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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

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
ALRR	Ames Laboratory Research Reactor
AP	anterior-posterior
AWE	atomic weapons employer
BNL	Brookhaven National Laboratory
cGy	centigray
Ci	curie
cm	centimeter
d	day
D&D	decontamination and decommissioning
DOE	U.S. Department of Energy
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ESE	entrance skin exposure
ft	foot
g	gram
GSD	geometric standard deviation
Gy	gray
HEPA	high-efficiency particulate air
HFBR	High Flux Beam Reactor
hr	hour
HVL	half-value layer
IADPH	Iowa State Department of Public Health
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
m	meter
MDA	minimum detectable activity
MDL	minimum detection level
MeV	megaelectron-volt, 1 million electron-volts

mg	milligram
mGy	milligray
mi	mile
mL	milliliter
mm	millimeter
mo	month
mR	milliroentgen
mrad	millirad
MRD	minimum recordable dose
mrem	millirem
mrep	millirep
MTR	Materials Test Reactor
MW	megawatt
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
PA	posterior-anterior
pCi	picocurie
POC	probability of causation
R&D	research and development
s	second
S/N	serial number
SEC	Special Exposure Cohort
SLAC	Stanford Linear Accelerator Center
SRDB Ref ID	Site Research Database Reference Identification (number)
TASF	Technical and Administrative Services Facility
TBD	technical basis document
TLD	thermoluminescent dosimeter
U.S.C.	United States Code
wk	week
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) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work.

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

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

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

1.2 SCOPE

This site profile provides information about U.S. Atomic Energy Commission (AEC) 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 7.0.

1.3 SPECIAL EXPOSURE COHORT

The Secretary of Health and Human Services has designated a class of employees at Ames Laboratory for addition to the Special Exposure Cohort (SEC) as authorized under the EEOICPA. This includes DOE employees or DOE contractor or subcontractor employees who worked at the Ames Laboratory in one or more of the following facilities/locations: 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 the 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 are monitored or should have been monitored.

NIOSH evaluated the feasibility for completing dose reconstructions for employees at Ames Laboratory from January 1, 1942, through December 31, 1954. 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-00038) class of employees (NIOSH 2006a). Table 1-1 summarizes the results of the feasibility findings for each exposure source for the period from January 1, 1942, through December 31, 1954.

Table 1-1. Feasibility findings for SEC-00038. Source: NIOSH 2006a, 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	

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 (Spedding 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, at the beginning of the Manhattan Project, the most pressing problem was the preparation of large amounts of pure uranium metal (Ames 1960). 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 1960). 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 (Ames 1960). Between mid-1942 and August 1945, more than 1,000 tons of pure uranium metal were supplied to the Manhattan Project (Ames 1960).

Once the potential need for thorium metal was recognized, the Ames Project began to develop methods for purifying thorium in 1943 (Ames 1960). 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 1960).

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 the Physical Chemistry Annex 1, the Chemistry Building (now Gillman Hall), and the Physics Building (now Physics Hall) (Ames 1967). In 1944, a new building, called Physical Chemistry Annex 2, was constructed to house operations to recover uranium from scrap material (Spedding 1947).

In November 1945, the buildings used by the Ames Project were designated as the Institute for Atomic Research (Ames 1962). 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 1962). The Metals Development Building was added in 1960 (Ames 1960). In May 1947, the AEC established a major research facility at Ames, known as Ames Laboratory (Ames 1962), 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 later 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.

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.

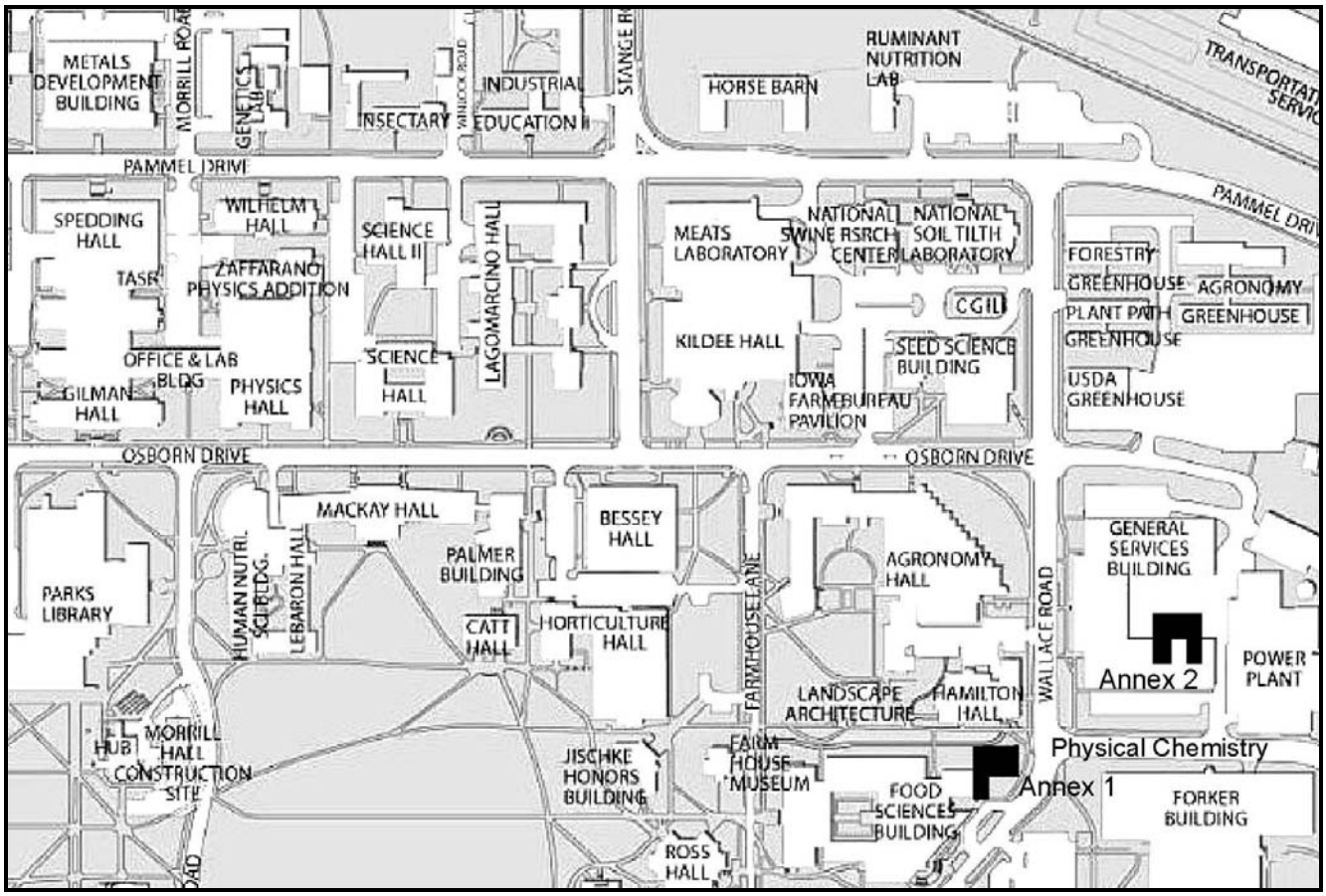


Figure 2-1. ISU campus map showing locations of Ames Laboratory buildings. The former sites of Physical Chemistry Annex 1 and Annex 2 are also shown. Source: ISU (2006).

2.1.1 Physical Chemistry Annex 1

The production of uranium metal was conducted in the Physical Chemistry Annex 1 building, which was an old wooden structure east of the Dairy Industries building and west of Wallace Road (Karsjen 2003). 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 1960). The Physical Chemistry Annex 1 building 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 the Chemistry Building (Spedding 1947). Uranium production operations were moved to the 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 (Spedding 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 1962). 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 1960). The Chemistry Building was decontaminated and surveyed in May 1976 (Voss 1979).

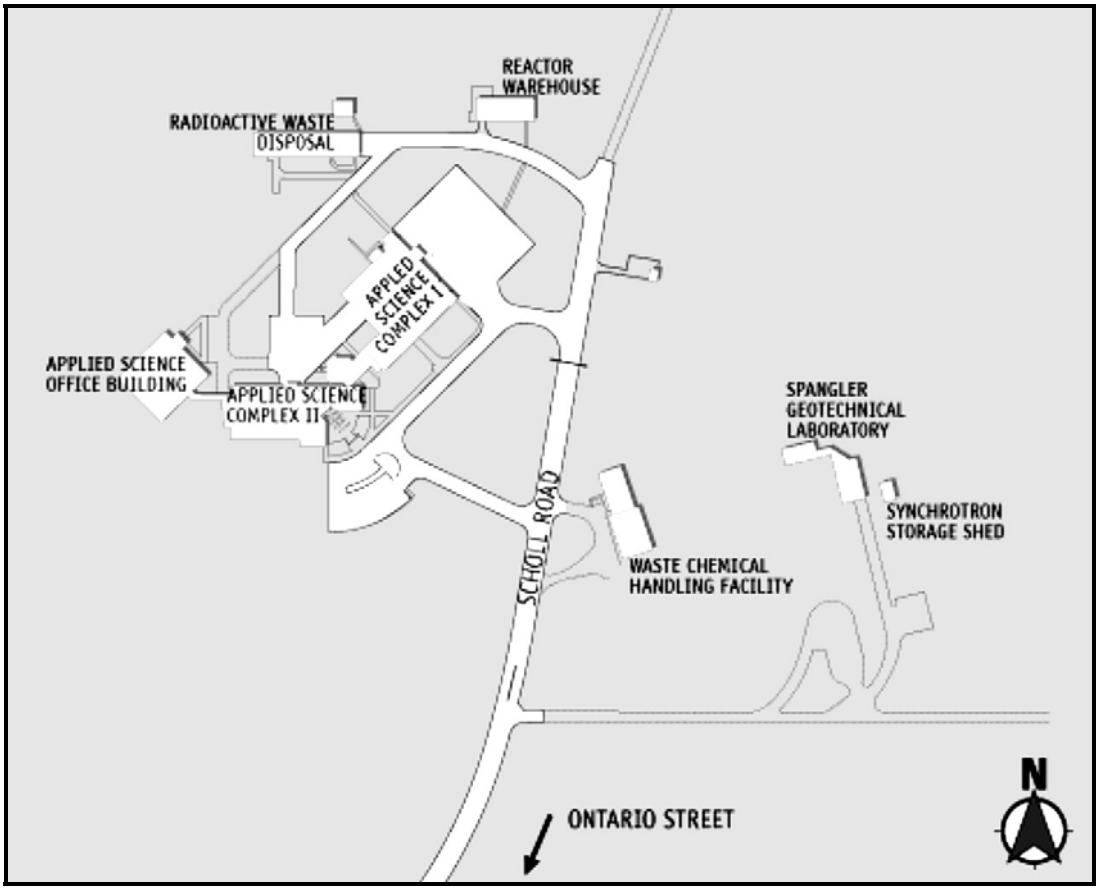


Figure 2-2. ISU campus map showing locations of reactor facilities and Synchrotron Building. Source: ISU (2006).

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, 1962). 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 1962).

2.1.4 Physical Chemistry Annex 2

The Physical Chemistry Annex 2 Building 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 (Spedding 1947). Operations in this building through December 1945 produced more than 300 tons of recovered uranium metal (Spedding 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 houses research directed toward the development of special metals and alloys used in nuclear energy projects (Ames 1962). Zircaloy was initially developed at Ames Laboratory as part of a basic study of the zirconium-tin alloy phase diagram (Ames 1962). The subject of reactor coolants was studied, as were the 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 (Ames 1962). A glovebox line in the Metallurgy Building was used to study the behavior of plutonium in molten metal systems (Ames 1962).

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 (Hokel et al. 1998). Poor contamination control practices and poor ventilation contributed to contamination of the building (Hokel et al. 1998). However, contamination levels have been reduced by mitigation, decontamination, remodeling, and renovation projects (Hokel et al. 1998). 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 the 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 1962). 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 (Ames 1967). 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 electronic structure of metals; and the separation, preparation, and measurement of properties of the rare earth metals (Ames 1967). The initial research on liquid metal coolants was done at Ames Laboratory in an engineering sodium test loop utilized in corrosion, fluid-flow, and heat transfer studies with liquid sodium (Ames 1967).

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 1962). 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 structure and to provide radioisotopes for nuclear spectroscopy (Ames 1962). 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 (Ames 1960). Its missions were to conduct process development research on larger-than-laboratory scale and to evaluate commercial feasibility of the processes developed (Ames 1960). 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 (Ames 1960). 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 1960). 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). Numerous other tubes and thimbles provided access to the reactor core for irradiation experiments. Research activities included radiation damage studies, the determination of the crystalline structure of solids, the determination of mechanical properties of reactor materials, and analysis of the decay products of nuclear fission (Ames 1967). 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 (Ames 1967). 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 (Voigt 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 by calcium metal. Finely ground uranium tetrafluoride 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 (Spedding 1947). Although more complex, the uranium production process was improved by replacing the calcium reagent with magnesium metal (Spedding 1947).

Most of the uranium metal production operations were conducted in the Physical Chemistry Annex 1 Building, 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 the Physical Chemistry Annex 2 Building from early 1944 to December 1945 (Spedding 1947). Radiation exposures to workers in this building during this period were similar to the uranium exposures in Annex 1, and these exposures 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 (Spedding 1947). Thorium tetrafluoride 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 (Spedding 1947). Production of thorium metal and thorium compounds continued until April 1953, when thorium production operations were turned over to industry (Ames 1960).

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 thorium tetrafluoride, and crushing the thorium tetrafluoride to a fine powder (Spedding 1947). This process was used throughout thorium production operations.

Thorium production operations, which were conducted in the Physical Chemistry Annex 1 Building after uranium operations ended in 1945, continued until 1949 or 1950, when the operation and equipment was moved to the new Metallurgy Building (Wilhelm Hall) (Ames 1960). 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.

The molding and machining of beryllium oxide crucibles raised a concern about toxic dust (Spedding 1947), which causes the serious lung disease berylliosis (Van Bemmel 1957). Ames Laboratory staff members were required to have physical examinations and chest X-rays every 6 mo (Van Bemmel 1957). The subject of chest X-rays is addressed in more detail in Section 3.

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

Table 2-1. Timeline of AEC activities at Ames Laboratory, 1942 to 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 1945	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 (Spedding 1947). 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 at Ames Laboratory (Spedding 1947).

2.3 SOURCE TERMS AND PERSONNEL MONITORING

Ames Laboratory staff members were exposed to a number of radiation sources, including medical X-rays, environmental effluents, radioactive intakes, and external radiation dose. Routine chest X-rays are discussed in the previous section and are addressed in more detail in Section 3.

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 could have contributed to the occupational environmental dose received by Ames Laboratory workers who were involved in the incident response and 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 (Spedding 1947). The principal sources of surface contamination and airborne dust were the processes of grinding uranium fluoride into a fine powder, transferring the uranium fluoride powder from the grinder, and then 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 (Spedding 1947). 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 (Spedding 1947).

There were frequent small explosions and fires associated with the uranium and thorium production operations (Payne 1992). 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 at Ames Laboratory 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 the 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 the uranium metals production, uranium scrap recovery, and uranium machining processes (NIOSH 2006b). 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 2006b). 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 examinations were conducted for the staff of the Ames Project (later the Ames Laboratory) as prescribed by the Medical Services of the Manhattan Project (Van Horn 1943; Wirth 1946). Preemployment examinations included a chest X-ray and urine analysis. Annual physical examinations also included a chest X-ray, and annual chest X-ray films dated back to 1946 were examined to verify an annual frequency. When work with beryllium oxide crucibles began in 1944, there was concern about beryllium dust and the potential for berylliosis (Spedding 1947). It is possible that more frequent chest X-rays were required for beryllium workers, but this could not be verified because no films or records from before 1946 were found [1]. X-ray film records from the early and mid-1950s indicated that some (possibly beryllium) workers did receive chest X-rays on a quarterly frequency; other workers had chest X-rays on an annual or semiannual frequency. In 1957, the frequency of chest X-rays was changed to twice per year for all Ames Laboratory staff members (Van Bemmel 1957). X-ray film records indicated that the frequency of chest X-rays was changed to annual in 1960 and biannual in 1982. The general frequencies of chest X-rays during various periods are summarized in Table 3-1. Although the projection for most chest X-rays was posterior-anterior (PA), some lateral (LAT) projections were also taken. The actual dates and projections of chest X-rays for workers were reported by Ames Laboratory in response to data requests for each claim.

Table 3-1. Chest X-ray frequencies.

Period	Beryllium workers[a]	Other workers
1942–1943		Annual
1944–1949	Quarterly	Annual
1950–1956	Quarterly	Semiannual
1957–1960	Semiannual	Semiannual
1960–1981	Annual	Annual
1982–present	Biannual	Biannual

a. If a claim mentions beryllium at any time between 1944 and 1953, the claimant should be considered a beryllium worker.

X-ray examinations of Ames Laboratory staff members were conducted at the Iowa State Student Health Center/College Hospital. The radiation dose delivered by the medical X-ray machine was determined in February 1957 (Voss 1957a). The measurements indicated the exposure dose per X-ray film ranged from 40 to 65 mrem, depending on the body thickness of the worker (Voss 1957a). All X-ray films examined in the archives at Ames Laboratory were 14-in. by 17-in. films; there was no evidence of any X-rays being taken with photofluorographic X-ray equipment.

In 1978, a Picker Model 851 (Serial Number 1101) X-ray machine was installed at the Iowa State Student Health Center. The Iowa State Department of Public Health (IADPH) conducted surveys (CRCPD 2005; HHS 1995) of this unit on February 2, 1988 (IADPH 1988) and April 20, 1995 (IADPH 1995). A Continental Model 6626.235 (Serial Number 971941) X-ray machine was installed on May 12, 1997; it was surveyed by IADPH on September 17, 1997 (IADPH 1997). The occupational medical dose for each chest X-ray delivered to Ames Laboratory staff members is estimated and summarized in Table 3-2 for specific periods.

Table 3-2. Chest X-ray doses delivered to Ames Laboratory staff members.

Period	X-ray machine	Half-value layer	Entrance kerma
1942–1956	Unknown	2.5 mm Al[a]	0.2 cGy[a]
1957–1977	Unknown	2.5 mm Al[a]	0.065 cGy[b]
1978–1996	Picker	3.0 mm Al[c]	0.020 cGy[e]
1997–present	Continental	3.5 mm Al[d]	0.016 cGy[f]

a. Default value from ORAUT (2005a).
b. Based on measurements found in Voss survey, February 1957 (Voss 1957a).
c. Based on measurements found in 1988 IADPH survey (IADPH 1988).
d. Based on measurements found in 1997 IADPH survey (IADPH 1997).
e. Based on 75th percentile of entrance skin exposures (ESEs) collected nationwide in nonhospital facilities in 1986 (Spelic 2006).
f. Based on 75th percentile of ESEs collected nationwide in all facility types in 1994 (Spelic 2006).

3.1 ORGAN DOSE CALCULATIONS

Guidance in ORAUT (2005a) was used to determine organ doses. For conventional chest X-rays, that document recommends a default entrance kerma of 0.2 rem for pre-1970 examinations (ORAUT 2005a, Table 3-4). In addition, ORAUT (2005a, Table 6-5) provides factors for converting kerma to organ dose for pre-1970 X-ray machines that might have had limited collimation. Because nothing is known about the type of X-ray machines used before 1978, it is favorable to the claimant to assume limited collimation. It is also assumed that the projection is PA. The dose conversion factors and organ doses from ORAUT (2005a) are listed in Tables 3-3 through 3-6 for 1942 to 1956, 1957 to 1977, 1978 to 1996, and 1997 to the present, respectively.

Table 3-3. Organ doses from PA chest X-rays, 1942 to 1956.

Organ	PA dose conversion factor (mGy per Gy air kerma) HVL 2.5 mm Al	Organ dose (rem)[a]
Thyroid	174	3.48E-2
Eye/brain	32	6.40E-3
Ovaries	N/A[b]	2.50E-2
Liver/gall bladder/spleen	451	9.02E-2
Urinary bladder	N/A[b]	2.50E-2
Colon/rectum	N/A[b]	2.50E-2
Testes	N/A[b]	5.00E-3
Lungs	451	9.02E-2
Thymus	451	9.02E-2
Esophagus	451	9.02E-2
Stomach	451	9.02E-2
Bone surfaces	451	9.02E-2
Breast	49	9.80E-3
Uterus	N/A[b]	2.50E-2
Bone marrow	92	1.84E-2
Skin[c]	N/A	2.70E-1

a. From ORAUT (2005a, Table 6-5, pre-1970).
b. N/A = not applicable. Organ dose based on measured values from Webster and Merrill (1957).
c. Included backscatter factor of 1.35 from NCRP Report 102 (NCRP 1989, Table B-8).

Table 3-4. Organ doses from PA chest X-rays, 1957 to 1977.

Organ	PA dose conversion factor (mGy per Gy air kerma) HVL 2.5 mm Al	Organ dose (rem)
Thyroid	174	1.13E-2
Eye/brain	32	2.08E-3
Ovaries	N/A[a]	2.50E-2
Liver/gall bladder/spleen	451	2.93E-2
Urinary bladder	N/A[a]	2.50E-2
Colon/rectum	N/A[a]	2.50E-2
Testes	N/A[a]	5.00E-3
Lungs	451	2.93E-2
Thymus	451	2.93E-2
Esophagus	451	2.93E-2
Stomach	451	2.93E-2
Bone surfaces	451	2.93E-2
Breast	49	3.19E-3
Uterus	N/A[a]	2.50E-2
Bone marrow	92	5.98E-3
Skin[b]	N/A	8.78E-2

a. N/A = not applicable. Organ dose based on measured values from Webster and Merrill (1957).
b. Included backscatter factor of 1.35 from NCRP Report 102 (NCRP 1989, Table B-8).

Table 3-5. Organ doses from PA chest X-rays, 1978 to 1996.

Organ	PA dose conversion factor (mGy per Gy air kerma) HVL 3.0 mm Al	Organ dose (rem)
Thyroid	46	9.20E-4
Eye/brain	46	9.20E-4
Ovaries	1.8	3.60E-5
Liver/gall bladder/spleen	535	1.07E-2
Urinary bladder	1.8	3.60E-5
Colon/rectum	1.8	3.60E-5
Testes	0.01	2.00E-7
Lungs	535	1.07E-2
Thymus	535	1.07E-2
Esophagus	535	1.07E-2
Stomach	535	1.07E-2
Bone surfaces	535	1.07E-2
Breast	69	1.38E-3
Uterus	2.3	4.60E-5
Bone marrow	117	2.34E-3
Skin[a]	N/A[b]	2.80E-2

a. Included backscatter factor of 1.4 from NCRP Report 102 (NCRP 1989, Table B-8).
b. N/A = not applicable.

Table 3-6. Organ doses from PA chest X-rays, 1997 to present.

Organ	PA dose conversion factor (mGy per Gy air kerma) HVL 3.5 mm Al	Organ dose (rem)
Thyroid	62	9.92E-4
Eye/brain	62	9.92E-4
Ovaries	3.2	5.12E-5
Liver/gall bladder/spleen	610	9.76E-3
Urinary bladder	3.2	5.12E-5
Colon/rectum	3.2	5.12E-5
Testes	0.01	1.60E-7
Lungs	610	9.76E-3
Thymus	610	9.76E-3
Esophagus	610	9.76E-3
Stomach	610	9.76E-3
Bone surfaces	610	9.76E-3
Breast	91	1.46E-3
Uterus	3.0	4.80E-5
Bone marrow	146	2.34E-3
Skin[a]	N/A[b]	2.24E-2

a. Included backscatter factor of 1.4 from NCRP Report 102 (NCRP 1989, Table B-8).
b. N/A = not applicable.

4.0 OCCUPATIONAL ENVIRONMENTAL DOSE - INTRODUCTION

The occupational environmental dose refers to the dose received by workers on the site but outside facilities (e.g., buildings). These doses can be internal and/or external depending on the characteristics of the individual radionuclides. Radionuclides used at Ames Laboratory during various periods included uranium, plutonium, thorium, and small amounts of other radionuclides used in the R&D program (Spedding 1947). Tritium, argon, and krypton were released at the Ames Laboratory Research Reactor (ALRR) (Voigt 1973). 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, 2006b), and it was not calculated from environmental media concentrations until 1962 (Voss 1963a). Sources of potential environmental exposures (releases to the environment) were not measured until 1962 (Voss 1963a).

Different activities were carried out during distinct periods of the 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 the Physical Chemistry Annex 1 Building (see Figure 2-1) from mid-1942 through August 5, 1945 (Karsjen 2003). Uranium scrap recovery occurred in the Physical Chemistry Annex 2 Building (see Figure 2-1) from early 1944 through December 1945, and some operations continued until 1954. Thorium production occurred first in the Physical Chemistry Annex 1 Building from 1944 through 1949 when the operation moved to the new Metallurgical Building, now 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 onsite 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 (Voss 1963b).

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 1962).

There were room and hood ventilation stacks on the facilities but no filters on either (Skoog 1952, Section e; Spedding 1952). 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 (to be modeled as ^{234}U, type M or S) was 5 pCi. This intake applies outside the Physical Chemistry Annex 1 for July 1942 through August 1945 and outside the 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 the Physical Chemistry 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 2006).

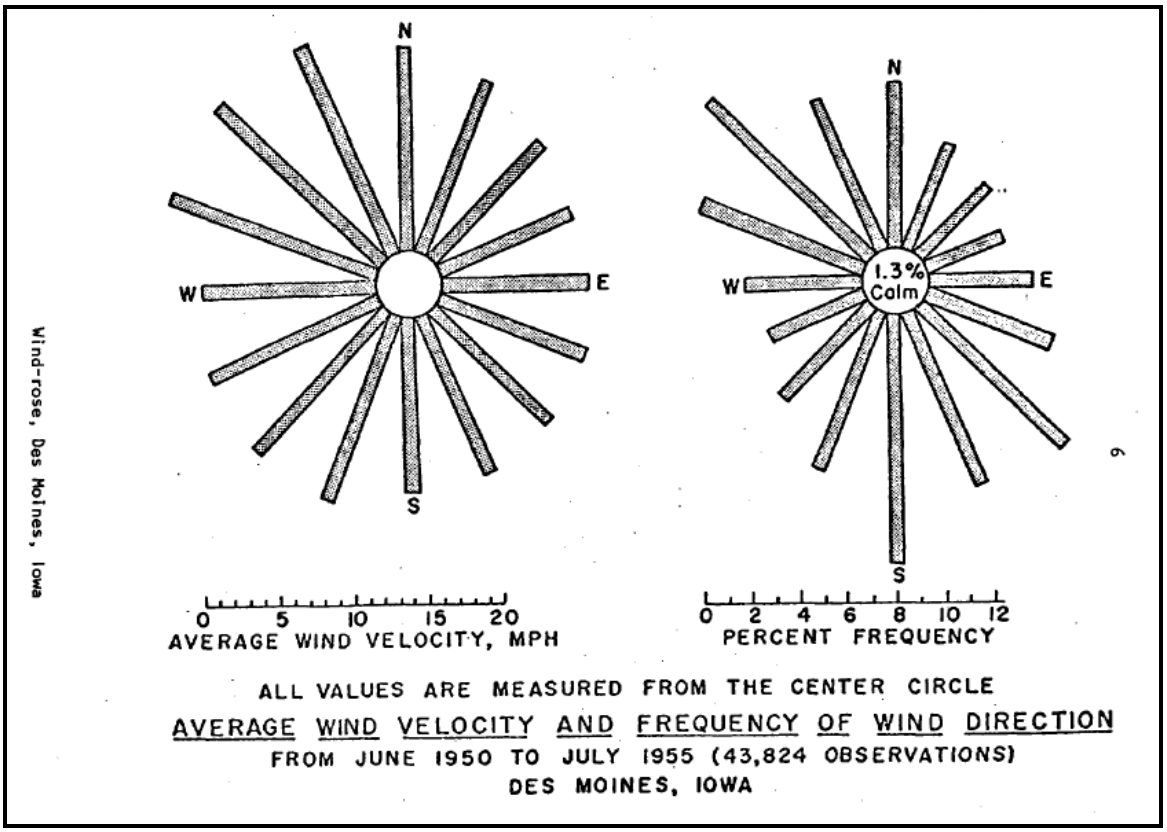


Figure 4-1. Wind rose for Des Moines, Iowa, 1950 to 1955. Source: Voss (1981).

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 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 [2]. No routine measurements of direct gamma or neutron radiation were made outside the buildings, but a detailed survey of the facility, including fence-line gamma dose rates, was made on May 16, 1961, during a special synchrotron operation (Ames 1961). The results of that survey are listed in Table 4-1 (Ames 1961).

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

Location	Dose rate, mrem/hr[a]	Location	Dose rate, mrem/h[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 0.5-1.0 mrem/hr.

The fence line gamma dose rates measured in the radiation survey (Ames 1961) were worst case and were produced with the maximum beam current on a target and direction that would produce maximum dose rates at the fence [3]. 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 (tens of mR/hr) (Ames 1961). Dosimeter results for Synchrotron staff 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 set-up, maintenance, system start-up, etc., so the maximum operating time would have been about 2,000 hr/yr [4]. 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 given in Ames (1961) [5].

A topographical map that was part of the 1961 radiation survey (Ames 1961) indicates that 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 AMES LABORATORY RESEARCH REACTOR 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, which 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 were used to estimate the environmental dose to off-site 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 fence line location with the highest dose from gaseous effluents, the average annual dose to a person at this location for the entire year (8,760 hr) was estimated to be 4.2 mrem from ^{41}Ar during the years of reactor operation. An off-site 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 fence line location with the highest concentration of tritium effluents, the average annual intake of tritium to a person at this location for the entire year (8,760 hr) was estimated to be about 2.9 µCi/yr during the years of reactor operation. An off-site 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 2700 pCi/d.

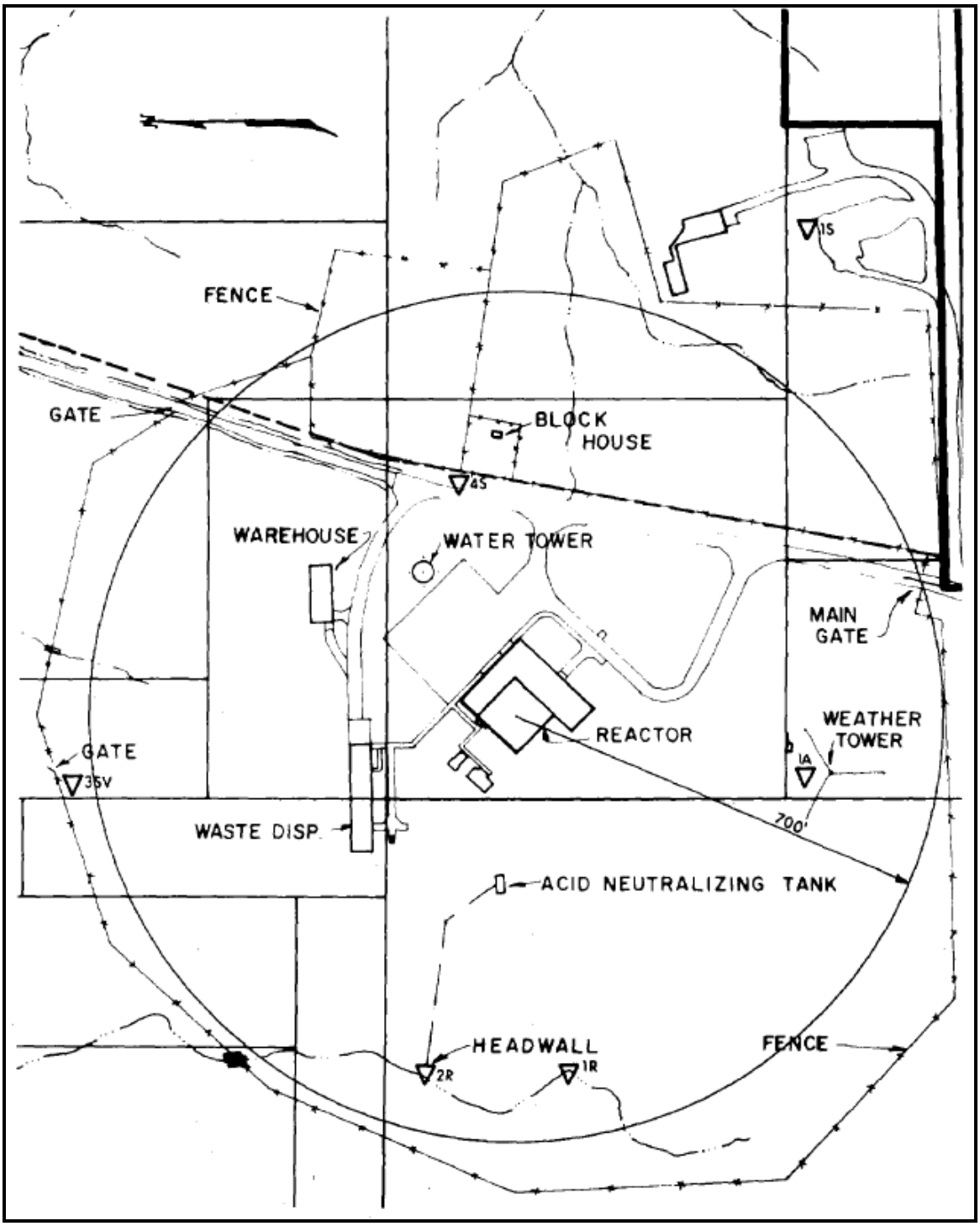


Figure 4-2. Ames Laboratory Research Reactor Site. Source: Voss (1974).

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 1960). Buildings owned by the government (AEC, the Energy Research and

Development Administration, 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).

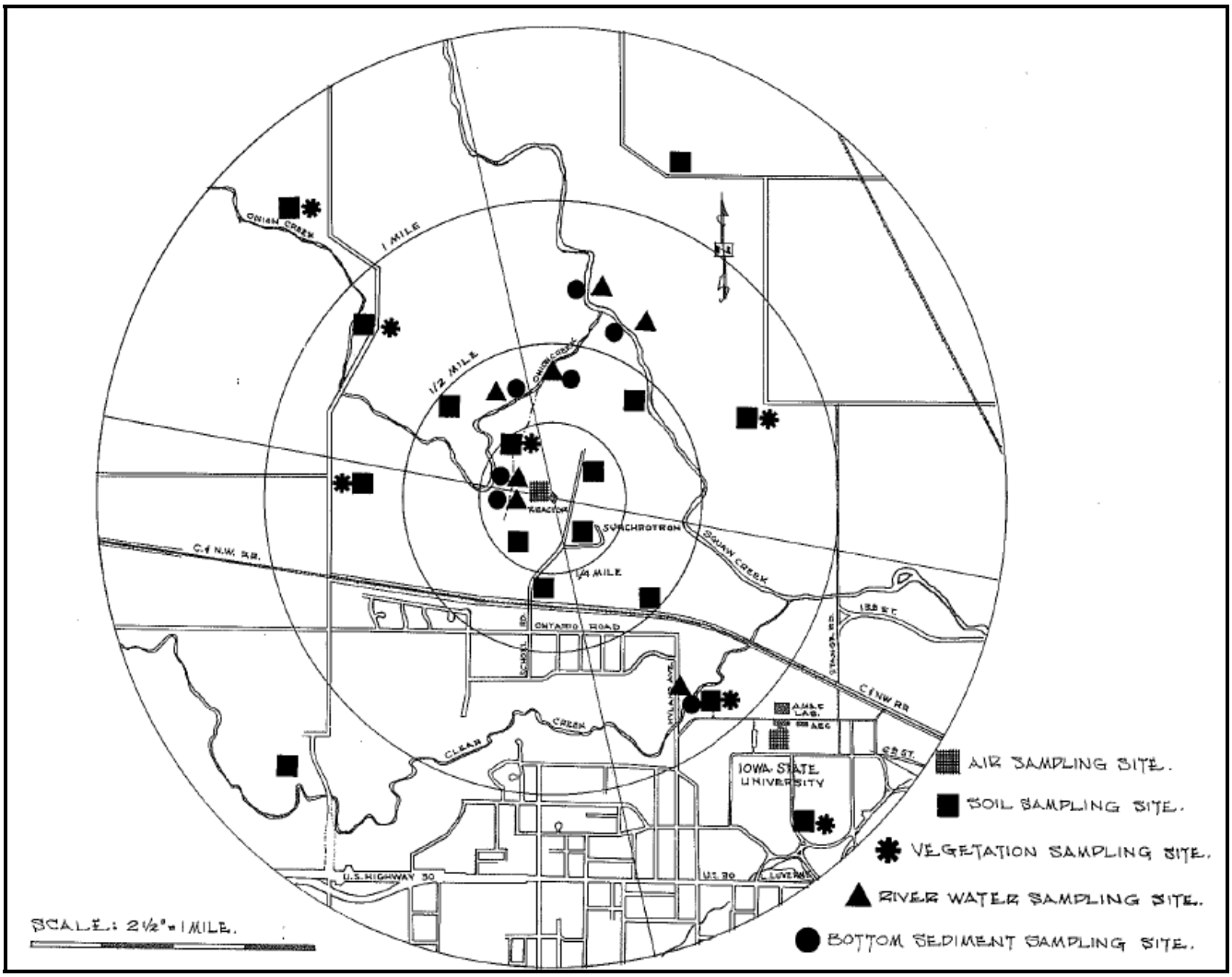


Figure 4-3. Environmental monitoring in the vicinity of the ALRR. Source: Voss (1975).

The Ames Laboratory facilities consist of (Ames 1989):

- 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 1989):

- 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 the staff, students, and public traffic [6]. 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. There was little, if any, exposure of these personnel in that the quantities of radioactive materials involved in the R&D work were small and little, if any, were released to the environment among the facilities (Skoog 1952, p.4).

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 to the production facilities, the releases from the R&D facilities were assumed to be one one-hundredth of the releases from the production facilities [7]. 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 other somewhat related sites (Argonne National Laboratory–East, Brookhaven National Laboratory, and Lawrence Berkeley National Laboratory) was made. Neither Argonne nor Brookhaven listed any environmental intakes of uranium or ^{232}Th (ORAUT 2006a, 2006e). The Berkeley Laboratory 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 2006b). 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. So the proposed uranium intake is comparable to the second highest gross alpha intake from the Berkeley Laboratory, 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 [8]. 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 [9].

A hot laboratory was operated in the Chemistry Building but 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 (Spedding 1947), at which time plutonium R&D and the staff who conducted it were transferred to Los Alamos (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" (Spedding 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 lab of 50 Ci/yr
- An airborne fraction of 0.002, for boiling liquids
- Filtration efficiency for average particle sizes of 99.5% (high-efficiency particulates air filters (HEPAs) 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 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-d cooled fuel from *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2007), 87 pCi/d intake of fission products is assigned to specific radionuclides as listed in Table 4-2.

Table 4-2. Annual intakes from the hot laboratory in the Chemistry Building.

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 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 [10]. It was replaced by a hot cell with improved ventilation and filtration in the Research Building in 1951, which in turn was removed from service and decontaminated in 1982. Releases from the hot cell are not documented but certainly would have been smaller than those estimated above for the hot laboratory. It is favorable to claimants to apply the Table 4-4 intakes for each year from 1943 through 1981 [11].

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 (AEC 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 decided upon, 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 (AEC 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/y). 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 C.F.R. pt. 20 2006).

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 released radioactive 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 (Voss 1974). By that time, uranium and thorium production and scrap recovery had ceased, and the Annex 1 Building had been demolished in 1953 (Karsjen 2003). Annex 2 operations had ceased in 1953, and the building was razed in 1972 (Ames 1985). The synchrotron operations had ceased in 1971, and the building was decommissioned in 1990 [12]. 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 shown 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.

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

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.

4.7 SUMMARY OF ENVIRONMENTAL EXTERNAL DOSES AND INTAKES

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

Table 4-6. Summary of 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
^{41}Ar from ALRR	1965–1977	1	Constant
All other R&D Buildings (see list in Section 4.4)	All	Negligible	Not applicable

a. The energy range for all environmental external dose is assumed to be 100% 30-250 keV (NIOSH 2006b).

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

Dates	Radionuclide/absorption	Intake (pCi/d)	Distribution
July 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 5-7.

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 2006a) for the period from January 1, 1942, through December 31, 1954, it was determined that internal doses from the production and casting of thorium metal cannot be reconstructed. Therefore, only doses from thorium contamination left over after 1954 are addressed here.

5.1 URANIUM EXPOSURE

Few data on uranium exposure were found (Ferretti, Price, and Schwartz 1951); however, the SEC petition evaluation report determined that doses could be estimated (NIOSH 2006a). 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 was unavailable, this analysis used data from two recent documents: Battelle-TBD-6000, *Site Profiles for Atomic Weapons Employers (AWE) that Worked Uranium and Thorium Metals*, and Battelle-TBD-6001, *Site Profiles for Atomic Weapons Employers that Refined Uranium and Thorium* (Battelle 2006a and 2006b) to estimate doses to workers in the Chemistry Building, Physical

Chemistry Annex 1, and Annex 2 buildings. These two documents are considered representative of the potential intakes that workers at Ames Laboratory might have had because the processes were often developed at Ames Laboratory and still remained similar to the processes used at the AWE sites. The two processes that were used here to estimate doses are Reduction to Metal from Battelle-TBD-6001 and Uranium Scrap Recovery and Casting from Battelle-TBD-6000.

5.1.1 Estimating Internal Doses from Uranium Metal

Historical aspects of the production of uranium metal at Ames Laboratory are discussed in Section 2.0 of this document. Although there was much historical information on the general methods for producing uranium metal (Spedding 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 abstracted from Battelle (2006a) and Battelle (2006b) as methods for assigning internal dose to workers.

The next two sections describe estimated doses for workers from January 1, 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 the Physical Chemistry Annex 1 (also known as Little Ankeny) can be assigned doses from inhalation of uranium [13]. Similarly, individuals who worked in the Chemistry Building during this period and individuals who worked in Physical Chemistry Annex 2 from 1944 through December 1945 should be assigned doses from inhalation of uranium [14]. Job titles of researchers acknowledged for their work in uranium production (found in Spedding 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 the production was being performed.

Because it is not clear if there were clerical, janitorial, 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 was derived from data in Battelle (2006a) and Battelle (2006b). In Battelle (2006b), there is a description of the reduction to metal that is similar to the process used at Ames Laboratory for production of uranium metal (Spedding 1947). The primary difference appears to be that at Ames Laboratory the process used granulated calcium metal and at the AWE sites the process used magnesium. Spedding (1947) also describes the process using magnesium, and it appears to be similar enough to be representative of the intakes at Ames Laboratory.

Battelle (2006b) provides data for multiple stages of the production operation and four categories of workers. 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 was taken from Table 8.29 in Battelle (2006b) and is for the Metal Reduction Bomb Preparation for an operator that works a 48-hr

week. This value was the highest intake rate for the metal reduction process. This number is then scaled for the potential for intake (see Table 5-1).

Battelle (2006a) provides data for a process of scrap recovery that is similar to the scrap recovery described in Spedding (1947). Data from Battelle (2006a, Table 7.8) are used in Table 5-1 for Ames Laboratory workers who worked in the Physical Chemistry Annex 2 building. The value used from Table 7.8 in Battelle 2006a represents the most conservative intake for the scrap recovery operations, similar to those performed in Physical Chemistry Annex 2.

For workers involved in research only from January through July 1942 in the Chemistry Building, an exposure of one-tenth of that of the workers involved in the production operations was assumed, which corresponds to smaller quantities of uranium [15]. There are very few details on the research activities that occurred in the Chemistry Building after July 1942; therefore, it was assumed that the one-tenth fraction applied to the researchers in the Chemistry Building through 1953.

Individuals supervising the production processes were assumed to be exposed for one-fourth of the time of the production staff [16].

For workers not directly associated with uranium metal research or production, an exposure of one-tenth of that of the supervisors was assumed (see Table 5-1) [17].

Table 5-1. Uranium intakes from inhalation (pCi/d).

Period	Chemistry Building	Physical Chemistry Annex 1	Physical Chemistry Annex 2
January 1942–June 1942	85[a,b]	Not applicable	Not applicable
July 1942–August 1945	8.5[a,b]	853[b]	Not applicable
January 1944 – December 1945	See above	See above	6,061[b]
September 1945–December 1950	8.5[a,b]	Not applicable	6,061
January 1951-December 1953	8.5[a,b]	Not applicable	5,556[b]

a. No data was available for determination of intakes in the Chemistry Building, therefore it was assumed that research activities would have one-tenth of the intake of that of production activities, and the research activities on uranium after uranium metal production ended would be one-tenth the intakes of the original research.
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.

The intakes in Table 5-1 were compared to the few actual bioassay results found for workers in approximately the same time period. Chapter 7 in Stone (1951), titled "Uranium Excretion Studies," provided data from a series of uranium bioassay obtained from Ames workers in 1944 and 1945. Of special interest were a series of samples taken from the supposedly highest exposed worker at Ames and samples taken from the most highly exposed group of workers at Ames (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 at Ames and assuming absorption type M uranium (the document indicated that the person was exposed to UF_4). The estimated intake was 1200 µg/d or 820 pCi/d assuming natural uranium. This is consistent with the intake estimate in Table 5-1 for Annex 1 and quite a bit lower than for Annex 2. The average bioassay result for the group of highest exposed workers was 75 µg/L. Assuming chronic intake for 1 yr prior to the bioassay, the estimated intakes were

- Absorption type F: 390 µg/d, 260 pCi/d
- Absorption type M: 1670 µg/d, 1100 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 Table 5-1. The type S intake estimates are greater than those in Table 5-1 but it is unlikely that anyone was actually exposed to just type S for long periods. UF_4 and metal were the principal forms of uranium in the production facilities.

Another small set of urinalysis data was found involving 5 or 6 samples from 6 workers during 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 yr 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 Table 5-1 are favorable to claimants [18].

5.1.1.2 Uranium Ingestion

As discussed in Battelle (2006b), ingestion intakes are determined using NIOSH (2004) and are shown in Table 5-2. 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) assumes chronic intakes, even during weekends, and that the number of hours worked in a year changed over time (Strom 2006). 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 and A is the median air concentration using the specified units, and h is the number of hours in a working 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 previously defined. This intake applies to the period from 1942 through August 1945. The same rationale used in Table 5-1 for reduced fractions for other workers is applied in Table 5-2.

Table 5-2. Uranium intakes from ingestion (pCi/d).

Period	Chemistry Building	Physical Chemistry Annex 1	Physical Chemistry Annex 2
January 1942–June 1942	0.87[a b]	Not applicable	Not applicable
July 1942–August 1945	0.08[a b]	8.7[b]	Not applicable
January 1944 – December 1945	See above	See above	124 [b]
September 1945–December 1950	0.08[a b]	Not applicable	124
January 1951-December 1953	0.08[a b]	Not applicable	114 [b]

a. No data was available for determining intakes in the Chemistry Building; therefore, it was assumed that research activities would have one-tenth of the intake of that of production activities, and the research activities on uranium after uranium metal production ended would be one-tenth the intakes of the original research.
b. 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.

5.1.1.3 Resuspension During Periods with No Uranium Operations

As described in Battelle (2006b), there was a potential for internal exposure to resuspended material from the 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 metalworking operations. The estimate used here assumes the maximum intake from inhalation for the researchers in uranium metal production (Chemistry Building), the metal production operations (Annex 1), and scrap recovery (Annex 2). The method used in Battelle (2006b) was applied to the maximum intake described for each process. Table 5-3 shows the intake in picocuries per day for each building for the specified periods. The annual inhalation intake from resuspension of deposited material for a worker in the Chemistry Building, Physical Chemistry Annex 1, and Annex 2 buildings, assuming the worst-case air concentrations for a 1-yr working operation, is shown in Table 5-3. These intakes apply to anyone in the building regardless of job description. For ingestion, NIOSH (2004) indicates that the ingestion rate, in terms of disintegrations per minute for an 8-hr 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 5-3.

Table 5-3. 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	Not applicable	17.5/1.6	Not applicable
January 1954–May 1976	1.8/0.16	Not applicable	124.7/11.2 (only through 1972), then 0[a]

a. The Physical Chemistry Annex 2 building 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 d, 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 [19]. F, M, and S are defined as fast, medium, and slow, respectively, in the figures.

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 2006a) and, therefore, internal dose is not addressed here for those radionuclides during those years. Research was conducted using small quantities of plutonium (micrograms) in the 1940s, and it involved methods of extraction of plutonium from up to 5 Ci of fission products in irradiated fuel, which would not have contained much plutonium (Spedding 1947). There is no indication of research that involved sufficient plutonium such that exposure to plutonium after 1954 would have been significant.

However, one building, Wilhelm Hall at Ames Laboratory, was the location of the main activities with thorium and is still in use. In March of 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 a document titled 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 over from the early production years. This information is described in Hokel et al. (1998). The following discussion using data from both Hokel 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 are provided in Table 5-4.

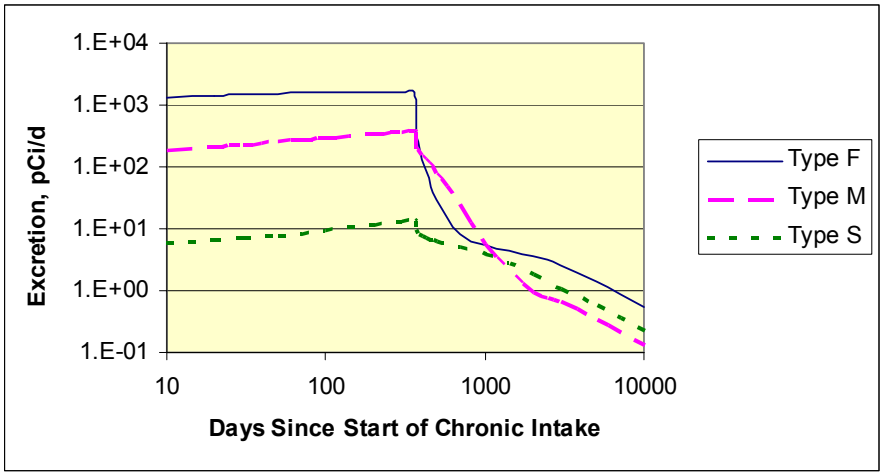


Figure 5-1. Urinary excretion of uranium expected from 365-d chronic intake at 6,061 pCi/d inhalation and 124 pCi/d ingestion.

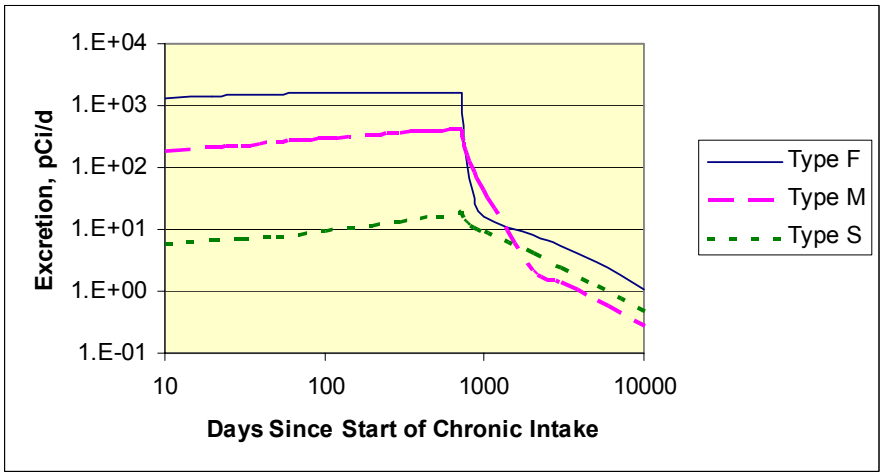


Figure 5-2. Urinary excretion of uranium expected from 730-d chronic intake at 6,061 pCi/d inhalation and 124 pCi/d ingestion.

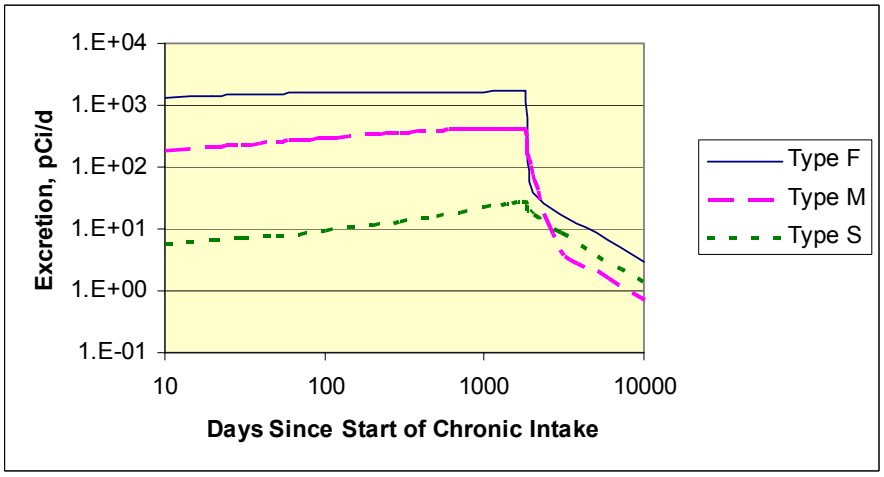


Figure 5-3. Urinary excretion of uranium expected from 1,826-d chronic intake at 6,061 pCi/d inhalation and 124 pCi/d ingestion.

Table 5-4. 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	0.5	448	1330	40.4
1956	1.5	392	1160	35.3
1957	2.5	343	1020	30.9
1958	3.5	300	887	27.0
1959	4.5	262	776	23.6
1960	5.5	229	679	20.6
1961	6.5	200	594	18.1
1962	7.5	175	519	15.8
1963	8.5	153	454	13.8
1964	9.5	134	397	12.1
1965	10.5	117	347	10.6
1966	11.5	103	304	9.24
1967	12.5	89.7	266	8.08
1968	13.5	78.4	232	7.07
1969	14.5	68.6	203	6.18
1970	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
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

Year	t (yr)	Air concentration (dpm/m$^{3)}$	Chronic inhalation intake (pCi/d)	Chronic ingestion intake (pCi/d)
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

All of the data in Hokel was reviewed and considered. Much of the data was related to locations that were hard to access and considered not to be of an issue for inhalation. 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). For this reason, the recommendation for assessing dose from thorium from 1954 through the present was provided. The accessible areas of the building, including the rooms, air ducts, hallways, stairwells, transformer rooms, etc., were surveyed starting in 1984. The data in Hokel et al. (1998) present an overview of the survey results. In 1996, some measurements were made using an Alpha Continuous Air Monitor in the subbasement 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 considered 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.

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 [20]. This means that the ^{232}Th activity was approximately 200 dpm/100 cm^2. Applying a resuspension factor of 10^{-4} [21]:

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

The value of 2 dpm/m^3 on 11/15/1995, 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.

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 was used to represent air concentrations in 1955 with some 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 room 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 used from the Hokel document 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 1/1/1955 to 11/15/1995 for a total of 14,929 days,

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

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

$$\text{intake (pCi/d)} = (\text{air conc. dpm/m}^3)(1 \text{ pCi}/2.22 \text{ dpm})(1.2 \text{ m}^3/\text{hr})(2000 \text{ hr/yr})/365\text{d/yr})$$

The estimates in Table 5.4 for ^{232}Th assumes an occupancy factor of 1 (100% for a working year). This is considered 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 [22]. 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 also shown in Table 5-4.

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 [23]. 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 5.4 [24].

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 the reactor operations. Because of 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.

A document titled *Program for Monitoring Personnel for Tritium, Sources of Tritium at the Ames Laboratory* describes the potential sources of tritium exposure to Laboratory personnel (Voss 1971). 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 (Voss 1971). Other possible sources of tritium were spent targets from the neutron generator, tritium gas in cylinders, and tritiated compounds that would have only been present occasionally (Voss 1971).

The Voss (1971) document is summarized here to help understand the 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 (3) who were involved in operations in which the 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 is 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 is required.

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

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

Assumptions described in the document (Voss 1971) used 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 considered to be 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. Body water was then 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 d. If the half-life could not be determined from the data, a value of 12 d was used. The data points that showed a half-life greater than 15 d were treated as additional separate exposures.

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

Other considerations for assignment of dose that were discussed in the document (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 that 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 all similar to the information provided above. A document titled *Personnel Monitoring Program* provides the Tritium Dose Calculation Procedures used at the time (Voss 1974).

Personnel on 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 and 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 will be on the record.

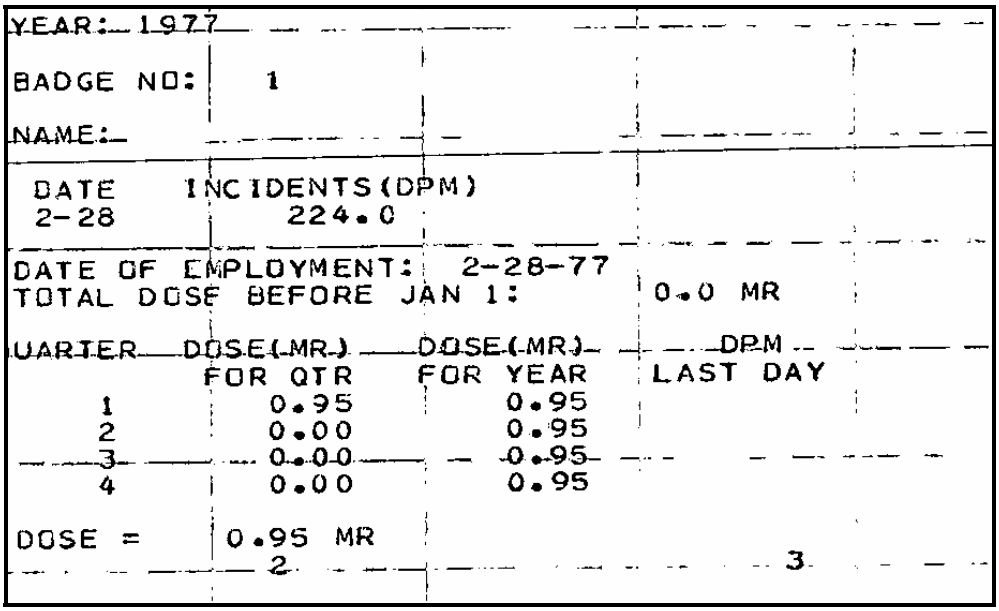
8. The four quarters for the year will be shown. The example in Figure 5-4 is for the first quarter and shows dose for quarter 1. For each subsequent quarter, the dose would be added and the total for the year calculated. It can be assumed that a "0.00" means the worker was not monitored, and not necessarily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

Documents from the early 1970s (Voss 1971; Ames 1974) indicate that a person on 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 to MDAs at other sites during this period and is consistent with the way the data were recorded [25].

5.3.2 Tritium Dose to Workers, 1982 to Present

An interoffice memorandum (Struss 1987) dated May 8, 1987, addresses radiation dose at the Applied Science Center building (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-hr workweek (Struss 1987).



YEAR: 1977

BADGE NO: 1

NAME:

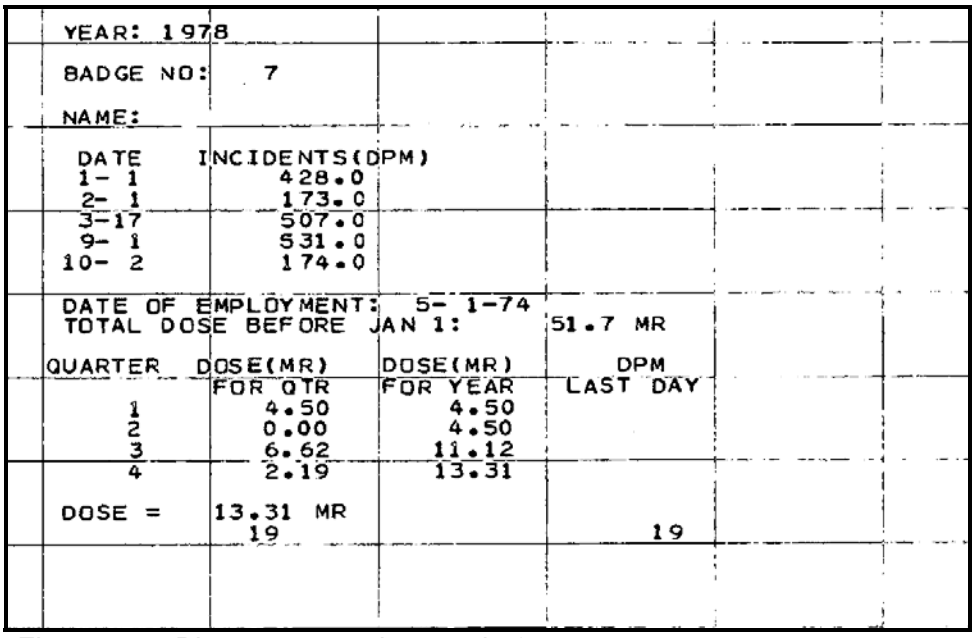
DATE	INCIDENTS(DPM)
2-28	224.0

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.

The dose reconstructor can consider assigning a dose of 8.6 mrem/yr for personnel working in the Applied Science Center building 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 to 1981

A coworker analysis of worker tritium bioassay was conducted on bioassay samples from 1965 through 1981 to provide default doses for unmonitored workers. Results of the coworker analysis are shown in Table 5-5, 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-

5, column 4, should be assigned. These doses apply to all organs. Assume a lognormal distribution with the geometric standard deviations (GSDs) as shown in Table 5-5, column 5.

Table 5-5. Results of tritium coworker analysis. [26]

	Coworker results		Assigned values	
Year	Annual dose (mrem)	GSD[a]	Annual dose (mrem)	GSD[a]
1965	0.44	1.6	0	N/A[b]
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 geometric standard deviation of 3.0 if the calculated value is less than 3 in Table 5-5
b. N/A – not applicable.

The tritium coworker doses were determined using methods described in *Analysis of Coworker Bioassay Data for Internal Dose Assignment* (ORAUT 2005b) and *Tritium Calculated and Missed Dose Estimates* (ORAUT 2004a). 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 considered insignificant and can be ignored when assigning doses; a minimum GSD of 3.0 was assigned if the calculated value was less than 3.

5.4 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" (Spedding 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* (NRC 1993).

The equation from NUREG-1400 is:

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

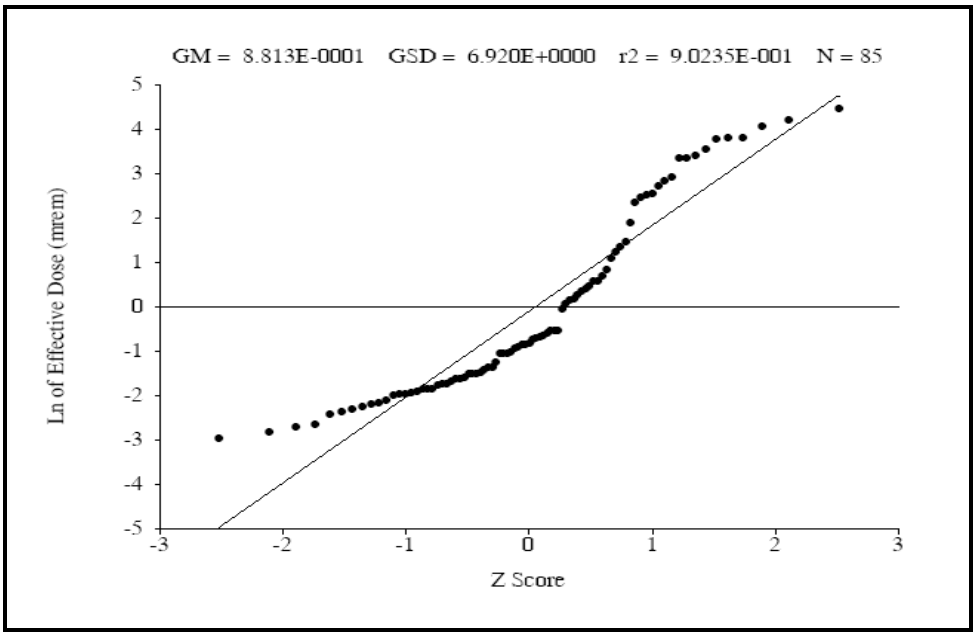


Figure 5-6. Distribution of worker doses from tritium in 1970.

where

I = intake
Q = source term for one 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

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

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

As described in Section 4.0, using the recommended fission product ratios for 180-d cooled fuel from *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2007), the 1,370 pCi/d intake of fission products is assigned to specific radionuclides as listed in Table 5-6.

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 [27]. 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, with complete separation of the source term from workers, in the Research Building in 1951. Intakes from the hot cell would have been negligible [28].

Table 5-6. Annual intakes from hot laboratory in the Chemistry Building.

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

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 one document, *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 also a negligible amount of fission products released during reactor operations; therefore, accounting for internal dose from the reactor operations is not necessary.

During D&D operations, radiation protection appeared to be acceptable for the time (Ames 1981) and external exposure and internal exposure for tritium was 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) (DOE 1977) based on the most conservative beta emitter (^{90}Sr).

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

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

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

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

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 5-7.

Table 5-7. Default intakes for Ames Laboratory (if no bioassay results are available) [29].

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[a]	Jan 1942–July 1942	Natural uranium	Chronic inhalation	F, M, or S	85	Constant	Dose
Researcher/Chemistry[a]	Jan 1942–July 1942	Natural uranium	Chronic ingestion	0.02 with F,M; 0.002 with S	0.87	Constant	Dose
All employees/Chemistry	Jun 1945–May 1976	Natural uranium	Chronic Inhalation	F, M, or S	1.8	Constant	Dose
All employees/Chemistry	Jun 1945–May 1976	Natural uranium	Chronic Ingestion	0.02 with F,M; 0.002 with S	0.16	Constant	Dose
Researcher in hot lab/Chemistry Building	1943–1951	Fission products	Chronic inhalation		See Table 5-6		
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 5-4	Constant	Dose
Anyone routinely in Wilhelm Hall	1955–present	Th-232	Chronic ingestion	5E-4 with M; 2E-4 with S	See Table 5-4	Constant	Dose
Anyone routinely in Wilhelm Hall	1955–present	Ra-228	Chronic inhalation	M	See Table 5-4	Constant	Dose
Anyone routinely in Wilhelm Hall	1955–present	Ra-228	Chronic ingestion	0.2	See Table 5-4	Constant	Dose
Anyone routinely in Wilhelm Hall	1955–present	Th-228	Chronic inhalation	M or S	See Table 5-4	Constant	Dose
Anyone routinely in Wilhelm Hall	1955–present	Th-228	Chronic ingestion	5E-4 with M; 2E-4 with S	See Table 5-4	Constant	Dose
Reactor workers and D&D workers	1965–1981	Tritium	Total of all modes	N/A[b]	N/A	Lognormal, see Table 5-5 for GSD	Dose; see Table 5-5
Workers involved with D&D of reactor, including former reactor workers	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	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).
b. N/A = not applicable.

6.0 OCCUPATIONAL EXTERNAL DOSE

Workers at Ames Laboratory received external radiation doses between 1942 and 1952 that were largely unmonitored [30]. Pocket chambers were available that might have been used to monitor external doses, but very few records could be found [31]. 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 [32]. 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 U.S. Department of Labor 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 was 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.

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 (Voss 1954). Dosimetry services evolved and improved over the following 50 yr; 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 individuals (Voss 1949). Thus, it is only possible to estimate any external doses before 1953. The

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	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, Brookhaven National Laboratory (BNL)	Apr 1954–June 1957	Biweekly	15	15	10	40	40	(d)
β/γ/NTA film, NCA	July 1957–June 1963	Biweekly	10	10	10	40	40	(d)
β/γ/NTA film, Atomic Film Badge Corporation	July 1963–Mar 1965	Monthly	10	10	10	40	40	(d)
Pocket Chambers, Ames Laboratory	Apr–June 1965	Daily		5			5	
β/γ/NTA film, Health Physics Services	July 1965–Oct 1979	Monthly	10	5	28	40	40	(d)
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. For years of NTA film use, between 1954 and 1979, the adjusted neutron dose is calculated using correction factor of 2.

maximum plausible external dose from uranium to Ames Laboratory workers during the period from 1942 through 1952 can be estimated using ORAUT (2005c).

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 (Voss 1954). Regular film badge service began in February 1953 for the Ames Laboratory staff. Beta/gamma film badges were exchanged every 2 wk (biweekly) and results were summarized monthly. However, between late September 1953 and early January 1954, there were three 4-wk and one 3-wk exchange periods (Voss 1954). The in-house film badge system again had variable exchange periods during 1954, 1957, 1958, 1959, and 1961. There was a 4-wk exchange period in January, a 3-wk period in February, and a 4-wk period in December 1954; otherwise, the exchange periods were 2 wk each (Voss 1955). The exchange frequency during 1955 and 1956 was every 2 wk, with no exceptions (Voss 1956, 1957b). There was a 4-wk exchange period in May and June 1957; all other exchange periods in 1957 were 2 wk (Voss 1958). There was a 3-wk exchange period in December 1958 and January 1959; all other exchanges periods in 1958 and 1959 were 2 wk (Voss 1959, 1960). The exchange frequency during 1960 was every 2 wk without exception (Voss 1961). During 1961, there were nine 3-wk exchange periods and 12 2-wk periods (Voss 1962).

Initially, the MRD was 10 mR, and no distinction was made between beta and gamma doses (recorded non-zero 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 (Voss 1957c, 1958, 1959, 1960, 1961, 1962).

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" (Voss 1954). 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 (Voss 1954). Between 1954 and 1961, neutron doses were added to the beta and gamma doses to give a total dose (Voss 1954, 1955, 1956, 1957b, 1957c, 1958, 1959, 1960, 1961, 1962).

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 1954, 1955, 1956, 1957b, 1957c). 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 (Voss 1958, 1959, 1960, 1961, 1962). 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 2006b). 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 (Voss 1955). The film badges were exchanged every 2 wk 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 (Voss 1955, 1962).

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 (Voss 1958). The film badges were exchanged every 2 wk 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 (Voss 1958).

All the data discussed above for the period from 1953 through 1961 was compiled on a spreadsheet and the inconsistencies were corrected or minimized (Martin 2006c). For example, when 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. In all cases when data were questionable, assumptions were made that would be favorable to claimants [33].

Dosimetry record keeping 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 (Voss 1963b). 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. 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 (Voss 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 (Voss 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 the reactor and synchrotron staff members and beta/gamma film badges for the remaining Ames Laboratory staff (Voss 1966). The MRDs were 10 mrad for beta, 5 mR for gamma, and 28 mrem for neutrons (Voss 1966).

TLD services for beta and gamma dosimetry were provided by Landauer from June 1980 through June 1995. Similar TLD 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 1981; Siemens 1996; ICN 1997). Dosimetry records compiled since 1981 are considered 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 film 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 multielement dosimeter; the neutron film was nuclear emulsion type A (NTA) (ORAUT 2006d). 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 [34]. From 1980 to the present, beta/gamma TLD service has been provided by Landauer, Siemens, ICN Dosimetry, 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 to 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.

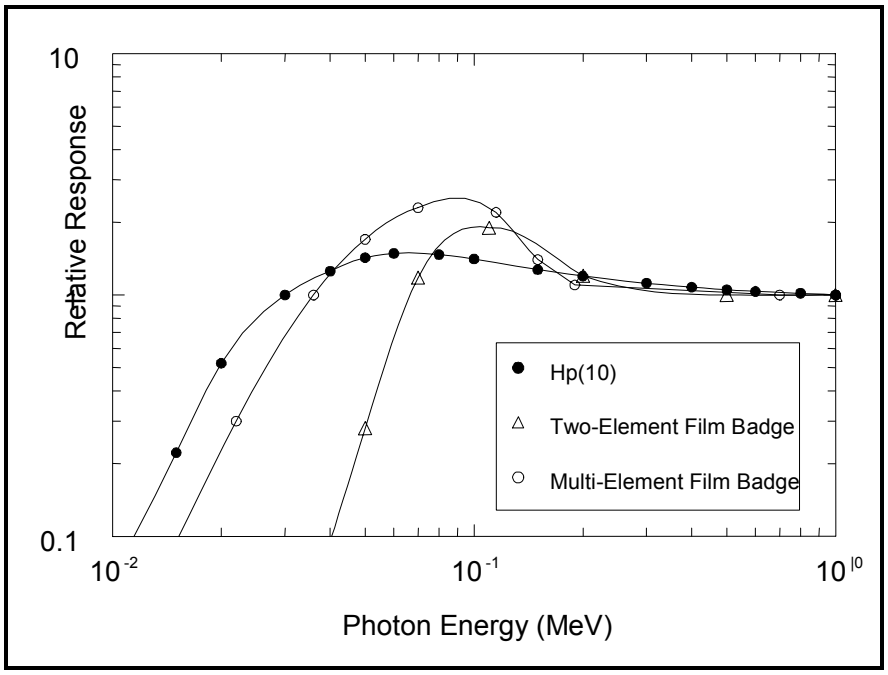


Figure 6-1. Comparison of *Hp(10)* for photons with energy responses for sensitive DuPont 502 emulsion in two-element film badge (Pardue, Goldstein, and Wollan 1944) and sensitive DuPont 555 emulsion in multi-element film badge (Thornton, Davis, and Gupton 1961).

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 2006d). 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 [35]. 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 (sic) (Voss 1955). 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 (Voss 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 (Voss 1963b). 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 (Voss 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 (Voss 1966, 1971, 1974, 1978, 1979, 1981). The calibration procedures were the same as those used during the period from 1953 to 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 1955, 1956, 1957b). 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 1955, 1956, 1957b).

From July 1957 to 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 (Voss 1958, 1959, 1960, 1961, 1962, 1963b, 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 hr 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 (NCA 1961). From June 1963 to 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 very few exceptions, the following sections show that, for external dose reconstruction purposes, 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 [36]. 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-d half-life) and ^{234m}Pa decay products (DOE 2000, 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 [37]. However, the activity of these sources was relatively small; the sources had a much smaller potential for beta radiation exposure to workers in comparison to 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 Ames Laboratory. Tritium exposures that occurred at the ALRR were monitored by means of urine bioassay, which is addressed in Section 5.0 of this site profile.

6.2.4.2 Photon Radiation

Brehmsstrahlung radiation doses from uranium metal ingots were calculated by Anderson and Hertel (2005). Contact photon dose rates of a few milliroentgen 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 the 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 [38]. 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 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 [39]. 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 non-zero 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 and film badges were provided by four commercial services from 1954 to 1981 [40]. 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 purposes [41].

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 that those experienced at other sites (e.g., the High Flux Beam Reactor at BNL) (ORAUT 2006e). The higher neutron doses at ALRR might have been due to less than adequate management of neutron shielding in the experimental areas [42]. 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 2006f). 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 2005d). 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 [43]. To allow for differences in the neutron spectra and other variables, a neutron dose correction factor of 2 is recommended for the Ames Laboratory synchrotron workers [44].

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 2005d). 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 was reanalyzed in ORAUT (2006e). 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 2006g). Because the MTR and the ALRR had very similar neutron spectra, a neutron dose correction factor of 2 is recommended for the 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 National Council on Radiological Protection and Measurements 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 (2006b). Table 6-2, which is from ORAUT (2006c), 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 MeV	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 foregoing information, which requires assessment of additional dose to be added to the measured photon dose from three primary causes as follows:

- 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* [45]. 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 considered to be 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 2005e).

The general approach to applying coworker data for cases with little or no individual external monitoring data is to assign either the 50th- or 95th-percentile doses 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 2005e).

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 2005e).

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 2005e). 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 6-3. The missed dose recommended for monitored workers in Table 6-4 was added to the 50th- and 95th-percentile values in Table 6-3. Specifically, one-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 2005e).

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 2006b) listed in Table 6-4 for the dosimetry systems. These missed doses are included in the values in Table 6-3.

Table 6-3. Assigned dose for unmonitored workers, mrem/year (McCartney 2006).

Year	Beta		Gamma		Neutron		Raw data files SRDB Ref ID No.
	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]	18821, 18822, 18833
1953	520	520	520	590[b]	650[c]	650[c]	18821, 18822, 18833
1954	520	520	520	737[b]	650	650	18834, 18835, 25816
1955	520	520	520	520[b]	650	650	18841
1956	520	520	520	520[b]	650	650	25169, 25171
1957	520	520	520	520[b]	650	650	25214, 25215, 25216
1958	520[a]	520[a]	520	520[b]	650	650	25459, 25460
1959	520[a]	520[a]	520	520[b]	650	650	25461, 25898
1960	520	520	520	520[b]	650	650	25898, 25907
1961	520[a]	520[a]	520	520[b]	650	650	25913
1962	520	520	520	520[d]	650	650[d]	25837
1963	240	240	240	265[e]	650	650	25837, 25914
1964	240	240	240	313[e]	650	650	25837, 25916
1965	240	240	240	408[e]	650	820[e]	25919
1966	240	306[e]	240	565[e]	806[f]	988[e]	25922
1967	240	466[e]	240	745[e]	932[f]	1,128[e]	25923
1968	240	307[e]	240	769[e]	876[f]	1,086[e]	25808
1969	240	395[e]	240	1,182[e]	1,030[f]	1,296[e]	25809
1970	240	240	240	671[e]	680[f]	932[e]	25810
1971	240	240	240	809[e]	722[f]	950[e]	25811
1972	240	240	240	512[e]	652[f]	988[e]	25834
1973	240	240	240	701[e]	650	932[e]	25839
1974	240	240	240	721[e]	650	876[e]	25859
1975	240	240	240	1,298[e]	650	870[e]	25860
1976	240	240	240	497[e]	650	958[e]	25861
1977	240	240	240	752[e]	650	904[e]	26613
1978	240	240	240	722[e]	650	652[e]	26615
1979	240	240	240	819[e]	650	650	26619
1980	240	240	240	1,595[e]	N/A[c]	N/A[c]	26624
1981	240[a]	240[a]	240	240	N/A[c]	N/A[c]	26626
1982	60[a]	60[a]	85	555	N/A[c]	N/A[c]	26628
1983	380	565	105	134	N/A[c]	N/A[c]	26631
1984	60	60	60	85	N/A[c]	N/A[c]	26632
1985	60	60	60	125	N/A[c]	N/A[c]	26642
1986	60	60	60	60	N/A[c]	N/A[c]	26645
1987	60	60	60	114	N/A[c]	N/A[c]	26647
1988	60	365	72	149	N/A[c]	N/A[c]	26653
1989	60	60	60	81	N/A[c]	N/A[c]	26657
1990	60	64	60	91	N/A[c]	N/A[c]	26659
1991	60	106	60	85	N/A[c]	N/A[c]	26664
1992	60	85	60	71	N/A[c]	N/A[c]	26666
1993	60	77	60	118	N/A[c]	N/A[c]	26668
1994	60	75	62	138	N/A[c]	N/A[c]	26672
1995	60	125	60	68	N/A[c]	N/A[c]	26679
1996	60	60	60	65	N/A[c]	N/A[c]	26681
1997	60	60	60	60	N/A[c]	N/A[c]	26683
1998	60	60	205	245	N/A[c]	N/A[c]	26685
1999	60	170	60	160	N/A[c]	N/A[c]	26750
2000	60	205	60	110	N/A[c]	N/A[c]	26799

Year	Beta		Gamma		Neutron		Raw data files SRDB Ref ID No.
	50th percentile	95th percentile	50th percentile	95th percentile	50th percentile	95th percentile	
2001	60[a]	60[a]	60	76	N/A[c]	N/A[c]	27038
2002	60	104	60	68	N/A[c]	N/A[c]	26808
2003	60	205	201	394	N/A[c]	N/A[c]	26809
2004	60[a]	60[a]	63	118	N/A[c]	N/A[c]	26810
2005	60	259	60	123	N/A[c]	N/A[c]	26811

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 2006b). 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 6-4. Potential missed dose for monitored workers.

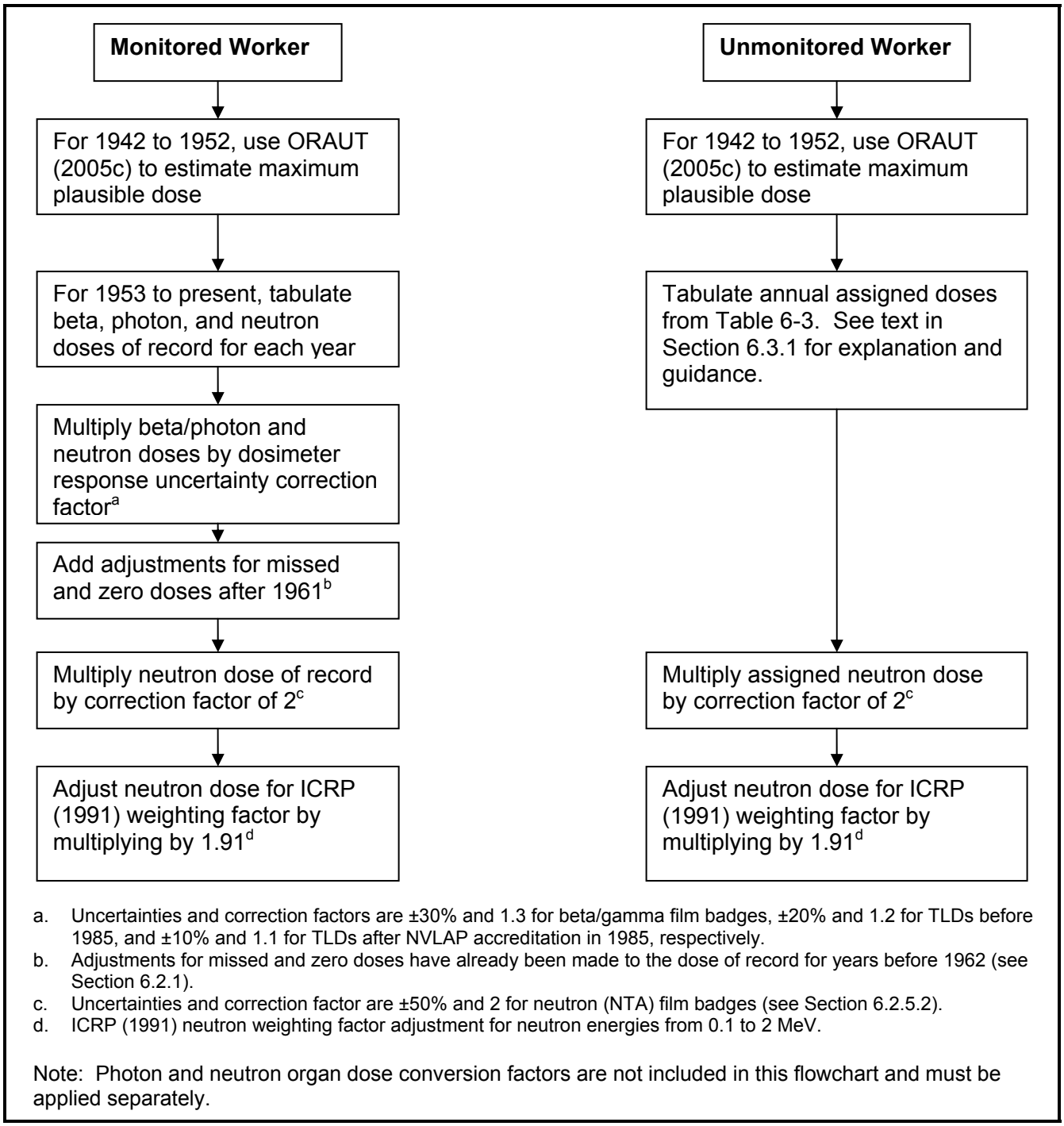
Dosimeter	Period	Exchange frequency[a]	MDL (mrem)			Missed annual mean dose (mrem)		
			Skin	Deep	Neutron	Skin	Deep	Neutron
Film badge-β,γ,n	1952–1956	Biweekly	40[b]	40[b]	50[b]	0[c]	0[c]	0[c]
	1957–1962	Biweekly	40[b]	40[b]	50[b]			
Film badge-β,γ,n	1963–1981	Monthly	40	40	50	240	240	300
TLD-β and γ	1981–present	Quarterly	30	30		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, 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 2006b). Therefore, no additional correction is required for monitored workers for the period from 1952 through 1961.

A flow chart is provided in Figure 6-2 to guide dose reconstructors in the application of all the foregoing tables and recommendations. The *Monitored Worker* side of the figure should be used if there are dose of 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 of 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 and several MeV (Morgan 1961). The estimated standard error in recorded film badge doses from photons of any energy is ±30% (ORAUT 2004b). The estimated uncertainty in doses recorded by TLDs is ±20% from 1982 through 1984 (ORAUT 2003) and ±10% since 1985 with NVLAP accreditation (Landauer 1985).



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. Uncertainties and correction factor are ±50% and 2 for neutron (NTA) film badges (see Section 6.2.5.2).
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

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). In cases where no nonpenetrating dose was recorded, the skin dose is assumed to

be equal to the whole-body penetrating dose. Additional guidance on determining skin dose can be obtained from *Interpretation of Dosimetry Data for Assignment of Shallow Dose* (ORAUT 2005f).

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 2006). 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.

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.

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 2006b). 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. Multiply the calculated neutron doses by the conversion factors from ambient dose equivalent to organ dose for AP irradiation from NIOSH (2006b, Appendix B). For photons applicable to Ames Laboratory (i.e., film badge era), use the conversion factor from exposure to organ dose.

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.
X-ray films and logbooks were examined to determine the frequency of routine chest X-rays received by early Ames Laboratory workers. No films or records prior to 1946 could be found. Numerous records for 1946 and later established the frequencies given in Table 3-1.

[2] 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 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 2006e), which found that the environmental dose from accelerator effluents was less than 1 mrem/yr and negligible.

[3] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
Common practice in accelerator health physics is to conduct a periodic fence line survey of maximum radiation dose rates. The purpose of such surveys is to determine the maximum off-site impact of accelerator operations and identify any special controls or limitations that may be needed for safe operation. In order to produce maximum fence line 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, thus the maximum dose rate conditions only existed long enough to obtain the fence line survey results. Normally, the accelerator was operated at reduced beam currents and with targets that produced much lower radiation dose rates both 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.

SRDB Ref ID 25766 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 fence line 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 fence line dose rates reported in Table 4-1.

[4] 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 various non-operational activities, such as experiment set-up, equipment start-up, maintenance, repairs, etc. Thus, about two-thirds of the total time is available for operations.

[5] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The radiation conditions during the fence line 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 locations would give an annual dose of greater than 2,000 mrem. A review of dosimetry records for Synchrotron staff 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.

[6] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The buildings of Ames Laboratory are located 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.

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

[8] 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.

[9] 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 of the default values in ICRP 68 were considered possible.

[10] 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.

[11] 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 lab" 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 considered unlikely that the controls of the hot lab were as good as those for the hot cell.

[12] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The decommissioning date of 1990 was determined in a personal communications with Michael J. McGuigan, Radiation Safety Officer, at Ames Laboratory, July 2006.

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

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

[15] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Because there was no information regarding the amount of time in any one 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 2006a and 2006b.

[16] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Because there was no information regarding the amount of time in any one 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 2006a and 2006b.

[17] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Because there was no information regarding the amount of time in any one 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 2006a and 2006b.

[18] Bihl, Donald E. ORAU Team. Principal Health Physicist. April 2007.
The calculations in the preceding 3 paragraphs were conducted by Mr. Bihl using IMBA from data presented 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 that the detection level was approximately 0.04 mg/L.

[19] 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 energy employee's 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 purposes.

[20] 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 of much help. Of the surface survey results in areas somewhat accessible to people, the highest result was 1224 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 clearly represent an upper bound.

[21] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
Indoors resuspension factors range from 10^{-6} to 10^{-3}. The IAEA ("Monitoring of Radioactive Contamination on Surfaces," Technical Reports Series No. 120, 1970) recommends an indoor resuspension factor of 5×10^{-5}. 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.

[22] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
Hokel et al. (1998) indicated that 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 have 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.

[23] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Review of Hokel et al. (1998) suggested that there were areas with higher contamination, such as crawl spaces 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 1998 on this subject.

[24] Hickey, Eva Eckert, ORAU Team. Principal Health Physicist. February 2007.
In various places 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.

[25] 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 Hanford, Savannah River, and Los Alamos, to 0.02 µCi/L at Oak Ridge and Livermore National Laboratories, to 0.0002 µCi/L at the Idaho National Engineering Lab. The Hanford and Savannah River values were recording levels with actual laboratory MDAs generally being lower. Review of the way data were recorded at Ames indicated that 0.1 µCi/L was the likely MDA or recording level.

[26] 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.

[27] Bihl, Donald E. ORAU Team. Principal Health Physicist. February 2007.
Strontium is type F except for the titanate form, which is rare. There is no reason to suspect Sr titanate was in use at Ames.

[28] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Hot cells are designed to be inherently safe with no material being released. Since no information was found that indicated that there was ever a problem or accident with the hot cell, it can be assumed that it functioned as expected.

[29] Hickey, Eva Eckert. ORAU Team. Principal Health Physicist. February 2007.
Table 5-7 is a summary of the intakes described in this Section. The data is taken from (1) Table 5-1 and the discussion in Section 5.1.1.1 on uranium inhalation; (2) Table 5-2 and the discussion in Section 5.1.1.2 on uranium ingestion; (3) Table 5-3 and the discussion in Section 5.1.1.3 on resuspension of uranium during periods of no operation; (4) the discussion in Section 5.2 on thorium exposure from left over thorium contamination; (5) doses from tritium as discussed in Section 5.3; and (6) Table 5-6 doses from fission product intakes as discussed in section 5.4.

[30] 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 Trybout (1944) and Voss (1949), which are very incomplete records. Either the radiation doses were not measured or essentially no records were kept. In either case, the workers during this period were unmonitored.

[31] 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.

[32] 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.

[33] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
As the spreadsheets were assembled, some adding 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.

[34] 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.

[35] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
As stated in Section 6.2.1 Dosimetry, record keeping 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 (Voss 1963b). 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.

[36] 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.

[37] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The early history of research at Ames Laboratory was described by Spedding (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.

[38] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The descriptions of research and production operations at Ames Laboratory (Spedding 1947, Ames 1985) established the presence of photons in the less than 30 keV, 30-250 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.

[39] 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.

[40] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
Commercial film badge service was supplied by Brookhaven National Laboratory 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).

[41] 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.

[42] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The neutron doses received by ALRR staff were somewhat higher when compared to similar staff at the High Flux Beam Reactor at Brookhaven (ORAUT 2006d). 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 may have been less than adequate.

[43] 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 neutron spectrum at SLAC included higher energy neutrons, thus the neutron radiation fields were not identical.

[44] Martin, Jerome B. ORAU Team. Senior Health Physicist. February 2007.
The SLAC Site Profile (ORAUT 2006f) 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 Laboratory synchrotron.

[45] 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.

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GLOSSARY

absorbed dose
Amount of energy in rads or grays deposited in a substance by ionizing radiation per unit mass of the substance. See *dose*.

absorption type
Categories for materials according to their speed of absorption in the body, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F for fast absorption (formerly inhalation class D), M for moderate absorption (formerly inhalation class W), and S for slow absorption (formerly inhalation class Y). 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. Alpha particles have low penetrating power and a short range (a few centimeters in air). Outside the body, the most energetic alpha particle generally fails to penetrate the dead layers of cells covering the skin or a sheet of paper. Alpha particles emitted by radionuclides inside the body are a more significant health risk.

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. Most of the direct fission products are (negative) beta emitters. Exposure to large amounts of beta radiation from external sources can cause skin burns (erythema), and beta emitters can be harmful inside the body. Thin sheets of metal or plastic can stop beta particles.

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*.

chronic exposure

Radiation dose to the body delivered in small amounts over a long period (e.g., days or years). For the purposes of input to the Interactive RadioEpidemiological Program, chronic exposure is selected under exposure rate for beta and neutron dose. See *acute exposure*.

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.

depleted uranium (DU)

Uranium with a percentage of ^{235}U lower than the 0.7% found in natural uranium. As examples, spent (used) fuel elements, byproduct tails, residues from uranium isotope separation, and some weapons materials contain DU. DU can be blended with highly enriched uranium to make reactor fuel or used as a raw material to produce plutonium.

dose

In general, the effects of ionizing radiation in terms of the specific amount of energy 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. Various terms narrow the type of dose, and some are additive:

- Absorbed dose is the amount of energy deposited in a substance by ionizing radiation.
- Collective dose is the sum of the doses to a specific population.
- Committed dose is the dose over time (usually 50 years for workers) to a specific organ or tissue from an intake of radioactive material.
- Cumulative dose is the sum of all doses to the same portion of the body or to the whole body over time.
- Deep dose is the dose at a 1-centimeter depth in tissue (1,000 milligrams per square centimeter).
- Effective dose is the sum of the equivalent doses in the principal tissues and organs of the body, each weighted by a tissue weighting factor that accounts for the probabilities of fatal and nonfatal cancers according to severity and the average length of life lost due to an induced cancer. It indicates the biological effect of the radiation exposure in that tissue.
- Equivalent dose or dose equivalent is the absorbed dose in a tissue or organ multiplied by a weighting factor for the particular type of radiation.
- Organ dose is the dose to a specific organ.
- Penetrating dose is that from higher energy photon (gamma and X-ray) radiation and neutron radiation that penetrates the outer layers of the skin. Nonpenetrating dose is that from beta and lower energy photon radiation.
- Personal dose equivalent is the dose equivalent in soft tissue below a specified point on the body at a specified depth.

- Shallow dose is the dose at a 0.07-millimeter depth in tissue (7 milligrams per square centimeter).
- Skin dose is the dose to the skin.
- Whole-body dose is the dose to the entire body excluding the contents of the gastrointestinal tract, urinary bladder, and gall bladder.

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*.

depleted uranium (DU)
Uranium with a percentage of ^{235}U lower than the 0.7% found in natural uranium. As examples, spent (used) fuel elements, byproduct tails, residues from uranium isotope separation, and some weapons materials contain DU. DU can be blended with highly enriched uranium to make reactor fuel or used as a raw material to produce plutonium.

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 (EU)
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*. In this document, exposure does not refer to the radiological physics concept of charge liberated per unit mass of air.

film
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 rays are very penetrating, but dense materials such as lead or uranium or thick structures can stop them. 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.

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.

intake

Radioactive material taken into the body by inhalation, absorption through the skin, injection, ingestion, or through wounds. Depending on the radionuclide involved, intakes are in units of mass, activity, or potential alpha energy.

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.

in vitro

Of or relating to a process that takes place under artificial conditions or outside a living organism (e.g., in the laboratory). From Latin meaning *in glass*.

in vivo

Of or relating to a process that takes place in a living organism. From Latin meaning *in life*.

minimum detectable activity or amount (MDA)
Lowest amount of radioactive activity or substance amount detectable by a specific instrument or process. Smallest amount or activity of a radionuclide in a sample or organ that yields a result above the detection level with a specific probability of a Type II (false negative) error while accepting an specific probability of a Type I (false positive) 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). Also called *minimum detectable limit* and *minimum detection limit* or *level*.

missed 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.

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.

occupational exposure
Exposure to radiation and/or to radioactive material from sources of radiation, whether in the possession of the DOE site contractor or other person, 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 exposure

Exposure to radiation or radioactive materials from medical diagnostic procedures during physical examinations that are required as a condition of employment. For dose reconstruction, occupational medical exposure is a component of occupational exposure.

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

Basic unit of electromagnetic radiation. Photons are massless "packages" of light energy that range from low-energy microwave photons to high-energy gamma rays. Photons have energies between 10 and 100 kiloelectron-volts. See *photon radiation*.

photon radiation

Electromagnetic radiation of light energy (photons) from microwaves to gamma rays. Gamma rays and X-rays are examples of ionizing photon radiation, which have enough energy to penetrate matter, including the body, and deposit energy in that matter.

probability of causation (POC)

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

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) that travel from one point to another, some of which can pass through or partly through solid materials including the human body. See *ionizing radiation*.

radioactivity

Disintegration of certain elements (e.g., radium, actinium, uranium, and thorium) accompanied by the emission of alpha, beta, gamma, and/or neutron radiation from unstable nuclei.

radiograph

Photographic image produced on film by gamma rays or X-rays. Some of the rays (photons) can pass through parts of an item, while more opaque parts partially or completely absorb them and thus cast a shadow on the film.

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.

sievert (Sv)

International System unit for dose equivalent, which indicates the biological damage caused by radiation. The unit is the radiation value in gray (equal to 1 joule per kilogram) multiplied by a weighting factor for the type of radiation and a weighting factor for the tissue; 1 Sv equals 100 rem.

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*.

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.

transuranic (TRU) elements

Elements with atomic numbers above 92 (uranium). Examples include plutonium and americium. All isotopes of the transuranic elements are radioactive, they are naturally either rare or nonexistent on Earth, and most are known only as a result of research using nuclear reactors and particle accelerators because of extremely short half-lives.

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 *depleted uranium*, *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.