, 2005, "Bremsstrahlung Doses from Natural Uranium Ingots," *Radiation Protection Dosimetry*, volume 115, number 1-4, pp. 298-301. [SRDB Ref ID: 29254]

Battelle, 2007, *Default Assumptions and Methods for Atomic Weapons Employer Dose Reconstructions*, Battelle-TIB-5000, Rev 00, Battelle, Pacific Northwest Division, Richland, Washington, April 2.

Battelle, 2011, *Site Profiles for Atomic Weapons Employers that Worked Uranium Metals*, Battelle-TBD-6000, Revision 1, June 2011, Battelle, Pacific Northwest Division, Richland, Washington, June 17.

Beckel, J., 2001a, letter to M. Saar (U.S. Department of Energy, Chicago Operations Office), Iowa State University, Ames Laboratory, Ames, Iowa, July 26. [SRDB Ref ID: 27038, p. 3]

Beckel, J., 2001b, letter to M. Saar (U.S. Department of Energy, Chicago Operations Office), Iowa State University, Ames Laboratory, Ames, Iowa, November 19. [SRDB Ref ID: 27038, p. 4]

Beckel, J., 2002, letter to M. Saar (U.S. Department of Energy, Chicago Operations Office), Iowa State University, Ames Laboratory, Ames, Iowa, February 11. [SRDB Ref ID: 27038, p. 5]

Beckel, J., 2004a, letter to R. Purucker (U.S. Department of Energy, Ames Area Office), Iowa State University, Ames Laboratory, Ames, Iowa, September 1. [SRDB Ref ID: 26810, pp. 2-3]

Beckel, J., 2004b, letter to R. Purucker (U.S. Department of Energy, Ames Area Office), Iowa State University, Ames Laboratory, Ames, Iowa, December 2. [SRDB Ref ID: 26810, pp. 4-5]

Beckel, J., 2005, letter to R. Purucker (U.S. Department of Energy, Ames Area Office), Iowa State University, Ames Laboratory, Ames, Iowa, February 14. [SRDB Ref ID: 26810, p. 6]

Bihl, D. E., 2006, "Estimated Releases from the Hot Laboratory in Ames Chemistry Building," memorandum to file, Pacific Northwest National Laboratory, Richland, Washington, December 29. [SRDB Ref ID: 29227]

BRH (Bureau of Radiological Health), 1970, *Radiological Health Handbook*, PHS-PUBL-2016, U.S. Department of Health, Education, and Welfare, Public Health Service, Food and Drug Administration, Rockville, Maryland, Revised Edition, January. [SRDB Ref ID: 75017]

Christofano, E., and W. B. Harris, 1960, "The Industrial Hygiene of Uranium Refining," *Archives of Environmental Health,* volume 1, pp. 438-460 (reprinted). [SRDB Ref ID: 15774]

Clayton, R. F., 1970, *Monitoring of Radioactive Contamination on Surfaces*, Technical Report Series No. 120, International Atomic Energy Agency, Vienna, Austria. [SRDB Ref ID: 26076]

DOE (Department of Energy), 2001, *DOE Standard Guide of Good Practices for Occupational Radiological Protection in Uranium Facilities*, Change Notice No. 3, DOE-STD-1136-2000, Washington, D.C., December. [SRDB Ref ID: 4617]

ERDA (Energy Research and Development Administration), 1977, *Standards for Protection Against Radiation*, Manual Chapter 0524, Washington, D.C., March 30. [SRDB Ref ID: 11595]

Friedell, H. L., 1942, *Supplemental Report to Complete Preliminary Survey of Plants Engaged in Production of Materials*, U.S. Army Corps of Engineers, December 23. [SRDB Ref ID: 8967]

Fulmer, E. I., 1947, *History of the Ames Project Under the Manhattan District to December 31, 1946,* ISC-10, Iowa State College, Ames, Iowa. [SRDB No. 18823, pp. 6-7]

Hankins, D. E., 1961, *A Method of Determining the Intermediate Energy Neutron Dose*, IDO-16655, Philips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho. [SRDB Ref ID: 8084]

Hayes, D. F., 1956, *A Summary of Accidents and Incidents Involving Radiation in Atomic Energy Activities, June 1945 thru December 1955,* U.S. Atomic Energy Commission, Division of Organization and Personnel, Washington, D.C., August. [SRDB Ref ID: 30143]

Hickey, E. E., G. A. Stoetzel, D. J Strom. G. R. Cicotte, C. M. Wiblin, and S. A. McGuire, 1993, *Air Sampling in the Workplace*, NUREG-1400, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Washington, D.C., September. [SRDB Ref ID: 20129]

Hokel, B. W., 1998, "CY 1997 Radiation Exposure Reports, Ames Laboratory, Contract No. W-7405-ENG-82," letter to Y. McCormick (SAIC), Iowa State University, Ames Laboratory, Ames, Iowa, May 6. [SRDB Ref ID: 26683]

Hokel, B. W., J. H. Withers, W. P. Waters, and K. J. Ewing, 1998, *An Assessment of the Causes, Mitigation Efforts, and Current Status of ^{232}Th, ^{238}U, and Beryllium Contamination in Wilhelm Hall at Ames Laboratory*, Iowa State University, Ames Laboratory, Ames, Iowa, April. [SRDB Ref ID: 18848]

HPS (Health Physics Services, Inc.), 1966, Radiation Exposure Reports for Ames Laboratory, Baltimore, Maryland. [SRDB Ref ID: 25922]

HPS (Health Physics Services, Inc.), 1967, Radiation Exposure Reports for Ames Laboratory, Baltimore, Maryland. [SRDB Ref ID: 25923]

HPS (Health Physics Services, Inc.), 1968, Radiation Exposure Reports for Ames Laboratory, Baltimore, Maryland. [SRDB Ref ID: 25808]

HPS (Health Physics Services, Inc.), 1969, Radiation Exposure Reports for Ames Laboratory, Baltimore, Maryland. [SRDB Ref ID: 25809]

HPS (Health Physics Services, Inc.), 1970, Radiation Exposure Reports for Ames Laboratory, Baltimore, Maryland. [SRDB Ref ID: 25811]

HPS (Health Physics Services, Inc.), 1971, Radiation Exposure Reports for Ames Laboratory, Baltimore, Maryland. [SRDB Ref ID: 25810]

HPS (Health Physics Services, Inc.), 1972, Radiation Exposure Reports for Ames Laboratory, Baltimore, Maryland. [SRDB Ref ID: 25834]

HPS (Health Physics Services, Inc.), 1974, Radiation Exposure Reports for Ames Laboratory, Baltimore, Maryland. [SRDB Ref ID: 25859]

HPS (Health Physics Services, Inc.), 1981, Radiation Exposure Reports for Ames Laboratory, Baltimore, Maryland. [SRDB Ref ID: 26626]

IAEA (International Atomic Energy Agency), 1990, *Compendium of Neutron Spectra and Detector Responses for Radiation Protection Purposes*, Technical Reports Series No. 318, Vienna, Austria, July 26.

ICN (ICN Dosimetry Service), 1997, Radiation Exposure Report, Fountain Valley, California. [SRDB Ref ID: 26681]

ICRP (International Commission on Radiological Protection), 1973, *Data for Protection Against Ionizing Radiation from External Sources: Supplement to ICRP Publication 15*, Publication 21, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1991, *1990 Recommendations of the International Commission on Radiological Protection*, Publication 60, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1995, *Dose Coefficients for Intakes of Radionuclides by Workers*, Publication 68, Pergamon Press, Oxford, England.

ISU (Iowa State University), 2006, Campus Map, found at http://www.fpm.iastate.edu/maps/

Karsjen, S. R., 2003, *The Ames Project: History of the Ames Laboratory's Contributions to the Historic Manhattan Project, 1942-1946*, IS-T 2564, Iowa State University, Ames Laboratory Public Affairs, Ames, Iowa, pp. 10-12. [SRDB No. 18859]

Klevin, P. B., 1952, *Ames Research Laboratory Occupational Exposure to Thorium and Beryllium,* Ames-1, U.S. Atomic Energy Commission, New York Operations Office, New York, New York, July 14. [SRDB Ref ID: 18942]

Landauer (R. S. Landauer, Jr. & Co.), 1982, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26628]

Landauer (R. S. Landauer, Jr. & Co.), 1983, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26631]

Landauer (R. S. Landauer, Jr. & Co.), 1984, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26632]

Landauer (R. S. Landauer, Jr. & Co.), 1985, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26642]

Landauer (R. S. Landauer, Jr. & Co.), 1986, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26645]

Landauer (R. S. Landauer, Jr. & Co.), 1987, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26647]

Landauer (R. S. Landauer, Jr. & Co.), 1988, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26653]

Landauer (R. S. Landauer, Jr. & Co.), 1989, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26657]

Landauer (R. S. Landauer, Jr. & Co.), 1990, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26659]

Landauer (R. S. Landauer, Jr. & Co.), 1991, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26664]

Landauer (R. S. Landauer, Jr. & Co.), 1992, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26666]

Landauer (R. S. Landauer, Jr. & Co.), 1993, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26668]

Landauer (R. S. Landauer, Jr. & Co.), 1994, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26672]

Landauer (R. S. Landauer, Jr. & Co.), 1998, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26685]

Landauer (R. S. Landauer, Jr. & Co.), 1999, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26750]

Landauer (R. S. Landauer, Jr. & Co.), 2000, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26799]

Landauer (R. S. Landauer, Jr. & Co.), 2002, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26808]

Landauer (R. S. Landauer, Jr. & Co.), 2003, Radiation Dosimetry Report, Glenwood, Illinois. [SRDB Ref ID: 26809]

Martin, J. B., 2006a, "ALRR Workers Badged?" e-mail to D. B. Shipler (Pacific Northwest National Laboratory), Dade Moeller & Associates, Richland, Washington, November 1. [SRDB Ref ID: 29257, p. 2]

Martin, J. B., 2006b, "Unbadged Workers at the Synchrotron," e-mail to D. B. Shipler (Pacific Northwest National Laboratory), Dade Moeller & Associates, Richland, Washington, November 1. [SRDB Ref ID: 29256]

Martin, J. B., 2006c, Ames Laboratory Dose Records, 1952-2005, Dade Moeller & Associates, Richland, Washington. [SRDB Ref ID: 29229]

Mattmueller, F. C., 1965, "Contract No. W-7405-ENG-82," letter to F. H. Spedding (Ames Laboratory), U.S. Atomic Energy Commission, Contracts Division, Argonne, Illinois, August 25. [SRDB Ref ID: 25818]

McCartney, K. A., 2006, Statistical Analysis of Ames Laboratory External Dose Records (1952-2005), Dade Moeller & Associates, Cincinnati, Ohio. [SRDB Ref ID: 30977]

McGuigan, M., 2006, letter to R. Purucker (U.S. Department of Energy, Ames Area Office), Iowa State University, Ames Laboratory, Ames, Iowa, February 20. [SRDB Ref ID: 26811]

Morgan, K. Z., 1961, "Dosimetry Requirements for Protection from Ionizing Radiation", *Selected Topics in Radiation Dosimetry, Proceedings of the Symposium on Selected Topics in Radiation Dosimetry,* June 7-11, 1960, International Atomic Energy Agency, Vienna, Austria, pp. 3-23. [SRDB Ref ID: 8345]

Napier, B. A., 2006a, "Occupational Environmental Doses for Ames Laboratory, U and Th Operations," memorandum to file, Pacific Northwest National Laboratory, Richland, Washington, November 8. [SRDB Ref ID: 30770]

Napier, B. A., 2006b, *GENII Version 2 Users' Guide*, PNNL-14583, Pacific Northwest National Laboratory, Richland, Washington, August. [SRDB Ref ID: 30771]

Napier, B. A, D. L. Strenge, J. V. Ramsdell Jr., P. W. Eslinger, and C. J. Fosmire, 2004, *GENII Version 2 Software Design Document*, PNNL-14584, Pacific Northwest National Laboratory, Richland, Washington, November. [SRDB Ref ID: 24168]

NCRP (National Council on Radiation Protection and Measurements), 1971, *Protection Against Neutron Radiation*, Report 38, Bethesda, Maryland.

NCRP (National Council on Radiation Protection and Measurements), 1996, *Screening Models for Releases of Radionuclides to Air, Surface Water, and Ground Water*, Report 123, Bethesda, Maryland.

NCRP (National Council on Radiation Protection and Measurements), 2003, *Radiation Protection for Particle Accelerator Facilities*, Report 144, Bethesda, Maryland.

NIOSH (National Institute for Occupational Safety and Health), 1993, *NIOSH Research Issues Workshop: Epidemiologic Use of Nondetectable Values in Radiation Exposure Measurements, Cincinnati, Ohio, September 9 and 10, 1993*, Cincinnati, Ohio. [SRDB Ref ID: 13723].

NIOSH (National Institute for Occupational Safety and Health), 2004, *Estimation of Ingestion Intakes, Technical Information Bulletin*, OCAS-TIB-009, Rev. 0, Office of Compensation and Analysis, Cincinnati, Ohio, April 13.

NIOSH (National Institute for Occupational Safety and Health), 2006, *SEC Petition Evaluation Report, Petition SEC-00038*, Rev. 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, April.

NIOSH (National Institute for Occupational Safety and Health), 2007a, *SEC Petition Evaluation Report, Petition SEC-00075*, Rev. 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, May.

NIOSH (National Institute for Occupational Safety and Health), 2007b, *External Dose Reconstruction Implementation Guideline*, OCAS-IG-001, Rev. 3, Office of Compensation Analysis and Support, Cincinnati, Ohio, November 21.

NIOSH (National Institute for Occupational Safety and Health), 2010a, *Radiation Exposures Covered for Dose Reconstructions Under Part B of the Energy Employees Occupational Illness Compensation Program Act*, DCAS-IG-003, Rev. 1, Division of Compensation Analysis and Support, Cincinnati, Ohio, October.

NIOSH (National Institute for Occupational Safety and Health), 2010b, *SEC Petition Evaluation Report, Petition SEC-00166*, Rev. 0, Division of Compensation Analysis and Support, Cincinnati, Ohio, July.

NIOSH (National Institute for Occupational Safety and Health, 2010c, *Technical Information Bulletin: Best Estimate External Dose Reconstruction for Glovebox Workers,* DCAS-TIB-0010, *Rev 3*, Division of Compensation Analysis and Support, Cincinnati, Ohio, June. [SRDB Ref ID: 85226]

NIOSH (National Institute for Occupational Safety and Health), 2011, SEC Petition Evaluation Report, Petition SEC-00185, Rev. 0, Division of Compensation Analysis and Support, Cincinnati, Ohio June.

NRC (National Research Council), 1989, *Film Badge Dosimetry in Atmospheric Nuclear Tests*, National Academy of Sciences, National Academy Press, Washington, D.C. [SRDB Ref ID: 1905]

ORAUT (Oak Ridge Associated Universities Team), 2004, *Tritium Calculated and Missed Dose Estimates,* ORAUT-OTIB-0011, Rev. 00, Oak Ridge, Tennessee.

ORAUT (Oak Ridge Associated Universities Team), 2005a, *Analysis of Coworker Bioassay Data for Internal Dose Assignment,* ORAUT-OTIB-0019, Rev. 01, Oak Ridge, Tennessee, October 7.

ORAUT (Oak Ridge Associated Universities Team), 2005b, *Interpretation of Dosimetry Data for Assignment of Shallow Dose,* ORAUT-OTIB-0017, Rev 01, Oak Ridge, Tennessee, October 11.

ORAUT (Oak Ridge Associated Universities), 2006a, *Argonne National Laboratory – East – Technical Basis Document*, ORAUT-TKBS-0036-4, Rev. 00, Oak Ridge, Tennessee, February 9.

ORAUT (Oak Ridge Associated Universities Team), 2006b, *Effect of Threshold Energy and Angular Response of NTA Film and Missed Neutron Dose at the Oak Ridge Y-12 Facility,* ORAUT-OTIB-0051, Rev. 00, Oak Ridge, Tennessee, May 15.

ORAUT (Oak Ridge Associated Universities Team), 2006c, *Generating Summary Statistics for Coworker Bioassay Data,* ORAUT-PROC-0095, Rev. 00, Oak Ridge, Tennessee, June 5.

ORAUT (Oak Ridge Associated Universities Team), 2006d, *Technical Information Bulletin: A Standard Complex-Wide Correction Factor for Overestimating External Doses Measured with Film Badge Dosimeters*, ORAUT-OTIB-0010, Rev. 01, Oak Ridge, Tennessee, June 5.

ORAUT (Oak Ridge Associated Universities Team), 2006e, *Technical Information Bulletin: Technical Basis for Estimating the Maximum Plausible Dose to Workers at Atomic Weapons Employer Facilities,* ORAUT-OTIB-0004, *Rev 03 PC-2*, Oak Ridge, Tennessee, December 6.

ORAUT (Oak Ridge Associated Universities Team), 2007a, Technical Information Bulletin: *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses,* ORAUT-OTIB-0054, Rev. 00, PC-1, Oak Ridge, Tennessee, November 19.

ORAUT (Oak Ridge Associated Universities Team), 2007b, *Stanford Linear Accelerator Center Site Profile Document,* ORAUT-TKBS-0051, Rev. 01, Oak Ridge, Tennessee, August 17.

ORAUT (Oak Ridge Associated Universities), 2010a, *Site Profile for the Brookhaven National Laboratory*, ORAUT-TKBS-0048, Rev. 01, Oak Ridge, Tennessee, April 26.

ORAUT (Oak Ridge Associated Universities), 2010b, *Site Profile for the Lawrence Berkeley National Laboratory*, ORAUT-TKBS-0049, Rev. 02, Oak Ridge, Tennessee, May 10 .

ORAUT (Oak Ridge Associated Universities Team), 2010c, *Idaho National Laboratory and Argonne National Laboratory-West – Site Description,* ORAUT-TKBS-0007-2, Rev. 04, Oak Ridge, Tennessee, August 2.

ORAUT (Oak Ridge Associated Universities Team), 2010d, *Technical Basis Document: Basis for Development of an Exposure Matrix for the Mallinckrodt Chemical Company St. Louis Downtown Site and the St. Louis Airport Site, St. Louis, Missouri,* ORAUT-TKBS-0005, *Rev 03*, *Oak Ridge, Tennessee,* November 22.

ORAUT (Oak Ridge Associated Universities Team), 2011a, *Use of Coworker Dosimetry Data for External Dose Assignment*, ORAUT-OTIB-0020, Rev. 03, Oak Ridge, Tennessee, November 14.

ORAUT (Oak Ridge Associated Universities), 2011b, *Guidance on Assigning Occupational X-Ray Dose Under EEOICPA for X-Rays Administered Off Site*, ORAUT-OTIB-0079, Rev. 00, Oak Ridge, Tennessee, January 3 .

Pardue, L. A., N. Goldstein, and E. O. Wollan, 1944, *Photographic Film as a Pocket Radiation Dosimeter,* CH-1553 (A-2223), University of Chicago, Metallurgical Laboratory, Chicago, Illinois, April 8. [SRDB Ref ID: 8599]

Payne, C. S., 1992, *The Ames Project: Administering Classified Research as a Part of the Manhattan Project at Iowa State College, 1942-45*, PhD dissertation, Iowa State University, Ames, Iowa. [SRDB Ref ID: 18851]

PNNL (Pacific Northwest National Laboratory), 2004, *Radiological Work Permits*, Rev. 19, RCP-3.2.02, Richland, Washington, p. 4. [SRDB Ref ID: 25943]

Shleien, B., L. A. Slaback, Jr., and B. K. Birky, editors, 1998, *Handbook of Health Physics and Radiological Health, Third Edition*, Williams & Wilkins, Baltimore, Maryland, January.

Siemens (Siemens Medical Systems), 1996, Radiation Exposure Report, Hoffman Estates, Illinois. [SRDB Ref ID: 26679]

Simpson, S. A., 2001, letter to M. Saar (U.S. Department of Energy, Chicago Operations Office), Iowa State University, Ames Laboratory, Ames, Iowa, May 10. [SRDB Ref ID: 27038, p. 2]

Skoog, A. P., 1952, "Environmental Health Conditions-Ames Laboratory," letter to F. H. Spedding, Ames Laboratory, Iowa State University, Ames, Iowa, September 12. [SRDB Ref ID: 18813, pp. 46–47]

Spedding, F. H., 1952, letter to A. Tammaro (U.S. Atomic Energy Commission, Chicago Operations Office), Ames Laboratory, Iowa State College, Ames, Iowa, September 23. [SRDB Ref ID: 18813, pp. 10–13]

Stone, R. S. (editor), 1951, *Industrial Medicine on the Plutonium Project*, National Nuclear Energy Series, Manhattan Project Technical Section, Division IV, Plutonium Project Record, Vol. 20. [SRDB Ref ID: 739]

Struss, R. G., 1987, letter to Paul Neeson (U.S. Department of Energy, Chicago Operations Office), Iowa State University, Ames Laboratory, Ames, Iowa, May 27. [SRDB Ref ID: 18818]

Thornton, W. T., D. M. Davis, and E. D. Gupton, 1961, *The ORNL Badge Dosimeter and Its Personnel Monitoring Applications*, ORNL-3126, Union Carbide Corporation, Oak Ridge National Laboratory, Oak Ridge, Tennessee. [SRDB Ref ID: 8573]

Tybout, R. A., 1944, letter to F. H. Spedding (Iowa State College), U.S. Army Corps of Engineers, July 17. [SRDB Ref ID: 8959, pp. 8-9]

Voigt, A. F., 1973, "Radioactive Effluent Releases, Ames Laboratory of the U.S. Atomic Energy Commission, Contract No. W-7405-ENG-82," letter to H. N. Miller (U.S. Atomic Energy Commission, Chicago Operations Office), Iowa State University, Ames Laboratory, Ames, Iowa, December. [SRDB Ref ID: 25828, pp. 13-14]

Voigt, A. F., 1981, *Decommissioning of the Ames Laboratory Research Reactor, Final Report*, IS-4789, Iowa State University, Ames Laboratory, Ames, Iowa, December. [SRDB Ref ID: 25824]

Voss, M. D., 1957, letter to P. Van Bemmel and J. Bacon, Iowa State University, Ames Laboratory, Ames, Iowa, February 8. [SRDB Ref ID: 25846]

Voss, M.D., 1963, *Survey of Environmental Radioactivity*, Report No. 1, Iowa State University, Ames Laboratory, Ames, Iowa, December 31. [SRDB Ref ID: 25165]

Voss, M. D., 1971, "Sources of Tritium at the Ames Laboratory", Health Physics Memorandum – Bioassay, Iowa State University, Ames Laboratory, Ames, Iowa, September 1. [SRDB Ref ID: 25796]

Voss, M. D., 1975, *Environmental Monitoring at Ames Laboratory Calendar Year 1974*, IS-3613, Iowa State University, Ames Laboratory, Ames, Iowa, May. [SRDB Ref ID: 25446]

Voss, M. D., 1976, *Environmental Monitoring at Ames Laboratory Calendar Year 1975*, IS-3856, Iowa State University, Ames Laboratory, Ames Iowa, May. [SRDB Ref ID: 25445]

Voss, M. D., 1977, *Environmental Monitoring at Ames Laboratory Calendar Year 1976*, IS-4139, Iowa State University, Ames Laboratory, Ames Iowa, April. [SRDB Ref ID: 25444]

Voss, M. D., 1978, *Environmental Monitoring at Ames Laboratory Calendar Year 1978*, IS-4679, Iowa State University, Ames Laboratory, Ames, Iowa, April 1979. [SRDB Ref ID: 18839]

Voss, M. D., 1979, "History and Current Radiological Condition of the Ames Laboratory," letter to W. E. Mott (U.S. Department of Energy) and A. Wollen (Aerospace Corporation), Iowa State University, Ames Laboratory, Ames, Iowa, December 3. [SRDB Ref ID: 18840, pp. 2-4]

Voss, M. D., 1981, *Environmental Summary for Ames Laboratory Calendar Year 1981*, IS-4798, Iowa State University, Ames Laboratory, Ames, Iowa, April. [SRDB Ref ID: 25429]

Voss, M. D., 1987, "Tritium Exposures/Doses – Committed Dose – Applied Science Center," memorandum to R. Struss and L. Mathison, Iowa State University, Ames Laboratory, Ames Iowa, May 8. [SRDB Ref ID: 18817]

Wilson, R. H., 1960, *Detection Level of the Film Badge System*, HW-67697, General Electric, Hanford Atomic Products Operation, Richland, Washington, November 28. [SRDB Ref ID: 15403]

Wilson, R. H., 1987, *Historical Review of Personnel Dosimetry Development and its Use in Radiation Protection Programs at Hanford, 1944 to the 1980s*, PNL-6125, Battelle Memorial Institute, Pacific Northwest Laboratory, Richland, Washington, February. [SRDB Ref ID: 262]

Wilson, R. H., J. J. Fix, W. V. Baumgartner, and L. L. Nichols, 1990, *Description and Evaluation of the Hanford Personnel Dosimeter Program from 1944 through 1989*, PNL-7447, Battelle Memorial Institute, Pacific Northwest Laboratory, Richland, Washington, September. [SRDB Ref ID: 4793]

GLOSSARY

absorbed dose
Amount of energy (ergs or joules) deposited in a substance by ionizing radiation per unit mass (grams or kilograms) of the substance and measured in units of rads or grays See *dose*.

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

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

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

background radiation
Radiation from cosmic sources, naturally occurring radioactive materials including naturally occurring radon, and global fallout from the testing of nuclear explosives. Background radiation does not include radiation from source, byproduct, or Special Nuclear Materials regulated by the U.S. Nuclear Regulatory Commission. The average individual exposure from background radiation is about 360 millirem per year.

becquerel (Bq)
International System unit of radioactivity equal to 1 disintegration per second; 1 curie equals 37 billion (3.7×10^{10}) becquerels.

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

bioassay
Determination of kinds, quantities, or concentrations, and in some cases locations of radioactive material in a living body, whether by direct measurement (*in vivo* measurement) or by analysis and evaluation of materials excreted or removed from the body (*in vitro* measurement). Also called *radiobioassay*.

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

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

contamination

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

curie (Ci)

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

deep dose equivalent (DDE, $H_{d,}$ $H_p(10)$)

Dose equivalent in units of rem or sievert for a 1-centimeter depth in tissue (1,000 milligrams per square centimeter). See *dose*.

dose

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

dose equivalent (*H*, DE)

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

dosimeter

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

dosimetry

Measurement and calculation of internal and external radiation doses.

dosimetry system

System for assessment of received radiation dose. This includes the fabrication, assignment, and processing of external dosimeters, and/or the collection and analysis of bioassay samples, and the interpretation and documentation of the results.

enriched uranium

Uranium in which processing has increased the proportion of ^{235}U to ^{238}U to above the natural level of 0.7%. Reactor-grade uranium is usually about 3.5% ^{235}U; weapons-grade uranium contains greater than 90% ^{235}U.

exposure

In general, the act of being exposed to ionizing radiation. See *acute exposure* and *chronic exposure*.

film

In the context of external dosimetry, radiation-sensitive photographic film in a light-tight wrapping. See *film dosimeter*.

film dosimeter

Package of film for measurement of ionizing radiation exposure for personnel monitoring purposes. A film dosimeter can contain two or three films of different sensitivities, and it can contain one or more filters that shield parts of the film from certain types of radiation. When

developed, the film has an image caused by radiation measurable with an optical densitometer. Also called *film badge*.

fission products

(1) Radionuclides produced by fission or by the subsequent radioactive decay of radionuclides. (2) Fragments other than neutrons that result from the splitting of an atomic nucleus.

gamma radiation

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

gray (Gy)

International System unit of absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium; 1 Gy equals 1 joule per kilogram or 100 rads.

high-efficiency particulate air (HEPA) filter

Dense filter that removes contaminants from air flows before return to the working environment or discharge to the outside air (exhaust).

hot cell

Shielded laboratory for handling of radioactive materials with the aid of remotely operated manipulators. The walls and windows are made of materials that protect workers from radiation.

ionizing radiation

Radiation of high enough energy to remove an electron from a struck atom and leave behind a positively charged ion. High enough doses of ionizing radiation can cause cellular damage. Ionizing particles include alpha particles, beta particles, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, photoelectrons, Compton electrons, positron/negatron pairs from photon radiation, and scattered nuclei from fast neutrons.

***in vitro* bioassay**

Measurements to determine the presence of or to estimate the amount of radioactive material in the excreta or in other biological materials removed from the body.

***in vivo* bioassay**

The measurements of radioactive material in the human body utilizing instrumentation that detects radiation emitted from the radioactive material in the body.

intake

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

internal dose or exposure

Dose received from radioactive material in the body.

internal dose assessment

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

minimum detectable activity or amount (MDA)

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

minimum detection level (MDL)

Lowest amount (mass or activity) of a substance detectable by a specific instrument or process. Often assumed to be the level at which a dose is detected at the 2-sigma level (i.e., 95% of the time).

missed dose

(1) In relation to external dose, dose to monitored workers that was not measured or recorded due to such factors as a missing or damaged dosimeter or a result below the detection limits of the dosimeter. Missed dose is especially important in the early years of radiation monitoring, when relatively high detection limits were combined with short exchange periods. (2) In relation to internal dose, potential dose that could have been received by a bioassay program participant but, because of limitations in the monitoring system, was undetected.

monitoring

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

natural uranium (U, U-nat, NU)

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

neutron

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

neutron film dosimeter

Film dosimeter with a nuclear track emulsion, type A, film packet.

nuclear track emulsion, Type A (NTA)

Film sensitive to fast neutrons made by Eastman Kodak. The developed image has tracks caused by neutrons that visible under oil immersion with about 1,000-power magnification.

occupational dose

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

occupational exposure

Exposure to radiation and/or to radioactive material from sources of radiation in a restricted area or in the course of employment in which the individual's assigned duties. Occupational exposure does not include exposure to background radiation, as a patient from medical

practices, from voluntary participation in medical research programs, or as a member of the public.

occupational medical dose

Dose from X-ray procedures performed for medical screening of workers as part of an occupational health program. Doses from X-rays used to diagnose diseases or injuries, even if incurred on the job, are not considered occupational and are therefore not eligible to be included in dose reconstruction under EEOICPA.

personal dose equivalent [$H_p(d)$]

Dose equivalent in units of rem or sievert in soft tissue below a specified point on the body at an appropriate depth *d*. The depths selected for personal dosimetry are 0.07 millimeters (7 milligrams per square centimeter) and 10 millimeters (1,000 milligrams per square centimeter), respectively, for the skin (shallow) and whole-body (deep) doses. These are noted as $H_p(0.07)$ and $H_p(10)$, respectively. The International Commission on Radiological Measurement and Units recommended $H_p(d)$ in 1993 as dose quantity for radiological protection.

photon

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

photon radiation

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

pocket ionization chamber (PIC)

Cylindrical monitoring device commonly clipped to the outer clothing of an individual to measure ionizing radiation. A PIC may be self-reading or require the use of a outside device to be able to read the dosimeter. Also called pencil, pocket pencil, pencil dosimeter, and pocket dosimeter.

probability of causation (POC)

For dose reconstruction under the Energy Employees Occupational Illness Compensation Program Act, the percent likelihood that a worker incurred a particular cancer from occupational exposure to radiation.

progeny

Nuclides that result from decay of other nuclides. Also called decay products and formerly called daughter products.

rad

Traditional unit for expressing absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium. A dose of 1 rad is equivalent to the absorption of 100 ergs per gram (0.01 joule per kilogram) of absorbing tissue. The rad has been replaced by the gray in the International System of Units (100 rads = 1 gray). The word derives from radiation absorbed dose.

radiation

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

radioactivity

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

radiograph

Static images produced on radiographic film by gamma rays or X-rays after passing through matter. In the context of EEOICPA, radiographs are X-ray images of the various parts of the body used to screen for disease.

rem

Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The average American receives 360 millirem a year from background radiation. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

roentgen (R)

Unit of photon (gamma or X-ray) exposure for which the resultant ionization liberates a positive or negative charge equal to 2.58×10^{-4} coulomb per kilogram (or 1 electrostatic unit of electricity per cubic centimeter) of dry air at 0°C and standard atmospheric pressure. An exposure of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher energy photons (generally greater than 100 kiloelectron-volts).

routine monitoring

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

shallow absorbed dose (D_s)

Absorbed dose at a depth of 0.07 centimeters (7 milligrams per square centimeter) in a material of specified geometry and composition.

shallow dose equivalent [SDE, H_s, $H_p(0.07)$]

Dose equivalent in units of rem or sievert at a depth of 0.07 millimeters (7 milligrams per square centimeter) in tissue equal to the sum of the penetrating and nonpenetrating doses.

special monitoring

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

synchrotron

Roughly circular particle accelerator in which the particles travel in synchronized bunches at fixed radius.

thermoluminescent dosimeter (TLD)

Device for measuring radiation dose that consists of a holder containing solid chips of material that, when heated by radiation, release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

unmonitored dose

Potential unrecorded dose that could have resulted because a worker was not monitored. See *missed dose*.

uranium (U)

Heavy, metallic, and radioactive element with atomic number 92. Most natural uranium as found in ores is ^{238}U with trace levels of other isotopes. Uranium-235 (0.7% of natural uranium) is fissile by itself and used in nuclear weapons as well as reactors. Uranium-238 (99.3% of natural uranium) is fissionable by fast neutrons and used in nuclear reactors. Natural uranium contains a minute amount of ^{234}U. See *enriched uranium* and *natural uranium*.

whole-body (WB) dose

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

X-ray

(1) See *X-ray radiation*. (2) See *radiograph*.

X-ray radiation

Penetrating electromagnetic radiation (photons) of short wavelength (0.001 to 10 nanometers) and energy less than 250 kiloelectron-volts. X-rays usually come from excitation of the electron field around certain nuclei. Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.