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Site Profile for Battelle Memorial Institute, King Avenue and West Jefferson Sites, Columbus, Ohio

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
03/26/2010	00	New site profile for the Battelle Memorial Institute sites in Columbus, Ohio (King Avenue) and West Jefferson, Ohio (West Jefferson). Incorporates formal internal and NIOSH review comments. Training required: As determined by the Objective Manager. Initiated by Janet L. Westbrook and Vincent A. King.
06/27/2016	01	Revision initiated to include the Special Exposure Cohort class determination for the King Avenue facility due to inadequate internal radiological exposure monitoring (April 16, 1943, through June 30, 1956) and inadequate external radiological exposure monitoring (April 16, 1943, through February 13, 1951) and to include the class determination for the King Avenue facility due to inadequate internal monitoring of thorium exposures (July 1, 1956, through December 31, 1970). Revised Section 3.0 on occupational medical dose to incorporate updated Project information from ORAUT-OTIB-0006, ORAUT-OTIB-0079 and additional references. Modified Section 4.0 thorium information for internal dose section to include SEC classes, additional detail on chemically separated thorium, and ratios for recycled uranium radionuclides. Added information in Section 5.0 on occupational external dose section to address interpretation of dosimeter results expressed in “arbitrary units.” Added Section 5.8 to include assessment of ambient external radiation exposures in dose reconstructions. Added Section 6.0 to include assessment of internal environmental exposures in dose reconstructions. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Vincent A. King.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AGC	Alpha-Gamma Cell
ANL	Argonne National Laboratory
AP	anterior-posterior
APPR	Army Package Power Reactor
ATR	Advanced Test Reactor
AWE	atomic weapons employer
BCL	Battelle Columbus Laboratories (i.e., Battelle Memorial Institute)
BCLDP	Battelle Columbus Laboratories Decommissioning Project
BMI	Battelle Memorial Institute
BRR	Battelle Research Reactor
BWR	boiling-water reactor
BZ	breathing zone
CAA	Controlled Access Area (at JN-1)
CAL	Critical Assembly Laboratory (JN-2, 1957 to 1963)
CAM	continuous air monitor
CEMP	Columbus Environmental Management Project
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
Con Ed	Consolidated Edison
cpm	counts per minute
D&D	decontamination and decommissioning
d	day
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
DU	depleted uranium
EDTA	ethylenediaminetetraacetic acid
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ERDA	U.S. Energy Research and Development Administration
ETR	Engineering Testing Reactor
EU	enriched uranium
FP	fission product
ft	foot
g	gram
gal	gallon
GCRE	Gas-Cooled Reactor Experiment
GM	geometric mean
GSD	geometric standard deviation
HAPO	Hanford Atomic Power Operations
HEC	High Energy Cell (JN-1)
HEDL	Hanford Engineering Development Laboratory

HEPA	high-efficiency particulate air
HEU	highly enriched uranium
HP	health physicist
hr	hour
ICRP	International Commission on Radiological Protection
in.	inch
INL	Idaho National Laboratory
JN	prefix for some West Jefferson buildings (north part of the site)
JS	prefix for some West Jefferson buildings (south part of the site)
keV	kiloelectron-volt, 1,000 electron-volts
kg	kilogram
KPA	kinetic phosphorescence analysis
L	liter
LAT	lateral
lb	pound
LLC	Low Level Cell (JN-1)
LLNL	Lawrence Livermore National Laboratory
mCi	millicurie
MDA	minimum detectable amount
MDL	minimum detection level
MED	Manhattan Engineer District
MeV	megaelectron-volt, 1 million electron-volts
MFP	mixed fission products
mg	milligram
MGCR	Maritime Gas-Cooled Reactor
MHW	multihundred-watt (type of ^{238}Pu fuel for RTGs)
mi	mile
mm	millimeter
mrem	millirem
mL	milliliter
ML-1	Mobile Low-Power Reactor 1
MPBB	maximum permissible body burden
mph	miles per hour
MPC	maximum permissible concentration
MPL	maximum permissible limit
mR	milliroentgen
mSv	millisievert
MT	metric ton
MW	megawatt
NAC	Nuclear Assurance Corporation
NASA	National Aeronautics and Space Administration
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurement
NIOSH	National Institute for Occupational Safety and Health
NLO	National Lead Company of Ohio
NP	neutron-to-photon
NRC	U.S. Nuclear Regulatory Commission

NSEC	Nuclear Science and Engineering Corporation
NTA	nuclear track emulsion, type A
NU	natural uranium
NYOO	New York Operations Office
OMR	organic-moderated reactor
OMRE	Organic-Moderated Reactor Experiment, aka Organic Moderated Solvent Burning Experiment
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
PA	posterior-anterior
pCi	picocuries
PRTR	Plutonium Recycle Test Reactor
PyC	pyrolitic carbon
R	roentgen
R&D	research and development
RAL	Radioanalytical Laboratory
RCRA	Resource Conservation and Recovery Act of 1976
RTG	radioisotope thermoelectric generator
s	second
S&M	surveillance and maintenance
SID	source-to-image distance
SME	Salt-Moderated Experiment
SNAP	Systems for Auxiliary Nuclear Power
SNM	special nuclear material
SRDB Ref ID	Site Research Database Reference Identification (number)
SS	stainless steel (as a material specification); Super S (as a biosolubility type)
t	ton
TLD	thermoluminescent dosimeter
TRU	transuranic
U.S.C.	United States Code
UNH	uranyl nitrate hexahydrate
vol %	volume percent
W	watt
wt %	weight percent
wk	week
µCi	microcurie
µg	microgram
µm	micrometer
§	section or sections

1.0 INTRODUCTION

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historical background information and guidance to assist in the preparation of dose reconstructions at particular Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies). These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used to refer to an area, building, or group of buildings that served a specific purpose at a DOE or AWE facility. It does not mean, nor should it be equated to, an "AWE facility" or a "DOE facility." The terms AWE and DOE facility are defined in sections 7384l(5) and (12) of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA), respectively. A DOE facility is defined 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 [DOE] (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program);" and with regard to which the DOE has or had a proprietary interest, or "entered into a contract with an entity to provide management and operation, management and integration, environmental remediation services, construction, or maintenance services." 42 U.S.C. § 7384l(12). On the other hand, an AWE facility means "a facility, owned by an atomic weapons employer, that is or was used to process or produce, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining or milling." 42 U.S.C. § 7384l(5). The Department of Energy (DOE) determines whether a site meets the statutory definition of an AWE facility and the Department of Labor (DOL) determines if a site is a DOE facility and, if it is, designates it as such.

Accordingly, a Part B claim for benefits must be based on an energy employee's eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility's designated time period and location (i.e., covered employee). After DOL determines that a claim meets the eligibility requirements under EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and types of radiation exposure to be included in an individual dose reconstruction. Under EEOICPA, eligible employment at an AWE facility is categorized as employment either (1) during "a period when the employer was processing producing, for the use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining and milling," (i.e., the operational period); or (2) during a period that NIOSH has determined that "there is a potential for significant residual contamination outside of the period in which weapons-related production occurred," (i.e., the residual contamination period). 42 U.S.C. § 7384l(3).

Based on the abovementioned definition for eligible employment during an AWE facility's operational period, NIOSH includes radiation exposures incurred in the performance of duty, such as medical X-rays received as a condition of employment for participating in DOE projects, at an AWE facility in dose reconstructions. This may include radiation exposure related to the Naval Nuclear Propulsion Program and any radiation exposure received from the production of commercial radioactive products that were concurrently manufactured by the AWE facility during the operational period. In contrast, only two categories of radiation exposure as defined in 42 U.S.C. § 7384n(c)(4) should be included in dose reconstructions for claims involving employment during the residual contamination period. First, NIOSH must include exposures to radiological contaminants resulting from activities that had a nuclear-weapon nexus or conducted by or on behalf of the DOE (with an exclusion of activities related to, among other things, the Naval Nuclear Propulsion Program) that took place during the operational period. 42 U.S.C. § 7384n(c)(4)(A). Second, radiation doses from sources not included in the first

category but which cannot be distinguished through reliable documentation should also be included in dose reconstructions. 42 U.S.C. § 7384n(c)(4)(B). Furthermore, because all DOE-related activities have ceased during the residual contamination period, NIOSH does not include doses from medical X-rays performed during the residual contamination period (NIOSH 2007) in dose reconstructions.

Likewise, NIOSH does not consider the following types of exposure as those incurred in the performance of duty as a condition of employment for DOE-related activities at an AWE facility. Therefore these exposures are not included in dose reconstructions for either the operational or residual contamination period (NIOSH 2010):

- Background radiation, including radiation from naturally occurring radon present in conventional structures

- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

1.1 PURPOSE

The purpose of this document is to provide a site profile that contains technical basis information for the evaluation of occupational dose for EEOICPA claimants who were employed at the Battelle Memorial Institute (BMI) or Battelle Columbus Laboratories (BCL) King Avenue and West Jefferson sites in Columbus, Ohio. In this site summary document, BMI and BCL (the Columbus-based division of BMI operations) are referred to as Battelle. When it is necessary to distinguish between the two sites, they will be referred to as King Avenue and West Jefferson, respectively.

1.2 SCOPE

Section 1.3 describes the Special Exposure Cohort (SEC) class for the King Avenue site. Section 2.0 describes the King Avenue and West Jefferson sites and their operations. Section 3.0 discusses occupational medical dose, and Sections 4.0 and 5.0 detail known information about occupational internal and external dose, respectively. Section 6.0 addresses assessment of internal environmental exposures in dose reconstructions. Section 7.0 discusses exposure from residual contamination after the end of Battelle operations. Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 8.0.

Evaluation of drinking water and soil as potential sources of intake for workers was evaluated. Soil samples were collected to demonstrate compliance with environmental release requirements and results were reported in Battelle annual environmental reports (listed in the Environmental Reports section of the References list). Drinking water samples, although not required for environmental compliance monitoring, were collected beginning as early as May of 1966 (Kizer 1971). No results above regulatory limits were recorded for either soil or drinking water samples, and no potential pathway for worker exposure was identified.

Attachment A provides an overview chronology of significant events at Battelle, including the start and end of operations. Attachment B is a list of buildings, their major subareas and uses, and the dates of their use. Attachments C and D report the history of operations by building and material and the types and quantities of material, respectively. Attachment E lists the incidents that were reported at the sites. Attachment F provides the analysis of measured neutron and photon dose radiation survey data, and Attachment G addresses the estimation of onsite air concentrations.

1.3 SPECIAL EXPOSURE COHORT

The Secretary of the U.S. Department of Health and Human Services has designated the addition of the following class to the Special Exposure Cohort (SEC) for the Battelle King Avenue site (Sebelius 2013):

> *All Atomic Weapons Employees who worked at the King Avenue facility owned by Battelle Laboratories in Columbus, Ohio, during the period from April 16, 1943, through June 30, 1956, for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees included in the Special Exposure Cohort.*

Dose reconstruction guidance in this document for periods through June 30, 1956, is presented to provide a technical basis for partial dose reconstructions for claims not compensated under the SEC (i.e., nonpresumptive cancers and SEC employment less than 250 days). NIOSH found that it was not feasible to reconstruct internal doses for inadequately monitored radionuclides such as uranium, thorium, and their progeny through June 30, 1956, nor was it feasible to reconstruct external doses from beta, gamma, and neutron radiation through February 13, 1951. Although NIOSH found that it is not possible to completely reconstruct radiation doses for the proposed class, it intends to use any internal and external monitoring data that might become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures). Therefore, dose reconstructions for individuals employed at the Battelle King Avenue facility during the period from October 1, 1943, through June 30, 1956, but who do not qualify for inclusion in the SEC, can be performed using these data as appropriate (Sebelius 2013).

In addition, the Secretary of the U.S. Department of Health and Human Services has designated the addition of the following class to the Special Exposure Cohort (SEC) for the Battelle King Avenue site (Burwell 2016):

> *All Atomic Weapons Employees who worked at the facility owned by the Battelle Laboratories at the King Avenue site in Columbus, Ohio, during the period from July 1, 1956, through December 31, 1970, for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees included in the Special Exposure Cohort.".*

Dose reconstruction guidance in this document for the period from July 1, 1956, through December 31, 1970, is presented to provide a technical basis for partial dose reconstructions for claims not compensated under the SEC (i.e., nonpresumptive cancers and SEC employment less than 250 days). NIOSH found that, due to insufficient personnel and workplace monitoring and source term data, it lacked sufficient information necessary to complete individual dose reconstructions with sufficient accuracy for internal exposures to thorium and thorium progeny to which these workers may have been subjected during the period (Burwell 2016). Although NIOSH found that it is not feasible to completely reconstruct radiation doses for this proposed class, NIOSH intends to use any internal and external monitoring data that may become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures). Therefore, dose reconstructions for individuals employed at the Battelle Laboratories King Avenue facility during the period from July 1, 1956, through December 31, 1970, but who do not qualify for inclusion in the SEC, may be performed using these data as appropriate.

2.0 SITE DESCRIPTION

The DOE Facility List Database provides the following information for the King Avenue and West Jefferson sites (DOE 2014a, 2014b):

- *Battelle Laboratories - King Avenue*

  Also Known As: Battelle Columbus Laboratories (BCL); Battelle Memorial Institute (BMI)

  State: Ohio; Location: Columbus

  Time Period: AWE 1943-1986; BE 1943-1961; DOE 1986-2000 (remediation); Residual Radiation 2001 - March 1, 2011

  Facility Type: Atomic Weapons Employer/ Beryllium Vendor/ Department of Energy

  Facility Description: From 1943 to 1986, Battelle Memorial Institute performed atomic energy research and development as well as beryllium work for the Department of Energy and its predecessor agencies. The Battelle Laboratories have two separate locations in Columbus - King Avenue and West Jefferson. Battelle's research supported the government's fuel and target fabrication program, including fabrication of uranium and fuel elements, reactor development, submarine propulsion, fuel reprocessing, and the safe use of reactor vessels and piping.

  The following activities were performed at the King Avenue location: processing and machining enriched, natural, and depleted uranium and thorium; fabricating fuel elements; analyzing radiochemicals; and studying power metallurgy. Beryllium work was conducted from 1943 until at least 1961.

- *Battelle Laboratories - West Jefferson*

  Also Known As: Battelle Memorial Institute (BMI); Battelle Columbus Laboratories (BCL); West Jefferson Plutonium Facilities

  State: Ohio, Location: Columbus

  Time Period: AWE 1956-1975; Residual Radiation 1976-1985; DOE 1986-present (remediation)

  Facility Type: Atomic Weapons Employer/ Department of Energy

  Facility Description: From 1943 to 1986, Battelle Memorial Institute performed atomic energy research and development for the Department of Energy and its predecessor agencies. The Battelle Laboratories have two separate locations in Columbus - King Avenue and West Jefferson. Battelle participated in research on fabrication of uranium and fuel elements, reactor development, submarine propulsion, fuel reprocessing, and the safe use of reactor vessels and piping.

  At the West Jefferson location, Battelle operated a large hot cell facility and a research reactor. Reactor operations began in October 1956, and ended in December 1974. The reactor was defueled and partially dismantled in 1975 and Battelle's license was changed to possession-only status.

During the period of residual contamination, as designated by the National Institute for Occupational Safety and Health and as noted in the dates above, employees of subsequent owners and operators of this facility are also covered under the Energy Employees Occupational Illness Compensation Program Act.

Battelle was a research institution and undertook a wide variety of radiologically significant projects as part of its work for the U.S. Department of Energy and its predecessors[1] along with other defense and commercial work. These projects ranged from low-level biological and industrial radiotracer work to high-level work involving alpha gloveboxes and hot cells.

The relative locations of the King Avenue and West Jefferson sites are shown in Figure 2-1 (BCL 1974). Individual plan views of the King Avenue and West Jefferson sites are shown in Figures 2-2 and 2-3, respectively (Jenson 2003). Plan views are also available for the Melting Facility at Building 3 (Kirsch 2000a); the Building 4 Radioisotope Laboratory and various subareas (Sunderman and Dickerson 1962; Brown 1966); the Hot Cell Laboratory (JN-1) and subareas (Peters 2001; Sunderman and Dickerson 1962; Dickerson 1956; Brown 1966); the Critical Assembly Laboratory (JN-2) (Hogan et al. 1958; Jankowski and Chastain 1958; Kirsch 1987); the Battelle Research Reactor (BRR, JN-3) (Chastain et al. 1955; Chastain 1957; Anno and Plummer 1962); the Plutonium Laboratory (JN-4) (Flynn et al. 1987; Freas et al. 1971; Rudolph, Kirsch, and Toy 1984); the King Avenue site (Sunderman and Dickerson 1962; BMI 1977); and the West Jefferson site (Evans and Woodward 1978; BCL 2002a).

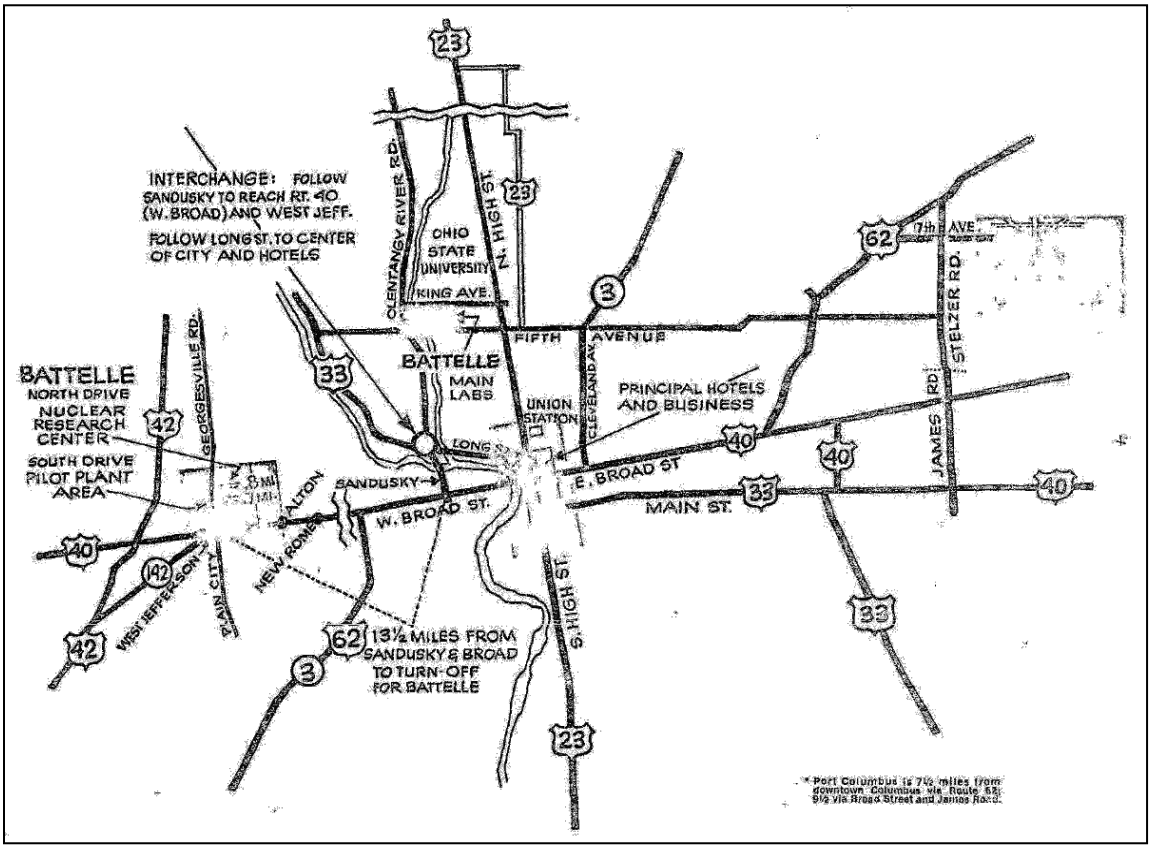

Figure 2-1. Relative locations of the King Avenue and West Jefferson sites (BCL 1974).

1. Manhattan Engineer District (MED), U.S. Atomic Energy Commission (AEC), and U.S. Energy and Research Development Administration (ERDA).

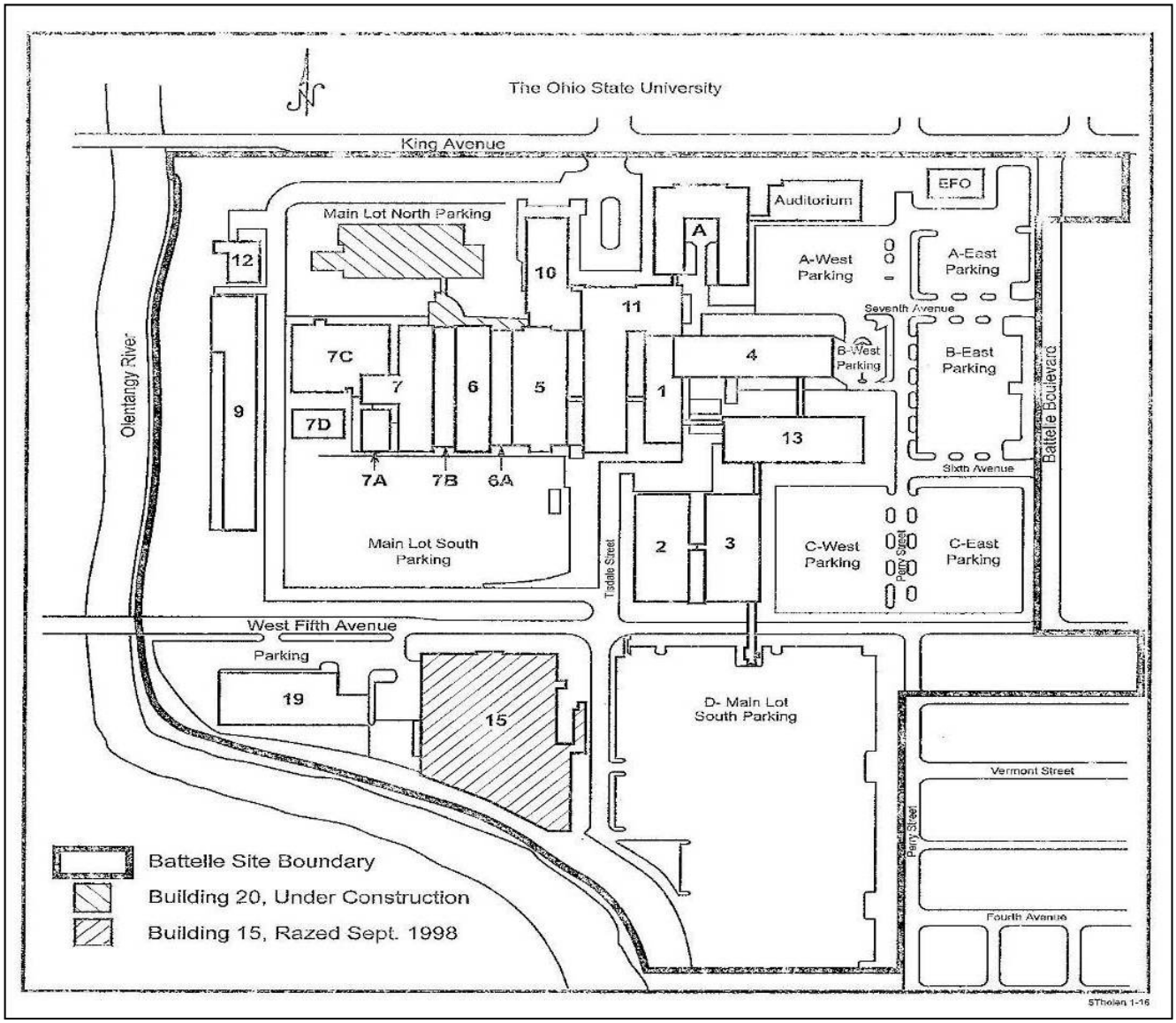

Figure 2-2. Plan view of the King Avenue site (Jenson 2003).

2.1 BUILDINGS OF CONCERN AND THE ASSOCIATED RADIOACTIVE MATERIALS AND OPERATIONS

Attachment A provides an overview chronology of significant events at Battelle, including the start and end of operations. Attachment B is a list of buildings, their major subareas and uses, and the dates of their use. Attachment E lists the incidents that were reported at the sites.

Most of these did not result in a significant internal or external exposure to any worker, but at times Battelle had to do additional monitoring of one or more workers to determine this. The most notable incidents for external exposure from the point of view of potential or actual exposure are those of August 8, 1956, February 10, 1959, August 9, 1961, and July 2, 1980. Those for internal exposure are February 10, 1959, May 9, 1974, January 16, 1979, May 2, 1980, February 27, 1984, and May 1984.

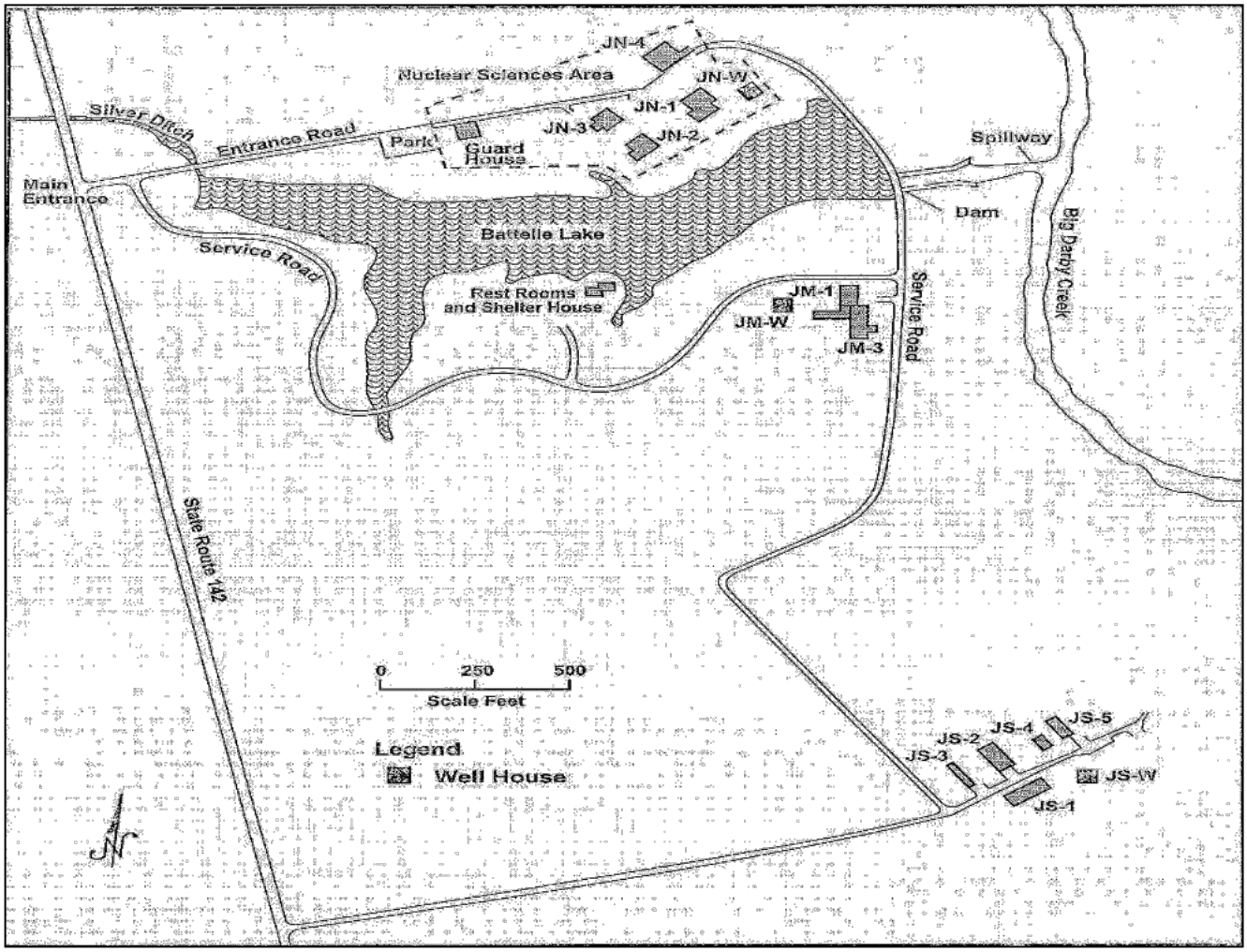

Figure 2-3. Plan view of the West Jefferson site (Jenson 2003).

2.2 ACCESS CONTROLS

At the Radioisotope Laboratory in Building 4, the doors from the corridor into the medium- and high-level laboratories were locked from the outside at all times. Access was otherwise through the locker room, which itself was entered through the office area (Sunderman and Dickerson 1962). These doors were used only for transporting casks and heavy equipment in and out of the laboratory. The laboratory supervisor's permission was required to open the doors.

At the Gamma Facility (Building 6 where Battelle conducted ^{60}Co irradiation), permission for non-staff members to enter the facility had to be obtained from the facility staff (Sunderman and Dickerson 1962). This facility was kept locked overnight, on holidays, and on weekends to prevent unauthorized personnel from entering.

The Hot Cell Laboratory (JN-1) began operations in 1955 (Peters 2001). Access was controlled at all times and, in compliance with 10 CFR Part 20, the entire building was classified as a restricted area (Sunderman and Dickerson 1962; Sunderman and Gates 1965). The locker rooms were divided into two areas separated by walk-through showers; one area was used by personnel not in contact with radioactive material while the other was the dressing area for entry into the contaminated areas (Sunderman and Dickerson 1962). Personnel access into the operating and contamination areas was normally through the hot locker room, but under special conditions entry could also be made through locked doors directly into the operating area or through an equipment storage area (Sunderman and

Dickerson 1962; Myers et al. 1994a). Access to the cells was only through locked doors at the rear of the cells, reached only through the Controlled Access Area (CAA) (Sunderman and Dickerson 1962; Myers et al. 1994a). Access to the Alpha-Gamma Cell (AGC) operating area and receiving dock area was restricted when an AGC dry box was being moved through these areas to the repair station (Gates 1964).

At the Critical Assembly Laboratory (JN-2), before operation each day, interlocks were checked, and there was a visual check of the facility and the critical assembly room (Hogan et al. 1958, Jankowski et al. 1957). The outer doors and doors between the control room and the critical assembly room were required to be locked during operation of the critical assembly; signs were posted before operation (Hogan et al. 1958). Only personnel engaged in activities directly connected with the current operation were allowed to be present in the control room during an experiment. There was a fence around the building to exclude personnel and trucks when necessary (Hogan et al. 1958).

At the Battelle Research Reactor (BRR or JN-3), color-coded badges were used to denote levels of access, depending on the level of training; the top level was a full access badge (Hogan et al. 1958, Plummer, Anno, and Chastain 1960). Visitors from outside Battelle, transient experimenters, maintenance staff, and Battelle non-BRR staff visitors were escorted by reactor staff or by Public Relations Office staff. In general, visitors were permitted only in areas free from experimental activity (Anno and Plummer 1962); otherwise, permission had to be obtained from the Operating Supervisor and personnel monitoring devices (such as visitor badges) were provided (Anno and Plummer 1962; Plummer, Anno, and Chastain 1960).

At the Plutonium Laboratory (JN-4), all outside doors and the doors between the laboratory area and the office area were kept locked from the inside (BCL 1977a).

3.0 OCCUPATIONAL MEDICAL DOSE

In 1943, a MED officer reported that no physical or laboratory medical examinations of Battelle employees had been performed because of their low and intermittent exposures to radioactivity (Ferry 1943). It was known in 1948 that beryllium was handled at Battelle, and that beryllium workers might need to be monitored for adverse health effects (Snapp 1948). An occupational medical program appears to have started at Battelle by 1949, which at first only included the beryllium workers. The medical program consisted of preplacement, an annual periodic exam, and a termination exam, with chest X-rays performed at least during the preplacement and periodic exams. The X-rays were done at University Hospital, which was a short distance away from Battelle (Tabershaw 1949). These X-rays are not eligible for inclusion in dose reconstruction under EEOICPA, because they were performed off site at a noncovered facility (ORAUT 2011a).

During the 1950s, there is evidence that the X-rays of Battelle workers were taken at a private physician's office and included chest fluoroscopy (Wright 2002a, 2002b). Chest fluoroscopy was not a widespread screening method for workers (ORAUT 2011b). By 1957, screening by chest fluoroscopy was discouraged by the American Trudeau Society (1957, p. 3). Chest fluoroscopy of Battelle workers probably ended in 1956 (Wright 2002a; Thomas 2007; Thomas 2010). However, because the chest fluoroscopy and other radiographic chest X-rays during this time occurred off site at a private physician's office, the dose from them is not eligible for inclusion in dose reconstruction (ORAUT 2011a).

It is not known where X-rays might have been taken during 1957-1968. A few claim file records indicate that some X-rays may have occurred offsite at a private physician's office (Thomas 2010). However, onsite X-rays cannot be ruled out for this period (Amstein 2008). Therefore, dose reconstructors should assign dose from chest X-rays starting in 1957, unless the claim file record indicates their X-rays were taken offsite.

The first evidence of medical X-ray equipment at Battelle is from 1968 (Buring 1968). This memorandum describes the shielding evaluation for a new X-ray machine installation in the health service area. The machine is described as a radiographic and fluoroscopic machine. However, there is no evidence that chest fluoroscopy continued to be performed after 1956, so it is assumed that only the radiographic mode was used for occupational health screening. There is no definitive technique factor information in the shielding evaluation documents that can be reliably used to develop site-specific organ doses for radiography using this machine.

Dose reconstructors should assign dose from screening chest X-rays at Battelle starting in 1957. It appears from the available records that the dose from a posterior-anterior (PA) chest X-ray examination should be assigned for the preemployment, annual, and termination physicals for radiation workers (Wright 2002a, 2002b; BCL 1968, p. 12). The records also indicate that a single annual lateral (LAT) chest X-ray was performed in addition to the annual PA radiographic chest X-ray for the period from 1975 through 1980. Some of the records use the phrase "teleo chest" X-ray, which was performed at the standard 72-in. source-to-image distance (SID) to evaluate heart size. The same PA chest X-ray doses should be assigned for these "teleo chest" examinations as for the conventional PA chest X-rays because the SID and X-ray equipment would have been the same for both. X-rays were performed on site at Battelle from 1968 through 1980 (Amstein 2008).

If individual X-ray records are available, the examination type and frequency should be assigned as identified in the records. In the absence of site or individual information, the examination types and frequencies in Table 3-1 should be assigned.

Evidence of when the X-ray machine at Battelle was removed has not been found. Therefore, dose reconstructors should continue to assign dose from chest X-rays until the end of the covered period,

Table 3-1. Default assumptions for X-ray procedure types and frequencies.

Period	Preemployment PA chest	Annual PA chest	LAT chest	Termination PA chest	ORAUT-OTIB-0006 dose
Beginning of operations–1956	All X-rays done off site at noncovered facility	All X-rays done off site at noncovered facility	All X-rays done off site at noncovered facility	All X-rays done off site at noncovered facility	N/A
1957–1967	Yes	Yes	No	Yes	Poor collimation before 1970
1968-1974	Yes	Yes	No	Yes	Proper collimation 1971-1985
1975–1980	Yes	Yes	Yes	Yes	Proper collimation 1971-1985
1981–2000[a]	Yes	Yes	No	Yes	Proper collimation 1971-1985 or Proper collimation 1986-present

a. Note that the end of the covered period for the West Jefferson facility is "the present".

which is 2000 for the King Avenue facility and the present for the West Jefferson facility. Occupational medical X-ray doses are not reconstructed for the residual periods at West Jefferson (1976 through 1985) or King Avenue (2001 to March 1, 2011) in accordance with the guidance of NIOSH (2010).

ORAUT-OTIB-0006, *Dose Reconstruction from Occupational Medical X-Ray Procedures* (ORAUT 2011a provides organ doses for PA and LAT radiographic chest examinations. The doses assigned from ORAUT-OTIB-0006 should be those for the period of "poor collimation before 1970" for Battelle X-rays from 1957 to 1967, and for the period of "proper collimation 1971 through 1985" and "1986 to the present", for Battelle X-rays 1968 to 1985 and 1986 to the present, respectively.

4.0 OCCUPATIONAL INTERNAL DOSE

4.1 IN VITRO BIOASSAY

4.1.1 Urinalysis

Routine urine samples appear nearly always to have been 24-hour samples, although the bioassay records note samples that represent longer or shorter collection periods, for which the results were corrected to 24-hour equivalents. A few spot samples were taken, probably as a check or as the result of an incident; these appear to be identified as such in the records. In the absence of specific recorded information, a urine sample should be considered to represent a 24-hour excretion period.

Table 4-1 is a summary of the primary types of routine in vitro bioassay samples and the associated sampling periods as observed in claimant analytical reports provided by DOE to the NIOSH project. Sampling frequencies are typically monthly, quarterly, or semiannually, depending on the activity in which the worker was involved, or driven by follow-up activities to specific incidents. The earliest urinalysis mention in records is of samples submitted in July 1956 to a toxicology consultant; these samples were assayed only for uranium and the results were given in milligrams (Blauer 1949). It is unclear who performed these initial urine analyses. In about September 1957, Battelle began to send the samples to Nuclear Science and Engineering Corporation (NSEC); these samples were analyzed for either uranium or gross beta (Edelmann 1958, 1959a) through March 18, 1960, then for uranium or gross beta as requested. By at least September 1959, NSEC was analyzing for plutonium (Edelmann 1959b) and by at least February 1962 for uranium, plutonium, gross beta, and tritium (Sunderman and Dickerson 1962). Only one type of analysis appears to have been done per sample; that is, aliquots were apparently not drawn off the same sample for different analyses by NSEC, although this might have been done by Battelle.

Table 4-1. Summary of in vitro bioassay information.[a]

Monitoring type	Radionuclide	Period
Urine, 24-hour	Gross beta	1957–1963
Urine, 24-hour	Uranium (total), radiometric uranium ("RU")[b]	1957–1998
Urine, 24-hour	U-234, U-235, and U-238, a.k.a. specific uranium ("SPU")[c]	1970–1994
Urine, 24-hour	Plutonium–total	1961–1998
Urine, 24-hour	Pu–isotopic (^{238}Pu, ^{239}Pu)	1964–1993
Urine, 24-hour	H-3	1962–1965
Urine, 24-hour	FPs/MFPs	1961–1998
Urine, 24-hour	Thorium–isotopic	1981–1994
Urine, 24-hour	Thorium–total	1993–1998
Fecal	Pu-239, FP, K-40, Ce-141-Ce/Pr-144, Ru-103-Ru/Rh-106, Z-95-Nb-95, U-234, U-235, U-38, Am-241, Sr-90	1975–1983

a. In addition to the information above, enriched uranium, Am-241, Sr-90, Ce-144, Cs-134, Cs-137, Co-60, P-32, I-131, Cr-51, and I-125 were all assayed for urinalyses at different times, as could be seen in the group reports from Eberline.
b. When "RU" is used in the bioassay records, it indicates "radiometric uranium," not recycled uranium (see Section 4.6 Uranium discussion).
c. When "SPU" is used in the records, it indicates the three U isotopes U-234, -235, and -238, not Pu (see Section 4.6 Uranium discussion).

From at least as early as August 1963 to early 1995 (based on inspections of individual records Sunderson and Gates 1965), the samples were analyzed by the Eberline Company. After this, samples were analyzed by Thermo Nutech, of which Eberline was the predecessor company. In the first few years, these samples were analyzed for fission products (FPs), plutonium, polonium, tritium, or uranium (Geiger 1963, 1964a, 1965); from inspection of urinalysis reports, the FPs appear to have been analyzed as gross beta. Later, they were analyzed for some or all of ^{3}H, ^{14}C, ^{32}P, ^{40}K, ^{51}Cr, ^{60}Co, ^{90}Sr, ^{134}Cs, ^{137}Cs, ^{144}Ca, ^{125}I, ^{131}I, europium, ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{241}Am, and/or mixed fission

products (MFPs). Some discussion of recovery for ^{59}Fe, ^{51}Cr, and ^{63}Ni is given in Geiger (1964b), indicating these radionuclides may also have been evaluated. Finally, a research scientist (probably a chemist) had a one-time ^{203}Hg urinalysis in 1971.

From 1957 on, the samples were collected over a nominal 24 hours (Geiger 1963), although it was often noted on records if the sample was actually over a longer period. A few spot samples were taken, probably as a check or as the result of an incident; these appear to be identified as such in the records.

It is not known on what general basis workers were selected for bioassay. Based on the available dosimetry records, claimant statements, and site documents, it appears that King Avenue workers at least the following categories were included in the urinalysis program:

- Workers in the Building 4 Radioisotope Laboratory complex and in the radiotracer laboratories in Buildings A and 1, including office personnel in these facilities;
- Workers in areas of Buildings 2, 3, and 5 where uranium and other radioactive materials were handled; and
- Workers in Building 6 laboratories (except the Gamma Laboratory), but possibly only for specific projects.

It appears that West Jefferson workers in at least the following categories were included:

- All workers at JN-1 during all years of operation and cleanup (about 1956 on),
- All workers at JN-2 during its use as a branch plutonium source testing laboratory (about 1964 to 1970) and possibly during its use as a critical facility (1956 to 1963),
- All workers at JN-3 during all years of operation and cleanup (1956 on), and
- All workers at JN-4 during all years of operation and cleanup (1960 on).

More details are given below. Note that workers in other buildings and facilities than those given above or discussed below might also have been included as necessary based on their job functions. Workers from King Avenue facilities appear to have received baseline bioassay samples before transferring to West Jefferson facilities (Peterseim 1961).

All of the Radioisotopes Laboratory (Building 4 and the two tracer laboratories in Buildings A and 1) along with all JN-1 personnel working with radioactive materials participated in the urinalysis program routinely; secretaries and administrative personnel at these facilities were included (Sunderman and Dickerson 1962). Staff members at JN-4 also submitted urine samples. The program included most of the workers in nonclerical and nonmanagement positions (above the level of supervisors), who were probably engaged in radiological work at least intermittently. There are urinalysis records for a metallurgical technician starting in 1957; this technician is known to have worked with uranium in Building 5. Similarly, from dosimetry records and incident reports, technicians in other areas of King Avenue and West Jefferson also appear to have had regular bioassay and also special bioassays or special attention to regular bioassays when there were incidents.

Staff members at JN-1 submitted urine samples quarterly, while personnel at the Radioisotope Laboratory submitted samples semiannually and secretaries and administrative personnel at both facilities submitted samples annually (Sunderman and Dickerson 1962). Staff members at JN-4 submitted samples quarterly and additionally as needed (e.g., in case of an incident; BCL 1977a). In

individual dosimetry files there are multiple notices sent to workers about their failure to submit urine samples, which are always termed "quarterly" samples. Some of the tabulations on the company cards have the notation "To be submitted [every] 6 months" and on others the notation "3 months."

A set of urine samples analyzed in 1974 were from workers involved in a May 9, 1974, incident in which plutonium was the major airborne contaminant (Geiger 1974); the bioassay records show that urine and fecal samples were submitted approximately weekly for some months and that the urine samples were analyzed initially for ^{239}Pu, MFPs, and ^{90}Sr. Later samples were analyzed for ^{239}Pu and FPs. A summary of the bioassay data for the most exposed worker is shown in Toy (1975a).

4.1.2 Fecal Sample Analysis

The earliest records of fecal samples date from May 1974 and go through at least March 1982, as shown in Table 4-1. Fecal sample analyses were all performed by Eberline and measured ^{40}K, ^{90}Sr, ^{95}Zr-^{95}Nb, ^{103}Ru-^{106}Rh, ^{141}Ce-^{144}Pr, ^{234}U, ^{235}U, ^{238}U, ^{239}Pu, ^{241}Am, and/or MFPs, depending on the year and the case. The set of samples analyzed in 1974 was from workers involved in a May 9, 1974, incident in which plutonium was the major airborne contaminant (Geiger 1974); the bioassay records show that urine and fecal samples were submitted approximately weekly for some months and that the initial fecal samples were assayed for ^{239}Pu, MFPs, and some specific FPs including ^{141}Ce-^{144}Pr, ^{103}Ru-^{106}Ru-^{106}Rh, and ^{95}Zr-^{95}Nb. Later samples were analyzed for ^{239}Pu and FPs.

Fecal samples were taken as appropriate after other incidents (e.g., Langendorfer 1977; Kirsch 1978a).

4.1.3 Nose Swabs

Nose swabs were taken after suspected intakes, as shown in several incident records (e.g., Selander 1959a; Saling 1963; Kirsch 1978a). In addition, it appears that it was routine or at least advisable to take a swab when exiting from dusty cleanup work in JN-4 even in the absence of a known high airborne level, as indicated by an incident report showing that the workers took nose swabs on their own initiative (BCL 1977b). Although information from nose swabs typically cannot quantify intake amounts for dose reconstruction purposes, it could serve to verify whether or not a positive intake occurred after an incident.

4.2 IN VIVO BIOASSAY

In vivo whole-body counting is assumed to have begun in August of 1970, because there were no records found before that date, and to continue through the most recent operations (latest analysis record was from 1998). In vivo counts by Helgeson Scientific included detailed documentation of methods, detection levels, counting parameters, and other related information. Radionuclides detected by whole-body counting varied with each analysis performed, but typically included ^{40}K (naturally occurring), ^{54}Mn, ^{58}Co, ^{60}Co, ^{110}Ag, ^{131}I, ^{134}Cs, and ^{137}Cs. Batch whole-body counting was performed for a group of workers at a frequency of twice per year (generally in March and again in October), although not all workers might have participated at this frequency. Before 1995, minimum sensitivities were recorded for each semiannual batch analysis (often designated as "Appendix III" of the analytical report provided by Helgeson) and are typically included with the individual worker records. If a worker's record indicates that a whole-body count was performed but there is no associated batch record of the minimum sensitivity for a radionuclide, the highest reported minimum detectable amount (MDAs) in the records are listed in Table 4-2 for each occupationally associated radionuclide and may be assumed as favorable to claimants.

Table 4-2. Whole-body counting analytical sensitivity.

Radionuclide	MDA (nCi)
Mn-54	4.4
Co-58	3.0
Co-60	2.0
Ag-110	3.1
I-131	3.6
Cs-134	3.0
Cs-137	3.6

Beginning in 1995, details of the whole-body counts [detectors, minimum detection levels (MDLs) for each radioisotope, etc.] are documented in the individual worker analysis reports provided by the analytical laboratory rather than for a batch of workers.

Although technical documentation included in Helgeson analytical reports described lung-counting protocols, no evidence has been found in the records indicating that lung counts were performed at Battelle.

4.3 LAPEL SAMPLERS

Lapel samplers were used from at least 1975 on (e.g., BCL 1977a; Kirsch 1978a, 1978b; Langendorfer 1977; BMI 1981).

At JN-4, lapel filters associated with routine operations were removed and analyzed daily, with results available before the start of work the next day (BCL 1977a). A new filter was used at the start of any operation with a high potential for exposure (e.g., glove changes, bagging in or out, breaches of primary containment), and the filter was counted at the end of the operation to associate the exposure with the particular operation (BCL 1977a). Lapel samplers were also used as needed at the High Energy Cell at JN-1 (i.e., spent fuel), based on data from counting a worker's personal air sampler filters from April to August 1981 given in BMI (1981). Although information from bioassay sampling should be used rather than from lapel sampling to quantify intake amounts for dose reconstruction purposes, lapel sampling information might serve to verify whether or not a positive intake occurred after an incident.

4.4 IN VITRO BIOASSAY DETECTION LIMITS

Table 4-3 lists the detection limits associated with urinalysis. It should be noted that these values are generally derived from analytical records for Battelle workers provided by DOE to the NIOSH Project; no documentation could be found that specified contractual MDAs in effect for the analytical laboratory, except for the information provided by ORAUT (2007). The detection limits discussed in this section, other than those provided in ORAUT (2007), were derived from stated less-than values as found in the worker records, or from other methods as described below.

Detection limits for urinalysis of a number of individually analyzed radionuclides found in the records (^{144}Ce, ^{134}Cs, ^{137}Cs, ^{60}Co, ^{131}I, ^{203}Hg, and ^{90}Sr) are not listed in Table 4-3 because nonpositive results for these samples typically include a reported MDA or less-than value. This is also true of fecal sample analyses and in vivo analyses. However, for fecal samples, if the report does not include this information, a detection limit of 0.1 pCi/g (ash weight) should be used for routine samples, with an optimum value of 0.03 pCi/g for special samples (ORAUT 2007).

Table 4-3. Urinalysis detection limits data to be used in dose reconstruction.

Isotope	Start	End	Routine detection limit	Units
Total uranium[a]	1957	1995	5	µg/L
Total uranium[a]	1996	1998	1	µg/L
U-234, U-235, U-238	1970	1991	0.3	dpm/sample
U-234, U-235, U-238	1992	1994	0.1	pCi/L
Total plutonium	1961	1991	1	dpm/24 hr
Total plutonium	1992	1998	0.1	pCi/L
Pu-238, Pu-239	1964	1991	0.2	dpm/sample or dpm/24 hr
Pu-238, Pu-239	1992	1993	0.06	pCi/L
H-3	1962	1965	1	µCi/L
Am-241	1974	1980	0.03	dpm/sample
Th-228, Th-230, Th-232	1981	1991	2	dpm/sample
Th-228, Th-230, Th-232	1992	1994	0.2	pCi/L
Total thorium	1993	1998	0.5	pCi/L
Various FPs	1957	1992	0.2	dpm/mL
Various FPs	1993	1998	30	pCi/L

a. Fluorometric analysis through 1995, KPA beginning 1996.

Detection limits for urinalysis were not routinely included in analytical reports before the 1990s, so the information below was derived from records available to NIOSH and information from Eberline (ORAUT 2007). The limits in Table 4-3 should be used when no information is available in the claim records; they do not take precedence over actual MDAs in claim records. In additions, the start and end dates should not be considered absolute if they conflict with information in the claim records.

Detection limits lower than the routine value in Table 4-3 could be achieved for special sampling situations by changing analytical count times, sample aliquot volumes, and other parameters (ORAUT 2007). Therefore, analytical results lower than the values listed above, but are reported as positive results, should be considered valid. Detection limits for the radionuclides in Table 4-3 are discussed further below.

Uranium

Fluorometric analyses for total uranium were performed from the beginning of site operations through 1995. In 1996, kinetic phosphorescence analysis (KPA) became the analytical method to determine total uranium in urine. Early sampling results for uranium were reported in milligrams per liter (1956 to 1964) or micrograms per liter (1964 to 1998) for total uranium mass. These units have been used throughout the operation of the site. Based on claimant records from DOE, the minimum reporting level was 5 µg/L from at least 1964 through 1995 (although not explicitly identified as an MDA, this is an effective MDA because positive results were not reported below this value). After the change to KPA analysis in 1996, the detection level was reduced to 1 µg/L. Some early records (1956 through approximately 1969) indicate that occasional sensitivities below 5 µg/L may have been achieved, but in the absence of specific information in the records the 5-µg/L value should be used for dose reconstruction for years before 1996, and 1 µg/L should be used for 1996 and later years. Note that these samples might have been designated as uranium (or U), fluorometric uranium (FU), total uranium (TU), or natural uranium (NU).

The first records in which reporting of isotopic uranium results (^{234}U, ^{235}U, and ^{238}U) were reported were from 1970. No contractual MDA could be identified for isotopic uranium analyses, which were reported in claimant records in dpm/sample or dpm/24 hours before 1992 and in pCi/L for 1992 and later. Results for 1992 and later, although not explicitly identified in any associated documentation as MDA values, were indicated as "<1 pCi/L," which is consistent with the isotopic MDA provided by Eberline (ORAUT 2007). This reference indicates that the later MDA provided by Eberline (ORAUT 2007) was also applicable for previous periods, which would equate to approximately an MDA of

0.3 dpm/24 hr. Consequently, an MDA value of 0.3 dpm/24 hr should be applied for the period before 1992 for isotopic uranium samples in the absence of information in the record, which is consistent with reported positive values for that period. (This seems to indicate that a change in reporting units, rather than a change in analytical sensitivity, occurred in 1992, which is also supported by the information supplied by Eberline in ORAUT (2007.)

Plutonium
Records of urine sample analyses for total plutonium content were found from the 1960s, then again beginning in 1989, with a change in reporting units in later records. The change in detection limits and the change in units from dpm/24 hr to pCi/L are assumed to occur in 1992 and are based on a general change in reporting units for several other radionuclides (no plutonium urinalysis records were found for 1992). Positive reported values before 1992 were often lower than the routine detection limit in the table, which could be an indication that many of these were possibly special samples that incorporated optimum analytical techniques.

Isotopic analyses for individual plutonium isotopes (^{238}Pu and ^{239}Pu) were also performed as early as 1964. Although the reporting units changed after 1991, the actual detection capability is comparable throughout Battelle operations. Later analytical reports (1990s) are more likely to include the actual MDA for the analysis performed.

Tritium
To date, bioassay analyses for tritium have only been found in the records for 1963 and 1965. One of the samples was positive; for the remainder of the samples, the listed detection limit for tritium urinalysis was 1 µCi/L. Based on the record, it is anticipated that the detection limit will be available in the record if tritium analysis was performed for a worker.

Americium
Urine samples were infrequently analyzed for ^{241}Am, with the earliest sample found in the records in 1977. One sample (in 1978) listed an implied detection limit of 0.03 dpm/sample, while all other samples were recorded as zero with an associated error of 0.03 dpm/sample.

Thorium
Urine samples have been infrequently analyzed for thorium at Battelle. The first sample analyzed for total thorium content found in the records was in 1993. The first isotopic thorium analyses for ^{228}Th, ^{230}Th, and ^{232}Th found in the records were in 1981. Battelle internal correspondence from 1981 indicates that the routine program by that time involved thorium analysis of approximately eight samples per year and shows that thorium urinalysis was included in the blank and spiked samples in the routine quality control reports (Kirsch 1981). Monitoring was performed on frequencies of 6 or 12 months, depending on the work (BCL ca. 1981). A further indication that thorium bioassay was not frequently implemented at Battelle is shown in request forms for individual monitoring, which list thorium bioassay as “other” rather than included in typical analysis types (BMI 1981).

Records indicate an apparent change in detection limit and in reporting units from 2 dpm/sample (1981 to 1991) to 0.2 pCi/L (beginning in 1992) for isotopic thorium analyses. Analytical MDAs are typically included in the analytical report for the latter period.

Fission products
Gross beta analysis of urine samples has been performed for a major part of Battelle operations, with the earliest result found in the records in 1957. The earliest analyses were for both gross beta (which were usually positive due to the contribution of naturally occurring ^{40}K) and for gross beta minus ^{40}K. Because the ^{40}K contribution to the gross beta results would typically have masked any measurement of occupationally related radionuclides, only the gross beta minus ^{40}K results for the early period,

assumed to represent FPs, are useful for assessing intakes. Beginning in 1961, analyses were designated as “FP”; beginning in 1980, they were designated as “MFP”.

Units were initially reported as dpm/mL, then changed to pCi/L in more recent operations; the year listed in the table in which the change in units occurred, 1992, is based on a general change in reporting units for several radionuclides and should be considered approximate. (It should be noted that a few sample results for the earlier period were recorded in units of dpm/24 hr, which is likely a transcription error because the magnitudes of the reported values and associated errors is generally the same as for the results reported in dpm/mL. Consequently, units in “dpm/24 hr” for FP or gross beta analysis results before 1992 should be interpreted as dpm/mL unless the record indicates otherwise.)

Detection capabilities for the analyses were seldom recorded in all but the most recent periods of Battelle operations. Detection capabilities appear to have been related to sample aliquot size, which made them vary from sample to sample. Complicating this were occasional sample results much lower, sometimes by an order of magnitude or more, than typical results, possibly as part of a special sampling or incident follow-up regime. In general, detection capabilities appeared to improve throughout operations. MDAs were not routinely reported until the 1990s. The MDA for the period between 1962 and 1991 was derived empirically based on counting parameters from available claimant analytical reports [1]. The MDA after 1991 is based on information from Eberline (ORAUT 2007).

4.5 INHALATION ABSORPTION TYPES, PARTICLE SIZES, AND ACTIVITY FRACTIONS

For the reactor, the irradiated materials were encapsulated before irradiation and were put into containers as they were removed from irradiation, so that the materials to which reactor workers and researchers were exposed as far as internal dose goes were the normal reactor contamination (FPs and activation products) such as would be encountered in working in the pool or in removing items from the pool. Similarly, the plutonium work in JN-2 involved only encapsulated sources and therefore only incidental contamination was available for intake.

Tables 4-4a and 4-4b provide mixture information for weapons-grade and fuel-grade plutonium from ORAUT (2012) and ORAUT (2005a). The data from these references are applicable to the Hanford and Savannah River sites but are appropriate for application to Battelle because these sites were the primary producers of plutonium for the DOE complex. These mixtures should be applied in the absence of specific information in the record. The activity ratios for “fresh” plutonium in Tables 4-4a and 4-4b should be used for Battelle dose reconstruction if the work involved recently separated weapons or reactor-grade material. If the age of material is unknown, the 10-year aged ratio should be used as favorable to claimants. For heat source material (as indicated by work activity or bioassay monitoring for ^{238}Pu), the activity ratios of all plutonium isotopes other than ^{238}Pu are insignificant, and the material may be assumed to be 100% ^{238}Pu (DOE 2006). The activity of ^{242}Pu in these mixtures does not contribute significantly to dose and need not be considered.

Table 4-4a should be applied to results listed as total plutonium activity; Table 4-4b provides results for associated radionuclides relative to ^{239}Pu when isotopic plutonium analyses were performed.

Table 4-4a. Isotopic composition as fraction of total plutonium alpha activity for plutonium work.

Isotope	Weapons grade material (fresh)[a]	Weapons grade material (10-year aged)[a]	Reactor grade material (fresh)[a]	Reactor grade material (10-year aged)[a]
Pu-238	1.07E–01	9.98E–02	1.77E–01	1.65E–01
Pu-239	7.23E–01	7.29E–01	5.43E–01	5.50E–01
Pu-240	1.70E–01	1.72E–01	2.81E–01	2.85E–01
Pu-241	1.03E+01[b]	6.42E+00[b]	3.19E+01[b]	2.00E+01[b]
Am-241	0.00E+00	1.32E–01	0.00E+00	3.98E–01

a. ORAUT (2005a, 2012).
b. Total plutonium alpha activity does not include Pu-241, which is primarily a beta emitter with insignificant associated alpha activity.

Table 4-4b. Isotopic composition relative to Pu-239 activity for plutonium work.

Isotope	Weapons grade material (fresh)[a]	Weapons grade material (10-year aged)[a]	Reactor grade material (fresh)[a]	Reactor grade material (10-year aged)[a]
Pu-238	1.48E–01	1.37E–01	3.25E–01	3.01E–01
Pu-239	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Pu-240	2.35E–01	2.35E–01	5.17E–01	5.18E–01
Pu-241	1.43E+01	8.82E+00	5.88E+01	3.64E+01
Am-241	0.00E+00	1.80E–01	0.00E+00	7.24E–01

a. ORAUT (2005a, 2012).

Various authors note that ^{241}Am ingrowth should be assumed for ^{239}Pu forms and contamination when the plutonium contains a few or more weight percent of ^{241}Pu and has aged. This ^{241}Am should be assumed to have the same solubility type as the host matrix (DOE 2006). Americium-241 data is included in Tables 4-4a and 4-4b because Flynn et al. (1987) remarks that for the major research activity at JN-4 (i.e., the Mixed Nitride Reactor Fuel Program), ^{241}Am was “reportedly associated” with much of the plutonium material.

If the chemical form of material involved in potential intakes for a worker can be obtained from the record, ICRP guidance (ICRP 1995) may be used to determine the appropriate choice of the inhalation absorption type. If this information cannot be discerned adequately from the claim records, assumptions should be used for these parameters that are favorable to the claimant.

ICRP (1995) indicates that plutonium materials in the workplace can be absorption types M or S, depending on the chemical and physical form of the material. Some intakes of plutonium oxides have exhibited long-term retention of plutonium in the lung exceeding that predicted by the standard type S model. Because oxidation or high-firing temperatures are applicable to some of the plutonium processes at Battelle, type Super S (or type SS) absorption, described in ORAUT (2010a), must be considered as well for situations in which type S plutonium could be encountered. The exception to this is for workers who were primarily exposed to ^{238}Pu, as identified by either work history (e.g., work with heat source material) or bioassay monitoring (e.g., specific analysis for ^{238}Pu only), for which type SS is not applicable in accordance with the guidelines of ORAUT (2010a). For workers primarily exposed to ^{238}Pu, absorption type should be evaluated in accordance with the provisions of ORAUT (2013c) to determine the appropriate model favorable to the claimant.

The dose reconstructor should assume that the route of intake for radionuclides was inhalation and the ICRP Publication 66 default particle size distribution of 5-µm activity median aerodynamic diameter (ICRP 1994). If specific particle size information is available in the record, that information should be used.

4.6 DOSE RECONSTRUCTION CONSIDERATIONS

Fission Products
For the particular case of MFPs for which bioassay was performed with results as only gross beta or gamma measurements, ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2015), may be used to arrive at appropriate source terms for internal dose reconstruction.

The guidance of ORAUT (2015) may be applied to urinalyses for workers at the reactor, JN-1 (the hot cell facility), and at any King Avenue facility where spent fuel (but not solely fission gases) was analyzed. ORAUT (2015) states that to determine if radioiodine intakes need to be used, site profile and claim-specific information should be reviewed to determine if chronic iodine intakes were feasible because iodine exposures are rarely seen in operations that used adequate ventilation and some form of collection, holdup, or filtration; for the workers described above, radioiodine exposure in the course of work with spent fuel or spent fuel samples need not be considered. The spent fuel cooling (decay) times are listed in Table 4-5 for each activity.

Table 4-5. Cooling times for FPs at Battelle.

Activity (facility)	Cooling time
JN-1 (West Jefferson)	1 year
Battelle Research Reactor (West Jefferson)	10 days
Fuel dissolution (King Avenue)	1 year

ORAUT-OTIB-0054 (ORAUT 2015) is applicable to the hot cell facility because the overwhelming bulk of its work was the examination of irradiated reactor fuel specimens and fuel assemblies; this fits into the constraints of ORAUT-OTIB-0054 because as Wastren (2001) states, the types of testing done on this material “did not involve separation of constituent elements from the fuel” and as Pasupathi and Toy (1990) state, operations at JN-1 were almost always dry except for the fuel dissolution work, the acid etching and water rinsing that might be done in the metallographic phase of the examination work, and the water used with the abrasive grinding wheel in fuel sectioning. Because of this, the high-efficiency particulate air (HEPA) filter loads for most cells would consist of particles of spent fuel and activated material (e.g., cladding bits). Irradiated fuel samples were also analyzed at King Avenue in limited ways, but these were small samples and were usually not highly irradiated, so as to limit the external dose rate. Therefore, a few King Avenue laboratory workers and on occasion a few machine shop workers (e.g., BCL 1966–1993) would probably have been exposed to this type of source. Finally, at King Avenue, some test dissolution of clad lightly irradiated fuel pins occurred on and off from 1958 to 1961, as different dissolution processes were tested on a small scale.

It appears that at Battelle, during the period covered by this site profile, all areas where materials containing FPs or activation products were handled had adequate ventilation. The reactor was the pool type and had exhaust over the pool in addition to the building and other exhausts. At JN-1, the spent fuel samples or whole pins or assemblies were opened in the (closed) hot cells, which had individual cell exhausts, and there was a building exhaust as well. At King Avenue such samples appear to have been opened and examined in a glovebox (in the Radioisotope Laboratory) or a hood, which of course had an exhaust system as did the buildings. None of these facilities had iodine filters (except that JN-1 had a charcoal filter from 1978 on to which the exhaust could be diverted if an ^{131}I monitor so indicated). However, HEPA filters were provided for the individual cells at JN-1 and for the individual hoods and glovebox at King Avenue, while prefilters and double HEPA filters were provided for the reactor and JN-1 building exhausts. From at least 1973 (and probably earlier), continuous air monitors were used for the West Jefferson building stacks and on the Building 3 exhaust stack; samples were changed and counted weekly (BCL 1974). The iodine levels were said to be well under ERDA standards. Therefore, these workers probably had little opportunity to be exposed to radioiodine.

The exception was when the radioiodine was part of a specific study, such as of fission gases, or when the iodine could have been used in radiotracer or similar work. In these cases, if the gas sample was taken off to a laboratory for analysis there was some potential for exposure although the sample container was likely opened in a hood or glovebox. However, the fission gases were generally collected off a sample or subassembly in a closed in-reactor sample loop or by vacuum container in a hot cell. In both of these cases the gases were counted in the loop or container or they were allowed to decay in a container so that the progeny could be trapped and counted. In the reactor, the sample or subassembly was removed underwater into a cask in an airtight condition and the gases did not escape to the pool (Basham and Rieder 1960). However, this fission gas work is not covered under ORAUT (2015). Urinalyses done specifically for ^{131}I only (i.e., not along with other FPs for the same urine samples) should generally be considered to be part of fission gas, radiotracer, or similar work. (A description of the fission gas work is given in a footnote to Table C-1.)

The cooling time is assumed to be 1 year for JN-1. This is consistent with Battelle's 1962 accident analysis for JN-1 (Sunderman and Dickerson 1962) that assumed a 4-month cooling time for an Organic-Moderated Reactor Experiment (OMRE) element that was then sent to Battelle for analysis. The actual cooling time was usually longer than 4 months, although it could be as short as 10 days for Fermi assemblies taken to JN-1 (Sunderman and Gates 1965). It is also consistent with the fact that the personnel exposure would have been after the campaign of which the fuel testing was a part (i.e., during cell cleanup when the cell might have been opened or at a minimum when waste and samples were moved out through the ports and transported for packaging or processing and for storage). The cooling time for the BRR and other facilities handling irradiated samples should be assumed to be 10 days, consistent with the short test reactor cycle or the likely turnaround time of the sample. Finally, the cooling time for the fuel dissolution at King Avenue should be assumed to be 1 year, which seems consistent with the actual case of about 10 months for Consolidated Edison test pins irradiated at the Oak Ridge National Laboratory (ORNL) and sent to Battelle for analysis (Dayton and Tipton 1960a).

Uranium

Urinalysis results for uranium from 1976 through 1982 were sometimes designated with the abbreviation "RU," which stood for "radiometric uranium"; this designation should not be confused with recycled uranium. This designation was associated with urinalysis for uranium based on the measurement of gross radioactivity in the processed sample as opposed to the measurement of the mass of uranium in the sample by fluoroscopy. While no rationale has been identified for this method of sample analysis and no detection limits have been found for these samples, it is possible that they were used in an attempt to quantify uranium exposures from enriched uranium (EU), for which measurement of uranium mass in urine samples would have been ineffective. Another designation, infrequently used, that was associated with uranium analysis was "SPU" or "specific uranium" (not to be confused with plutonium analyses).

If the uranium enrichment cannot be determined (e.g., from claim files or information in Attachments C or D of this document), the enrichment associated with fluorometric sampling results (in units of milligrams or micrograms) should be assumed to be natural uranium, using a specific activity of 0.683 pCi/µg. For sample results expressed in total uranium activity or as isotopic results (with units expressed in dpm, microcuries, or becquerel), the measured activity can be assumed to be ^{234}U for the purpose of intake and dose assessment in accordance with ORAUT (2014a).

The extent of the processing of recycled uranium at Battelle is not well known. For all DOE uranium after 1952, this analysis assumed the possibility that uranium from refineries was recycled uranium or contained recycled uranium. Table 4-6 provides the activity as a fraction of uranium activity that should be applied to all uranium intakes after 1952 (NIOSH 2011c).

Table 4-6. Activity fractions of contaminant in recycled uranium.

Recycled uranium contaminant	Pu-239	Np-237	Tc-99	Th-232	Th-228
Activity fraction of contaminant in uranium	0.00246	0.00182	0.379	2.73E-06	2.73E-06

Thorium

Work with thorium metal (forging, rolling, etc.) was performed at King Avenue beginning in 1945 (Dayton undated, p. 24) through 1952 (Thomas 1952), and records indicate thorium metal and ore on site periodically (Chapman 1947; Roth 1948; Saller 1947, 1950; AEC 1951a) through 1957 (Selander 1957a; Peterseim 1957). This is consistent with thorium inventory records indicating large quantities of thorium on site (hundreds of kilograms) through 1957, with significant reductions in inventory after 1957 (BCL 1988a, pp. 45–127). No record of similar work after this time was found, although laboratory research with small quantities of thorium is recorded for later periods through the 1960s (Norris 1960; Evans 1963).

For early work evaluating methods of thorium extraction from monazite sands (1949-1950), thorium and progeny radionuclides would have been in equilibrium in the source material. For work with thorium at other periods, the degree of chemical separation of thorium from its progeny are generally unknown. Consequently, for evaluation of available thorium bioassay results, dose reconstructors should use the methods described in ORAUT (2014b) to determine intakes favorable to the claimant with respect to the degree of disequilibrium of thorium from progeny radionuclides due to possible chemical separation.

In accordance with Battelle-SEC-00208 (NIOSH 2012) and Battelle-SEC-00229 (NIOSH 2015), internal dosimetry for thorium is considered infeasible from the beginning of site operations through December 31, 1970, due to lack of documented bioassay or air sampling records during periods in which thorium operations are assumed to have taken place based on records of thorium inventories and contamination surveys indicating potential thorium exposure. After this period, NIOSH has not identified data that suggest the potential for significant thorium exposure, and the Battelle internal dosimetry program had implemented radionuclide-specific bioassay monitoring to a degree so that workers performing significant work with thorium would have been monitored.

4.7 RECORDS AND REPORTS

The urinalysis and other bioassay records available for the Battelle sites are of three types. The first type is the bioassay report issued by the bioassay analytical service provider; the second type is the Battelle (company) summary card on which an individual worker's urinalysis information was tabulated, usually over a year; and the third type is the Battelle ad hoc documentation, such as incident reports that give information as to potential exposure levels and routes. The principal record to be used in dose reconstruction is the report from the bioassay analytical service provider, but the other two record types may provide information relevant to missing bioassay readings or to special bioassays sampling situations.

5.0 OCCUPATIONAL EXTERNAL DOSE

5.1 DOSIMETRY PROCESSING SERVICES

It is not clear when badging started at Battelle. The earliest date for a film badge that has been found in the available records is in March 1951. Badges appear to have been worn thereafter throughout the period of operation and the period of residual radioactivity, including during decontamination and decommissioning (D&D). Table 5-1 gives summary information about badge service providers.

Table 5-1. Dosimetry processing services.

Period	Processing service
March 1951–March 1956	Unknown
April 1956–1967	Landauer
1968–1969[a]	Eberline
1969[a]–June 1994	Landauer
July 1994–1996	Siemens/ICN
1997–present	Landauer

a. Records indicate that both Landauer and Eberline provided processing service in 1969.

Badges appear always to have been processed by a film service, although it is possible that the first badges (before 1956) were read by the AEC New York Operations Office (NYOO). Landauer performed film dosimeter processing from April of 1956 through 1967. In 1968, processing was performed by Eberline. In 1969, records show that both Eberline and Landauer were used as processors. Landauer was again the sole processor beginning in 1970 and continued film processing through June 1994; thermoluminescent dosimeters (TLDs) were added in 1984. It should be noted that some TLD badges were in use before 1984, and film badges continued to be used well after this date for some individuals. Processing was performed by Siemens/ICN from July 1994 through 1996, then again by Landauer beginning in 1997 through the present.

5.2 DOSIMETER TYPES

Film dosimeters were used to measure whole-body dose from the start of monitoring through at least 1984, with whole-body TLD dosimeters introduced in 1985. During the film badge period, either beta, gamma, and X-ray sensitive dosimeters or beta, gamma, X-ray, and Kodak nuclear track emulsion, type A (NTA) dosimeters were issued (BCL 1986). With the introduction of whole-body TLD dosimeters, neutron-sensitive TLD materials, and CR-39 neutron dosimeters (chemically etched plastic measuring recoil proton tracks from fast neutron interactions) were also implemented.

Ring or wrist badges were worn routinely by most laboratory, hot cell, machine shop, and similar workers who handled hot sources, equipment, or waste with their hands, even if lead gloves were also worn (e.g., Ottman 1959, 1962b, 1968, 1970; Kirsch 1976; Toy 1981). Ring badges appear to have used TLDs beginning in 1970. At JN-1, each person who was temporarily or permanently assigned to the facility was issued a ring badge along with a basic film badge (Sunderman and Dickerson 1962; Sunderman and Gates 1965). At JN-4, as of at least 1977 (the year before D&D began) staff members were issued beta-gamma and X-ray TLD ring badges and beta-gamma, X-ray, and neutron wrist badges (BCL 1977a); ring badges were required to be worn by all persons who worked with gloveboxes (BCL 1977a).

The use of pencil (pocket or self-reading) dosimeters is mentioned in several individual dosimetry records. It was noted specifically for the Radioisotopes Laboratory (Building 4), the Hot Cell Laboratory (JN-4), and presumably for the Gamma Facility (Building 6) that when staff members worked in areas where they might receive an exposure that approached the maximum permissible daily dose, they wore pocket ionization chambers to keep track of their whole-body gamma dose

(Sunderman and Dickerson 1962; Sunderman and Gates 1965). This apparently included every entry into a radiation area with contamination because a pocket dosimeter was listed among the standard protective clothing and equipment set for such entries (Sunderman and Dickerson 1962; Sunderman and Gates 1965).

5.3 DOSIMETER DETECTION CAPABILITIES

Table 5-2 gives a summary of the MDLs for evaluation of exposures that were measured by film badges and TLDs and for assessing missed dose (Landauer 1956–1997).

Table 5-2. Summary of MDLs (mR through 1958, mrem beginning 1959) for dosimeter badges.

Period	Photon	Beta	Neutron (fast)	Neutron (slow)
03/1951–03/1956	50	Not available	Not available	Not available
04/1956–03/1957	30	50[a]	60[b]	Not available
04/1957–1960	20	50	60	Not available
1961–1963	10	40	50[c]	Not available
1964–1984	10	40	50	10
1985–06/1994	10	40	50	10
07/1994–1996	10	40	50	Not available
1997–present	10	40	20	10

a. Generally no beta doses were reported before April 1957, although records indicate beta measurements might have been made at times for work with uranium. If beta dose monitoring is indicated, a 50-mrem MDL is recommended.
b. Based on values recorded as "under 10% MPL," the MPL for the period was 300 mR/wk, so 10% of that value for a 2-week period is assumed to be 60 mR.
c. There is inconsistency in the fast neutron MDL. In early 1964, an MDL of 15 mrem is stated. Later reports identify 20 mrem, and then 50 mrem from 1994 to 1996. Because it is unlikely that an actual decrease in sensitivity would occur, an MDL of 50 mrem is recommended from 1961 to 1996.

In accordance with the SEC described in Section 1.3 (Burwell 2016), NIOSH found that it was not feasible to reconstruct external doses from beta, gamma, and neutron radiation through February 13, 1951.

Film badge reports from before April 1956 list only gamma and X-ray exposure components that are tabulated from shielded and unshielded dosimeter readings; units are in milliroentgen, with the MDL listed as <50 mR.

Beginning in April 1956, neutron exposure was tabulated and specified as "fast neutron" with the results in percent of the maximum permissible limit (MPL); the MDL was typically recorded as "<10% of the MPL" with no other units listed. Because the MPL for the period (also listed on the reports) was 300 mR and the dosimeters were exchanged on a biweekly frequency, it is assumed that the MDL for fast neutrons was 60 mR (i.e., 10% of 300 mR/wk for a 2-week monitoring period) during this period. The bottom of the dose report forms stated that a recorded gamma exposure of zero indicated less than 30 mR unless otherwise noted.

In April 1957, the minimum reported dose value was changed to 20 mR for high-energy X- and gamma rays. Neutron exposures continued to be listed as described for 1956.

As early as May 1956, a category headed "Other" was occasionally included on reports. This category is assumed to represent beta dose because in February 1958 the category heading was changed to "Other, Arbitrary Units," and Landauer applied this terminology in reporting later Battelle beta dose and in reports and memoranda dealing with beta dose (e.g., Ottman 1959, 1961). Units of

millirep for uranium beta exposures were also noted during this period; the heading "Beta" was not routinely included in reports until 1961.

For doses reported during this period as "other" or "arbitrary units" (sometimes abbreviated "a.u."), uncorrected film readings were recorded representing the unshielded portion of the dosimeter reading (i.e., due to electrons), which were then converted to millirep of dose using factors derived based on assumed beta electron energy (Storm 1951; Landauer 1951). These values were then sometimes converted to reported dose to the skin (indicated as "D_s"), by adding the gamma dose to the converted electron dose. Battelle records occasionally include notations indicating either the assumed beta electron energy, the factor to be applied, or both. For exposures to "old" FPs (defined as greater than 6 months old), the assumed energy was listed as <0.7 MeV with an associated multiplication factor of 5.3. For exposures to "new" FPs (defined as less than 6 months old), the assumed energy was listed as <1.4 MeV with an associated multiplication factor of 2.2. Other factors were sometimes listed depending on the exposure conditions, but no associated assumptions have been found in the documentation other than for new or old FP exposures. For the period in which arbitrary units were used (i.e., before 1961), dose reconstructors should multiply recorded electron doses by factors as documented in the record, if available. For work with uranium, a factor of 1.6 is applied based on Landauer (1951) and notations in the records. For work with fission products, if no factor is documented in the dose record, but the exposure source is noted as "new fission products," the dose reconstructor should apply a multiplication factor of 2.2 to the recorded value to obtain the shallow dose. If no factor or beta energy is documented in the dose record, the approach most favorable to the claimant is to assume exposure to old FPs and apply a multiplication factor of 5.3 to obtain the shallow dose. Note that the use of arbitrary units was discontinued beginning in 1961 (although exposure questionnaire forms continued to include the abbreviation "a.u." for a few years afterwards). Beta doses reported in units of millirem or "millirep if U beta" do not need to be adjusted.

Beginning in 1961, Landauer reports began listing dose categories of gamma- and X-ray, beta, and neutron, with MDLs of 10 mrem for gamma- and high-energy X-rays, 40 mrem for beta, and 15 mrem for neutrons (Landauer 1956–1997). It should be noted that from 1961 through the first part of 1964, Landauer listed an MDL of 15 mrem for fast neutrons, which was changed to 20 mrem later in 1964 and for subsequent years through the present. The value of 50 mrem is used as a default value in this site profile as a more realistic estimate and because it is unlikely that a decrease in detection sensitivity occurred during the later period. The neutron doses were not categorized by energy, so it is assumed that the doses described fast neutrons, as was the case before 1961. It is also noted that neutron doses changed from percentages of the MPL to millirem in 1961.

Beginning in 1963, reports began to distinguish between thermal and fast neutron exposures, with stated MDLs of 15 mrem for fast neutrons and 10 mrem for thermal neutrons. As noted above, an MDL of 50 mrem is assumed for fast neutrons in this site profile. No changes to the Landauer MDLs were noted after 1963; therefore, the values of 10 mrem (gamma- and high-energy X-rays), 40 mrem (beta), 50 mrem (fast neutrons), and 10 mrem (thermal neutrons) were used for all periods of Landauer processing beginning in 1963 through 1996 (Landauer 1956–1997; SC&A 2016).

Landauer continued film dosimeter processing until 1967. In 1968, processing was performed by Eberline. Contract proposal documentation for Eberline indicates reporting ranges comparable to the Landauer MDLs (Eberline ca. 1967). Because no other documentation was found for Eberline that related to MDLs, the same values were used during the period of Eberline processing. As noted above, both Eberline and Landauer were used as processing services for 1969. Landauer was again the sole processor beginning in 1970 and continued film processing through June 1994; TLD dosimeters were added to the service in 1984. As noted above, records indicate some TLD badges were in use before 1984, and film badges continued to be used well after this date for some individuals.

Processing was performed by Siemens/ICN from July 1994 through 1996, with MDLs as noted in Table 5-2, then again by Landauer beginning in 1997 through the present.

5.4 DOSIMETER EXCHANGE FREQUENCIES

Table 5-3 shows the default dosimeter frequencies that were used to monitor the primary groups of workers at Battelle (BMI undated; BCL 1973; Author unknown ca. 1975).

Table 5-3. Default dosimeter exchange frequency by facility or work group.

Facility or work group	Account No. (beginning with 1969 Landauer reports)	2-week exchange frequency	4-week exchange frequency (before 1970) or monthly (after 1970)
Battelle Research Reactor (BRR)	6410	Yes	No
Critical Assembly Laboratory (CAL)	6412A	Yes	No
Hot cell[a]	6408	Yes	No
Plutonium Laboratory	6409	Yes	No
Uranium Laboratory (25 area)	6412B	Yes	No
Safety	--	Yes	No
Visitor biweekly	6412Z	Yes	No
Miscellaneous biweekly	6412D	Yes	No
Radioisotope Laboratory	2302	No	Yes
Analytical chemistry	2366	No	Yes
Radiography	2613C, 6412C	No	Yes
Visitor monthly	6413Y	No	Yes
Miscellaneous monthly	6412E, 6413E	No	Yes

a. Hot cell workers might also have been on a weekly exchange frequency (Sunderman and Dickerson 1962).

External dose records, computer-assisted telephone interviews, and program documentation from Battelle indicated that badges were exchanged at 1-, 2-, and 4-week, monthly, and irregular ("special") intervals dependent on the worker's job (BMI undated; BMI 1962–1963; BCL 1973; Author unknown ca. 1975). Records from Battelle document that the primary exchange frequencies were biweekly and every 4 weeks, as shown in Table 5-3 (the 4-week frequency was changed to monthly in 1970). The exchange frequency was typically stated on Landauer reports that date from at least 1961. The account numbers in Table 5-3 were also listed on Landauer reports from 1969 through about 1984. If information in claimant records is insufficient to determine the worker's dosimeter exchange frequency, then the dosimeter exchange frequency should be based on the work group that is shown in Table 5-3; if the work group cannot be determined, then a biweekly exchange frequency should be assumed to be favorable to the claimant.

Records show that dosimeter exchange frequencies were specified by Battelle based on work location and the type of work, and they might have been different for different work groups during the same period or for individual workers over their periods of employment. Records for one individual included weekly, biweekly, and monthly monitoring frequencies including 2 consecutive years when he was monitored with all frequencies. Another individual had several "special" dosimeters that were apparently multiple dosimeters (i.e., more than one dosimeter was issued to the individual), but that were issued and retrieved at nonoverlapping periods in 1965 – these might have been in accordance with multibadging or planned special exposure procedures (Stewart 1995). Concurrent use of film and TLD dosimeters was also noted. The dose reconstructor is responsible for evaluating available records to determine the monitoring frequency that is most appropriate to support missed dose assessments.

It should be noted that dose monitoring records appear to be missing for short intervals in some individual records. This could be due to actual missing records or due to the fact that Battelle based the level of monitoring on the exposure potential of the worker and task, so that individuals with little

or no potential exposure were not monitored (i.e., the lack of monitoring was intentional based on low dose potential). The dose reconstructor must evaluate available records and determine whether apparent gaps in monitoring require the assessment of unmonitored dose for those periods. Battelle records appear to be adequate to support interpolation methods outlined in OCAS-IG-001 (NIOSH 2007, Section 3) to determine doses for gaps in records that are deemed to be due to missing reports (rather than to the intentional nonmonitoring of workers with low potential for exposure). More significant gaps in monitoring data must be evaluated on a case-by-case basis to determine if additional requests for dosimetry records should be made.

5.5 WORKPLACE RADIATION FIELDS

Table 5-4 summarizes photon energy ranges for materials and their related radiation types at the King Avenue and West Jefferson sites (ORAUT 2009, 2010b, 2011b, 2013b).

Table 5-4. Photon and energy ranges for material types.

Energy	Natural or depleted uranium	Enriched uranium	Natural thorium, MFP	Plutonium
<30 keV	0%	0%	0%	65%
30–250 keV	50%	100%	25%	35%
>250 keV	50%	0%	75%	0%

If information on the specific materials to which a worker was exposed cannot be determined, 100% 30-250 keV photons should be assumed to provide a deep dose favorable to claimants.

For shallow doses, either 100% >15-keV electrons (for nonplutonium work) or >30-keV photons (for plutonium work) should be used.

For neutron doses, a 100% 0.1 to 2 MeV energy should be assumed to provide a dose favorable to claimants. Note that because the West Jefferson facilities did not start up until 1955 or later, exposure to reactor neutrons or plutonium-emitted neutrons was not a possibility, except for the limited ^{239}Pu work in a laboratory in Building 6 in 1953 to 1955 and 1961. However, the possibility of neutron exposure from the alpha-neutron reaction in appropriate forms, especially for UO_2 and ThO_2, should still be considered in accordance with the guidance of ORAUT-OTIB-0024, *Estimation of Neutron Dose Rates from Alpha-Neutron Reactions in Uranium and Thorium Compounds* (ORAUT 2005b).

For neutron doses, the measured doses should be modified by the ICRP Publication 60 correction factor for the energy range (ICRP 1991). This factor incorporates the ratio of the Publication 60 weighting factor to the group-averaged National Council on Radiation Protection and Measurement (NCRP) Report 38 (NCRP 1971) quality factor. This factor is 1.91 for the 0.1 to 2 MeV energy range (ORAUT 2006a).

Because little information is available on the exposure geometry for an individual, the standard assumption is that all exposures are for AP geometry, which is favorable to the claimant for most organs. In accordance with the NIOSH *External Dose Reconstruction Implementation Guideline* (NIOSH 2007), rotational and isotropic geometries must also be evaluated in accordance with that document for bone surface, red bone marrow, esophagus, or lung exposures. The geometry most favorable to the claimant from this comparison should be used for these organs unless workplace information indicates that AP geometry is applicable for a specific exposure situation.

Early film dosimeters were calibrated in roentgens. It is assumed that this continued until the implementation of TLD dosimeters in 1985, for which the dose equivalent, Hp(10), is the appropriate unit for calibration. Table 5-5 lists the dose units to use for organ dose conversion factors.

Table 5-5. Photon dose units for use with organ dose conversion factors.

Years	Dose unit
Start of operation–1984	Exposure - *R*
1985–present	Deep dose equivalent - *Hp(10)*

Workers at the West Jefferson site (JN-1, JN-2, JN-3, and JN-4) might have worked with gloveboxes. The presence of extremity monitoring (wrist or ring data) can be used as an indicator for glovebox work or handling radioactive material [2]. The application of the guidance in NIOSH (2011a) appears to be appropriate in relation to workers who handled plutonium in gloveboxes at JN-2 (1964 to 1970) and JN-4.

5.6 ASSESSMENT OF NEUTRON DOSE MONITORED BY FILM

The Battelle West Jefferson Nuclear Sciences Area contains several major buildings with the potential for workplace radiation fields (Amstein 2010) as described in Attachment F. Due to the underresponse of NTA film dosimeters to low-energy neutrons, some neutron exposures might have been unmonitored or inadequately monitored by NTA dosimeters. Consequently, except for exposures to neutrons from ^{252}Cf at JN-1 (as discussed in Attachment F), neutron doses for the film dosimeter period should be assessed based on neutron-to-photon (NP) exposure ratios derived from paired workplace radiation measurements (i.e., same time and location) of neutron and photon radiation dose rates from selected Battelle instrument survey records. These ratios should be applied to measured and missed photon doses to obtain neutron doses for workers for the period when film dosimeters were in use (i.e., before 1985) and neutron exposures would not have been adequately monitored by NTA film. These ratios should not be used if records indicate that TLD dosimeters were used to for monitoring. The recommended NP ratios are summarized in Table 5-6, based on the analysis in Attachment F.

Table 5-6. Recommended NP ratios for dose monitored by NTA film.

Building	Location/Work	Years	n	GM	GSD	95th%
JN-1	Non-Cf-252 operations	1955–1983[a]	(b)	(b)	(b)	(b)
JN-1	Cf-252 0perations	1971–1974	49	2.4	3.9	22.5
JN-2 Critical Assembly Laboratory	Critical experiments	1956–1960	(b)	(b)	(b)	(b)
JN-2	Plutonium Laboratory	1967–1975	14	1.0	1.3	1.5
JN-3 Battelle Research Reactor	Combined areas	1956–1974	2,487	0.2	4.2	2.1
JN-4 Plutonium Laboratory	Combined areas	1960–1978	41	0.7	5.0	9.9
JS-1	Naval Reactor Research	(c)	(c)	(c)	(c)	(c)

a. Neutron exposures probably very limited before construction and operation of alpha gamma cells in 1964.
b. No paired survey data received.
c. No paired survey data received. Depleted U-238 used in this building with essentially no potential for significant neutron radiation.

Paired workplace radiation measurements (i.e., same time and location) of neutron and photon radiation dose rates were obtained from selected instrument survey records extending over several years for Battelle facilities as follows:

- JN-1 Hot Cell Laboratory (1955 to 1983). Limited survey data has been received from Battelle consisting of 49 measurements for the period when ^{252}Cf was being processed and handled in very large quantities. NP ratio analysis based on this measured data presents an upper bound that if applied to claimants over an entire work career would result in a significant overestimate of their neutron dose.

- JN-2 Critical Assembly Laboratory (CAL) (1955 to 1975). Limited survey data consisting of 15 measurements has been received from Battelle and for the more recent operational period only. No survey data has been received from Battelle for the early years of operation. The Battelle-provided information for the Plutonium Laboratory in this building involved only sealed sources.

- JN-3 Battelle Research Reactor (BRR) (1960 to 1973). Substantial survey data was received from Battelle for the period from 1960 through 1974 consisting of 2,487 paired measurements selected for analysis from 3,188 paired measurements identified in the documentation. This is considered to be substantial data to develop a robust estimate of the NP ratio for this facility.

- JN-4 Plutonium Laboratory (1968 to 1976). Survey data was received from Battelle consisting of 67 paired measurements. This is considered to be sufficient to develop an estimate of the NP ratio for this facility.

- JS-1 Naval Reactor Research Laboratory. No paired neutron and photon dose measurements were received from Battelle apparently because there were no significant neutron radiation fields in this building. The work involved some depleted uranium (DU) historically. The building was decommissioned in 1990 and based on the type of operation would not be expected to have significant neutron radiation fields.

Results of statistical analyses of the measurement data are summarized in the Table 5-6.

5.7 INCIDENT REPORTS

A form for reporting minor incidents such as overexposures, lost badges, etc., was used as early as June 1958 (e.g., Selander 1958a, 1958b); the form was also used to document planned overexposures (e.g., Swigert 1963). While most of the copies of these forms found in the records apply to occurrences at the West Jefferson site (mostly at the BRR and the hot cell facility), a few apply to occurrences at the King Avenue site (e.g., Kirsch 1976). It is likely that there were fewer such reports for King Avenue because of the lower potential for significant exposures.

Attachment E contains a detailed summary of incidents identified at the King Avenue and West Jefferson sites.

5.8 ONSITE AMBIENT EXTERNAL DOSE

Annual environmental reports in records obtained from Battelle range from 1973 through 2005, with the exception of 1999, for which no report was found (doses for 1999 are addressed as described below). These reports are listed in the Environmental Reports section of the References list. Evaluation of records indicates that quarterly measurements of external radiation using lithium fluoride TLDs began 1978 at the West Jefferson site at 14 monitoring locations within a 0.75-mi radius of the site and 25 monitoring locations at the security perimeter. The innermost monitoring locations at the security perimeter were used to derive ambient external dose values for the site in this document. Monitoring locations were changed to the “recreation area and property boundary line” in 1985 (15 locations), increasing to 16 locations in 1991. In the fourth quarter of 2004, monitoring locations changed from 16 locations around the West Jefferson property boundary to 10 locations around the security perimeter of the innermost facility buildings.

Monitoring was also performed during remediation activities at 15 locations at the King Avenue facility from the fourth quarter of 1993 through 1998.In reporting potential dose to an individual based on environmental monitoring results, Battelle environmental reports indicate that monitoring results were consistently below levels expected from naturally occurring background radiation. Reports indicate a

terrestrial component of 0.060 rem/yr (based on aerial monitoring and stated to be consistent with the national average) and indicate a cosmic ray component for Ohio of 0.050 rem/yr (slightly above the national average of 0.045 rem/yr). However, because exact contributions from offsite sources cannot be determined, and to ensure results favorable to the claimant, this document assumes the entire reported result from environmental monitoring to be potentially due to site operations.

For years in which the monitoring locations were at the security perimeter (1978 through 1984, and 2005), the average annual doses were higher (0.145 rem) than when the monitoring locations were at the recreation area and property boundary line (1985 through 2004, with an average annual dose of 0.110 rem). This suggests that onsite ambient doses to workers are more accurately represented by monitoring stations in closer proximity to the facility buildings. For this reason, the higher annual average from security perimeter locations was applied in this document for 1985 through 2004 rather than the lower annual doses from when the monitoring locations were farther from the facility buildings. This higher annual average dose is also considered favorable to claimants for 1999, for which no annual environmental report was located.

Results from King Avenue measurements were comparable to, but consistently less than, those from the West Jefferson site for the corresponding year. Therefore, reported values for the West Jefferson site are applied as bounding doses that are favorable to claimants for the King Avenue site for all years, which also accounts for individuals who might have traveled between the two sites.

Annual doses obtained as described above are for an assumed calendar year of 8,760 hours. These annual doses are prorated in this document to the dose that would be received by an occupational worker using an occupancy factor of 2,000 working hours per year.

From the beginning of operations through 1977, prior to the availability of records for security perimeter locations, because no site activities at West Jefferson are evident that would suggest a substantial deviation from operations during the monitored period, the highest annual doses occurring between 1978 and 2005 may be applied as favorable to claimants. For years after 2005, the average annual dose from security perimeter locations is applied. Because King Avenue results were consistently lower than those from West Jefferson, and because materials with greater external dose potential were consistently in use at West Jefferson, these values may also be applied for the King Avenue site.

Table 5-7 provides the reported averages for annual external exposures for the calendar years indicated as measured by environmental TLD badges. Table 5-8 provides the annual doses, prorated for an occupational year as described above and assumed to be from 30- to 250-keV photons, to be applied in dose reconstructions for individuals not monitored for external exposure, applied as lognormal distributions with a geometric standard deviation of 3.

Table 5-7. Annual environmental dose (rem/yr) from external background radiation as measured by TLD.

Year	West Jefferson	King Avenue
Start of operations – 1977	No reports	No reports
1978	<0.133	No report
1979	<0.144	No report
1980	<0.143	No report
1981	<0.128	No report
1982	<0.129	No report
1983	<0.130	No report
1984	<0.211	No report
1985	0.120	No report
1986	0.120	No report
1987	0.120	No report
1988	<0.120	No report
1989	<0.120	No report
1990	<0.132	No report
1991	<0.120	No report
1992	<0.120	No report
1993	<0.120	<0.030[a]
1994	<0.120	<0.030[a]
1995	0.094	0.090
1996	0.089	0.088
1997	0.091	0.088
1998	0.088	0.081
1999	No report	No report
2000	0.109	No report
2001	0.105	No report
2002	0.096	No report
2003	0.104	No report
2004	0.111	No report
2005	0.143	No report
2006–present	No reports	No reports

a. Represents reported quarterly results, all of which were <0.030 (equivalent to annual value of <0.120)

Table 5-8. Prorated annual dose (rem/yr) from external gamma radiation.

Year	West Jefferson and King Avenue
Start of operations–1977	0.048[a]
1978	0.030
1979	0.033
1980	0.033
1981	0.029
1982	0.029
1983	0.030
1984	0.048
1985–present	0.033[b]

a. Based on maximum recorded annual value for all reported years.
b. Based on average annual dose from security perimeter locations.

6.0 INTERNAL DOSE FROM ONSITE ATMOSPHERIC RADIONUCLIDE CONCENTRATIONS

6.1 AIR SAMPLING AND RELEASE ESTIMATES

No environmental information is available for either the King Avenue site (operations began in 1943) or the West Jefferson Nuclear Sciences Area (operations began in 1956) until 1973. Information used to make estimates on inhaled radionuclides for this period is described below.

6.1.1 King Avenue, 1943 to 1972

No information is available for the King Avenue site until 1973. Release information for subsequent years is described below.

6.1.2 King Avenue, 1973 to 1975

Most air emission monitoring and sampling of BCL facilities occurred at the West Jefferson site. King Avenue stack sampling results for total released activity are available for 1973 (presumably the sum of gross alpha and gross beta measurements). The King Avenue results are less than 4% of West Jefferson gross alpha plus gross beta results in 1973; separate gross alpha and gross beta releases are available for 1974 and 1975. King Avenue sampling was directed at Building 3, where activities with uranium compounds took place (BCL 1974). In June of 1975 the King Avenue stack sampling system was placed on standby as the reduced inventory of radioactive materials did not warrant continued sampling (BCL 1976). Site boundary air concentrations are reported, and typically represent MDAs. The maximum releases reported from the King Avenue facilities between 1973 and 1975 were 3.18×10^{-8} Ci/y for gross alpha emitters (assumed to be uranium) and 2.46×10^{-6} Ci/y for gross beta emitters (assumed to be ^{137}Cs).

6.1.3 West Jefferson, 1956 to 1972

As noted above, no information is available for the West Jefferson Nuclear Science Area from before 1973. Release information for subsequent years is described below.

6.1.4 West Jefferson, 1973 to 2005

Stack release information for the West Jefferson site became more detailed from 1973 to 1975. In 1973 only gross alpha and gross beta releases (3.6 and 29 µCi, respectively) were reported for JN-1, and ^{239}Pu releases (0.36 µCi) were reported for JN-2 and JN-4. The gross alpha, gross beta, and ^{239}Pu released activities were similar to 1974 (1.9, 25, and 0.38 µCi, respectively) and 1975 (1.1, 25, and 0.36 µCi). Gross alpha and gross beta releases continued to decrease in subsequent years, to generally less than 1 µCi gross alpha and 10 µCi gross beta, as radionuclide-specific information became more detailed. In 1974 radionuclide-specific information was added for ^{131}I and ^{85}Kr. MFPs were first identified individually in 1975, and ^{235}U was first reported in 1977. The actual radionuclides reported varied from year to year, particularly for FPs; naturally occurring radionuclides such as ^{40}K were sometimes included. Uranium-238 and ^{241}Am began to be included in 1988 and ^{234}U in 1996. Iodine-131 was not reported after 1980 and ^{85}Kr after 1987. Where applicable, values below the MDA were reported at that activity. Activities released per year, average in-stack concentrations, and site boundary concentrations were typically given for each stack every year (data were not available for 1999). Figures 6-1 and 6-2 show the calculated intakes of alpha-emitting radionuclides and MFPs, respectively, from 1973 through 2005.

Air emissions were continuously monitored using in-stack air samplers on nine exhaust stacks from three different West Jefferson facilities during 1975, which was selected as a “typical” year for the period, the first for which relatively complete radionuclide release data were available (BCL 1976).

These release points, their descriptions, and the analyses conducted are shown in Table 6-1. Particulate filters were collected and analyzed weekly, while ^{131}I was analyzed monthly and ^{85}Kr releases were monitored continuously. Gamma analysis was conducted monthly on a composite of weekly samples from JN-1. MDAs were given as 3 x 10^{-16} µCi/mL for gross alpha; 2 × 10^{-16} µCi/mL for gross beta, ^{131}I, and ^{239}Pu; and 1 × 10^{-10} µCi/mL for ^{85}Kr.

The year 1975 represents a snapshot in time of West Jefferson operations in relation to air emissions. During 1975 the BRR (JN-3) was dismantled without incident after terminating operations December 31, 1974[2] (BCL 1987). The number of air emission points changed over subsequent years as facilities and operations were changed or retired. By 1980, for example, stack 005 no longer appeared for JN-2, and stacks 013 and 014 were added to JN-1 (BCL 1981a). By 1986 stacks 007, 008, and 009 were gone, while stack 012 had been added to JN-2 (BCL 1987).

From 1973 to 2005 a total of 4.7 × 10^{6} Bq (0.13 mCi) of alpha emitters other than ^{238}U and 4.1 × 10^{7} Bq (1.1 mCi) of MFPs were released. This release includes 32 of the 33 years during this period because data were not available for 1999. Antimony-125 (29%), ^{60}Co (21%), ^{137}Cs (15%) and ^{152}Eu (9%) accounted for 73% of released MFP activity. Iodine-131 accounted for nearly 8% of released activity with releases in 1974 to 1977 and 1980, with over 90% of the release in 1974 alone. For alpha emitters, ^{241}Am accounts for 50% of the activity (no activity before 1988) and ^{234}U accounts for 44% of the activity with the remainder due to ^{239}Pu. From 1973 until 1988 (when ^{241}Am began to be reported) the ^{235}U and ^{239}Pu activity was approximately equal. After that ^{235}U and ^{241}Am activity was

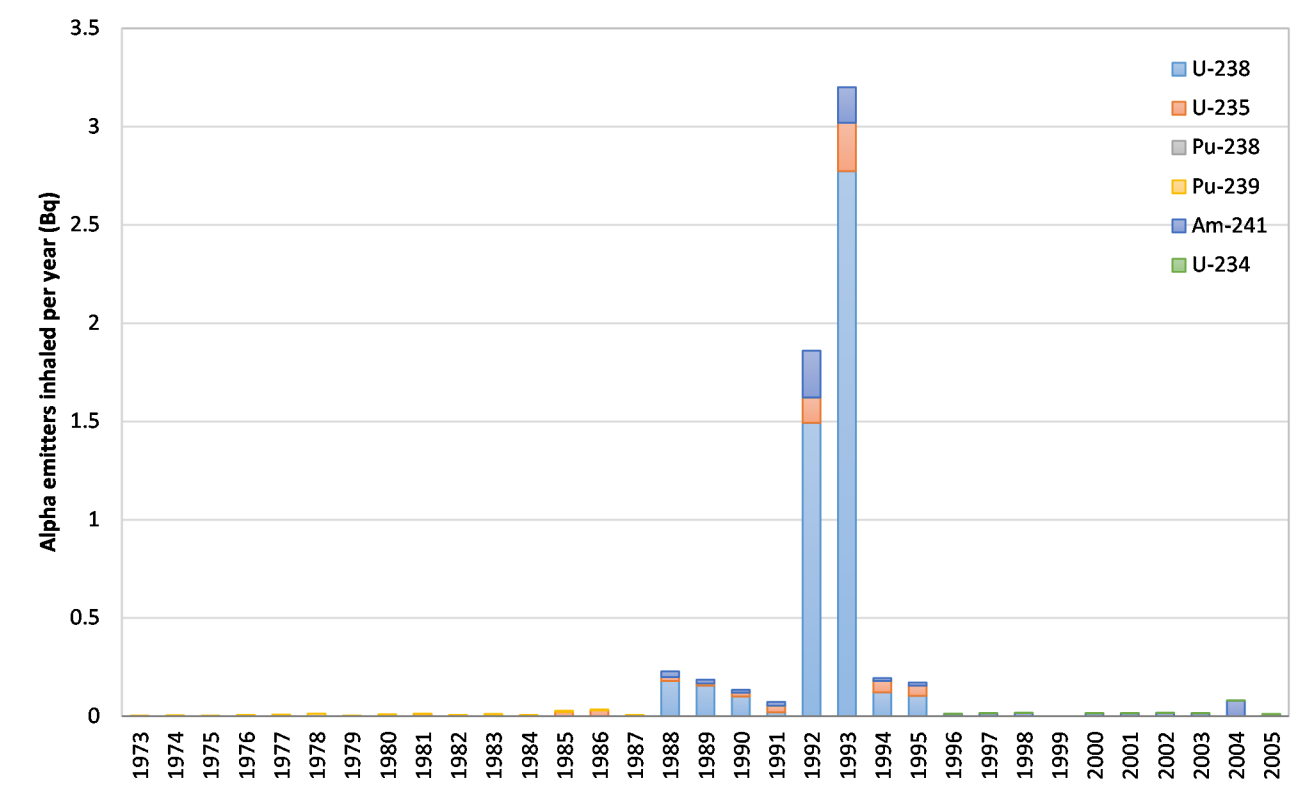

Figure 6-1. Occupational environmental inhalation of alpha emitters at West Jefferson.

2. However, neither the 1973 (BCL 1974) nor 1974 (BCL 1975) annual environmental reports show an air emission point for the operating research reactor.

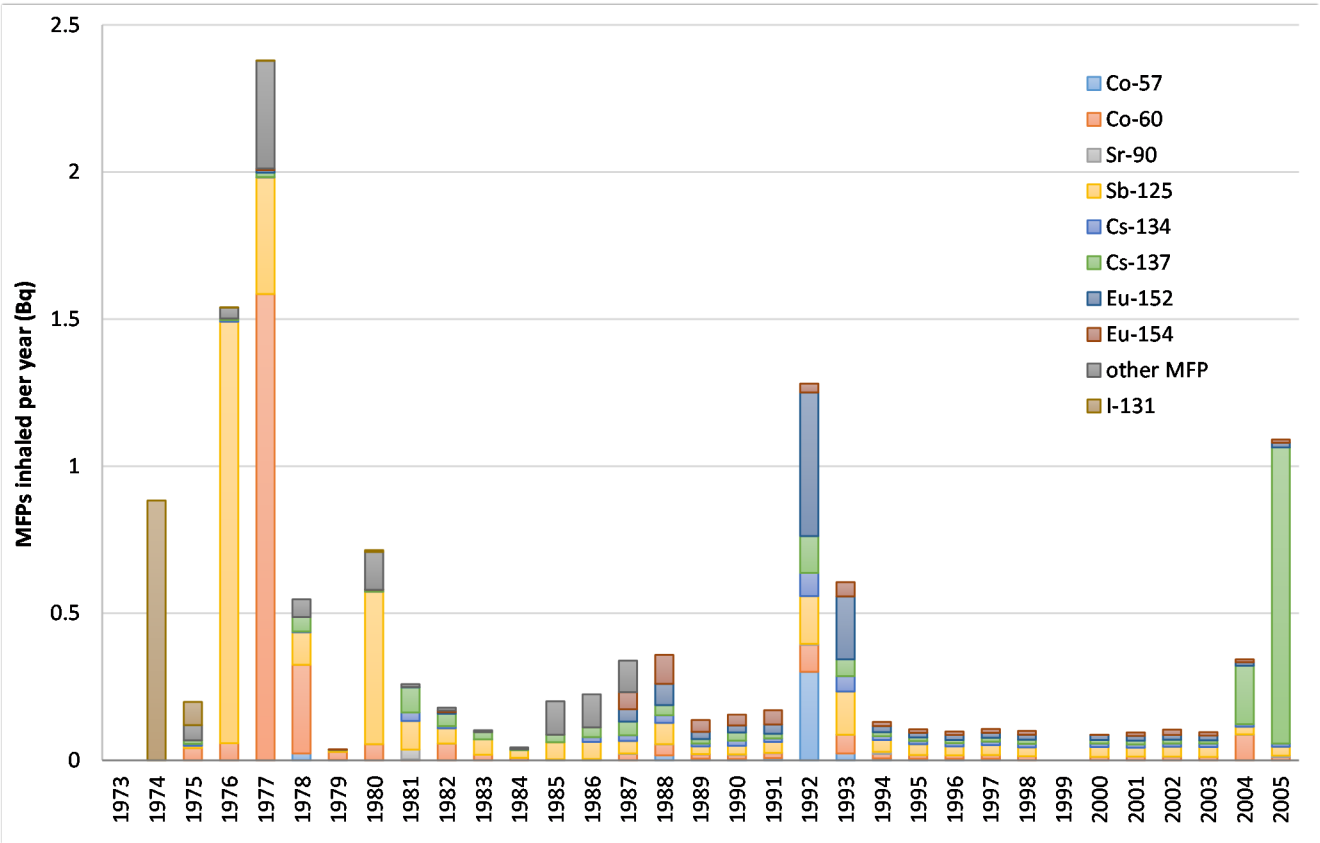

Figure 6-2. Occupational environmental inhalation of MFPs at West Jefferson.

Table 6-1. Air effluent release points at the West Jefferson site in 1975.

Release point	Exhaust stack description	Analysis[a]
001	JN-1 Hot Cell (old building)	Gross alpha, gross beta, gamma, ^{85}Kr
002	JN-1 Hot Cell (new building)	Gross alpha, gross beta, gamma, ^{131}I
003	JN-1 Hot Cell (control area)	Gross alpha, gross beta, gamma
004	JN-1 Hot Cell (liquid waste evaporator)	Gross alpha, gross beta, gamma
005	JN-2 Plutonium Laboratory	^{239}Pu
006	JN-2 Vault	^{239}Pu
007	JN-4 Old Plutonium Laboratory (east)	^{239}Pu
008	JN-4 Old Plutonium Laboratory (west)	^{239}Pu
009	JN-4 New Plutonium Laboratory	^{239}Pu

a. Weekly for particulates, monthly composite gamma analysis, monthly ^{131}I, continuous ^{85}Kr.

approximately equal while ^{239}Pu activity was much smaller. Uranium-238 was not included because it was not reported until 1988 when most of the BCL activities were already complete; this likely represents naturally occurring radioactivity.

6.2 ESTIMATED ANNUAL INTAKES FROM AIRBORNE RADIONUCLIDES

Because the site boundary location is somewhat distant from the West Jefferson facilities (500 m) the site boundary concentrations reported in the annual environmental reports are not representative of onsite concentrations near the buildings (see the Environmental Reports section at the end of the References section). For this site profile, onsite concentrations were calculated using the activities released per year (sum from all stacks) and near-field ground-level concentration screening method from NCRP Report 123 (NCRP 1996). The details of the method are included in Attachment G. This calculation resulted in the derivation of an onsite atmospheric dispersion factor of 4×10^{-3} s/m^3, which

compares reasonably and conservatively with site boundary atmospheric dispersion factors reported to be around 3×10^{-5} s/m^3 for the maximally exposed offsite individual at 500 m (BCL 1974, 1987). This conservative, generic atmospheric dispersion factor can also be used for King Avenue facilities.

Workers are assumed to be on site and potentially exposed for 2,000 h/yr. Workers are also assumed to breathe at a rate characteristic of "light activity" as described in ICRP Publication 66 (ICRP 1994). This rate is 1.2 m^3/h.

King Avenue
Using the reported maximum annual releases from Section 6.1.2 above and applying an onsite atmospheric dispersion factor of 4×10^{-3} s/m^3 derived in Attachment G, a breathing rate of 1.2 m^3/h, and an exposure time of 2,000 h/yr, annual intakes of 3.59×10^{-4} Bq/y for ^{234}U and 2.78×10^{-2} Bq/y for ^{137}Cs are calculated. At these levels of intake the dose to any organ is less than 0.001 rem/yr. Therefore, occupational environmental intakes of inhaled radionuclides do not need to be considered for King Avenue facilities for the period from 1973 through 1975. However, intake values for the West Jefferson site should be applied for King Avenue site workers who might have traveled between the two sites during this period.

West Jefferson
Occupational environmental inhalation of alpha-emitting radionuclides is dominated by ^{239}Pu, ^{241}Am, and ^{235}U at various times during the West Jefferson operating history. Annual intakes of alpha-emitting radionuclides are shown in Figure 6-1. Based on this information, dose reconstructors should apply the annual intakes for all detected radionuclides shown in Table G-1 of Attachment G. Prior to the period represented in Table G-1 for which individual alpha-emitting radionuclides are reported (beginning in 1973), it is assumed that highest reported intake values from 1993 conservatively represent prior operations for 1971 and 1972.

Inhalation of MFPs varied over the operational period at West Jefferson. Annual intakes of MFPs are shown in Figure 6-2. As noted above the main constituents are ^{125}Sb, ^{60}Co, and ^{137}Cs. Based on this information, dose reconstructors should apply the annual intakes for all detected radionuclides shown in Table G-2 of Attachment G. Prior to the period represented in Table G-2 for which individual fission and activation product radionuclides are reported (beginning in 1975), it is assumed that highest reported intake values from 1977 conservatively represent prior operations from 1971 through 1974.

The potential organ dose from two other MFPs that were released at West Jefferson was evaluated individually because these are not well represented by particulate MFPs. Iodine-131 is volatile and inhalation results primarily in dose to the thyroid, while ^{85}Kr is a noble gas and external dose results from immersion in a plume of the gas. At the levels of release indicated for West Jefferson the potential organ doses from both radionuclides would be less than 0.001 rem/yr. Therefore, these radionuclides do not need to be considered separately from MFPs.

Summary
Table 6-2 summarizes the intakes dose reconstructors should use for occupational environmental inhalation of radionuclides at the King Avenue and West Jefferson sites of the Battelle Columbus Laboratories. Values for the West Jefferson site should be applied for King Avenue site workers who might have traveled between the two sites. Intakes should be applied as lognormal distributions with a geometric standard deviation of 3.

Table 6-2. Occupational environmental inhalation of radionuclides at BCL.

Site	Period	Radionuclides	Intake (Bq/yr)	Distribution
King Avenue	1973–1975	U-234, Cs-137	Negligible[a]	Not applicable
West Jefferson	1971–1972	Alpha emitters	Apply Table G-1, 1993 intakes for all years	Lognormal
West Jefferson	1973–Present	Alpha emitters	Table G-1	Lognormal
West Jefferson	1971–1974	Fission/activation products	Apply Table G-2, 1977 intakes for all years	Lognormal
West Jefferson	1975–Present	Fission/activation products	Table G-2	Lognormal

a. Results in annual doses <0.001 rem.

7.0 RESIDUAL EXPOSURE

Based on the *Report on Residual Radioactive and Beryllium Contamination at Atomic Weapons Employer Facilities and Beryllium Vendor Facilities* (NIOSH 2011b), the West Jefferson site had the potential for residual contamination from 1976 to 1985 after the period in which weapons-related production occurred and the King Avenue site had the potential for residual contamination from 2001 through the present.

Individuals who worked at the West Jefferson and King Avenue sites in radiological areas were monitored for internal and external radiation exposure during the period of potential residual contamination. The use of the existing monitoring records in the performance of dose reconstructions is sufficient to account for internal and external doses during this period. No additional dose assessment is necessary to complete dose reconstructions during the West Jefferson residual contamination period from 1976 to 1985 or the King Avenue residual contamination period from 2001 through the present [3].

8.0 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 in the Attributions and Annotations section, with information to identify the source and justification for each associated item. Conventional References, which are provided in the next section of this document, link data, quotations, and other information to documents available for review on the Project's Site Research Database (SRDB).

[1] King, Vincent. Oak Ridge Associated Universities (ORAU) Team. Dose Reconstructor. May 2007.
The FP MDA for 1962 through 1991 was determined empirically by Thomas La Bone using information in available analytical records from Battelle.

[2] Robinson, Sallie. ORAU Team. Dose Reconstructor. June 2007.
The conclusions in this paragraph were made based on a review of the records provided for Battelle Columbus.

[3] Robinson, Sallie. ORAU Team. Dose Reconstructor. June 2007.
The conclusion not to add additional dose for residual contamination is based on the Battelle Columbus records that indicate the workers who had the potential for radiological dose were monitored.

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GLOSSARY

23 recovery program
Program to recover thorium from various materials (e.g., monazite sands) using anion exchange in a carbonate solution.

25 Area
Uranium receiving and dispensing area in Building 3 (Materials).

Atomic Science Center
See Nuclear Sciences Area.

berry bucket or can
One-gallon can for holding an irradiation or other sample or waste in a hot cell during or after an experiment.

coffee cup
Foam coffee cup into which dissolved slugs in liquid form were poured; concrete was added to solidify the liquid in the cup.

Contractor Pool
The instrument and experiment end of the Battelle Research Reactor pool, also called the Shielding Pool.

Davison
Predecessor or subsidiary of W. R. Grace and Company.

Engineering Area
Auxiliary (south) part of West Jefferson site, including Buildings JS-1 (Hot Isostatic Processing Facility for fabricating military reactor fuel) and JS-10 and JS-12 (studies of explosive forming and bonding techniques, ballistic studies using nuclear materials).

fissium
Simulated fission-product-containing material as a mixture of nonradioactive isotopes of those elements that form as radioactive fission products for study of chemical and physical behavior of fission-product-bearing material.

junior cave
Medium-shielded room similar to a hot cell for handling or storage of lower level sources than a fully shielded hot cell or high-level cave or vault would be.

Lamson tube
Pneumatic tube for moving items between the Low Level Cell and Alpha-Gamma Cell 4 (JN-1).

Metmount
Fuel rod section specimen set in 1.75-inch frame to prepare it for metallurgical examination. Also the frame itself.

Myrnalloy
Thorium metal or thorium alloy.

New Plutonium Laboratory
Part of Building JN-4.

Nuclear Research Center
See Nuclear Sciences Area.

Nuclear Sciences Area
Main (north) part of the West Jefferson site, including Buildings JN-1 (Hot Cell Laboratory), JN-2 (criticality assembly, adjunct Plutonium Laboratory), JN-3 (Battelle Research Reactor), and JN-4 (main Plutonium Laboratory).

Old Plutonium Laboratory
Original section of Building JN-4.

reactor pool
The core end of the Battelle Research Reactor pool.

sheep shed
Waste Storage Shed (associated with JN-1).

slug
At Battelle, burnup analysis specimen waste, after being dissolved, studied, mixed with cement, and allowed to solidify in a foam cup.

soda salt
Sodium uranate, $Na_2U_2O_7$.

target room
A target fabrication area at the King Avenue site.

test cell
Hole in the reactor core to insert test specimens (not a hot cell).

Tu-Be alloy
Uranium-beryllium alloy (Tu was synonymous with uranium).

Tuballoy, tuballoy
Uranium, uranium alloy, or Uranium-beryllium alloy (Tu was synonymous with uranium).

weasel
Special type of sample capsule, or the Battelle Research Reactor fast-flux irradiation facility in which it was used. The capsule held an aluminum container that had about 5 grams of powdered uranium dioxide ($UO_{2)}$ packed into quartz tubes. The facility was hydraulically operated and could be positioned in various regions near the core.

West Jefferson North
See Nuclear Sciences Area.

West Jefferson South
See Engineering Area.

White Cross
Hospital in Columbus used by Battelle.

ATTACHMENT A
CHRONOLOGY OF EVENTS

Table A-1. Chronology of events.

Event	Period/ start date	Description	Reference[a]
Initial work for U.S. Office of Scientific Research and Development under Contract No. OEMsr-85	1942	Rolling; metallurgical studies of gas evolution; consulting on extrusion	Smith et al. 1988; BCL undated a
Experimental and pilot work for MED begins under Contract No. W-7405-ENG-92	16 April 1943–1944	Done at irregular intervals, mostly on U metal: analysis, extrusion into rods, hammering, rolling of billets, dipping, machining, plating. Some studies of uranates, U carbides	Author unknown 1992; Ferry 1943; Langsam and Carter 1986; BMI 1945a
R&D and other work with Th metal and alloys (except extraction)	1944–1954	Mostly R&D (1944–1946); forming ingots into Hanford slugs (1947); hydriding (1951); electroplating metals on Th (1948–1953); arc-welding Ames Th, alloys; iodide process (1951–1954); bomb reduction of Th halides	BMI 1945a, 1945b; Udy and Boulger 1951; Monroe, Martin, and Voldrich 1952; Schwope, Muehlenkamp, and Marsh 1952; Veigel, Sherwood, and Campbell 1953; Beach et al. 1953; Goldhoff, Ogden, and Jaffee 1953; Berry, Pray, and Peoples 1954; Saller, Dickerson, and Foster 1955
R&D and pilot programs to extract uranium from various materials	1946–1954	Extraction from phosphate rock (Sep 1946–1950), shales (Sep 1946–1949, 1952–1954). Mechanical beneficiation; pyro-, electro-, and hydrometallurgical methods; shotgun assays	Beyer 1948a, 1948b; Kelley 1948a; BCL undated a
Studies of U compounds and alloys, including alloying Be with U; start of work with UO_2	1947 on	Properties of U-Be compounds and alloys; graphite-U fuel rods, rolling techniques for Al-U alloys. In 1947, listed as “Tu-Be alloy” and “TuO2+BeO”	AEC 1947, p. 7; Grenell 1947; Randall 1947; Keeler and Hare 1954; Gallagher, Blosser, and Mann 1955, Vaughan, Bridge, and Schwartz 1957; Jankowski and Chastain 1958; Cunningham, Seaver, and Edgar 1959; Egen et al. 1960
Research on graphite-U piles	1947–?	For the reactor at ORNL	Author unknown ca. 1987; AEC 1947, p. 7 ; Boehm and Groner 1986
Process development program for extraction of uranium from various uranium ores and concentrates	Aug 1947	Carnotite ores (Aug 1947–1949); washer slimes/slurries (1950); Western ores (1952–1955); low-grade sulfide concentrate (1953); U ores (ammonium carbonate leaching) (1955–1957)	Kelley 1947a, 1947b, 1948b; Blatz 1949; Ewing et al. 1953; Bearse et al. 1953
AEC demonstrates use of radiation detectors	Dec 1947	To check U, Th metal, surfaces, clothing, etc. Battelle did not have such instruments before this date.	Morgan 1947

ATTACHMENT A
CHRONOLOGY OF EVENTS (continued)

Event	Period/ start date	Description	Reference[a]
AEC contract for experimental studies on U- and Th-bearing materials, including monazite sands	1947–1950	Contract AT-30-1 GEN-228; laboratory-level work started in December 1947	Brown 1951; Kelley 1950, 1951; Blatz 1949; Harris 1952; Kelley 1948b, 1948c
Work for the Air Force's Aircraft Nuclear Propulsion Program	1947–1961	Work for the Air Force's Aircraft Nuclear Propulsion Program	Peters 2001; BCL undated a
Process development program for recovering U from phosphate materials	Jan 1948–1949(?)	Development of practical processes for leaching and recovery of U	Beyer 1948a
Active AEC-Battelle contracts for U, Th extraction from monazite sands, phosphates, shales, and carnotite ore	1949–1950	All ongoing as of Apr 1949: Contract GEN-228: extraction of U, Th from monazite sands; ENG-27: U from phosphate rock; GEN-202: U from shales; GEN-258, U from carnotite	Blatz 1949; Calkins et al. 1950; Calkins and Filbert 1950; Wilson et al. 1954
Nuclear submarine fuel element prototype development	1948	Nuclear submarine fuel element prototype development	BCL undated a
Conversion of various forms to UO_2	1950	Conversion of pitchblende to UO_2 (1950–1951); refining MgX (concentrate), V-20 soda salt to UO_2 (1950–1951)	Wesner et al. 1950; Langston, Tangel, and Richardson 1950; Ewing, Kiehl, and Bearse 1950; Ewing et al. 1950a
R&D of large-batch reduction	1950–1955	Mostly defense-related or reactor development work; some processing done for government production reactors	BMI ca. 1987
R&D and studies of U metal, its alloys, U carbides, U nitrides, etc.	1950–1962	Included making and testing fuel forms	Schwope, Muehlenkamp, and Marsh 1952
Battelle said to have provided 10 kg of uranium to Oak Ridge firm as ingots	1951	Presumably for Aircraft Nuclear Propulsion Program because firm was a major contractor for that program	Author unknown ca. 1987
Laboratory, pilot plant study of Th extraction from mantle-grade Th nitrate tetrahydrate $Th(NO_3)_4$	Aug 1951–Feb 1952	Solvent extraction pilot plant design by Catalytic Construction Co; pilot plant started in Jan 1952	Kelley 1951; Brown 1951; Harris 1952; Ewing at al. 1952
Completion of tracer studies for a Ce-Pr mixture	1 Nov 1951	In connection with studies of Th extraction from Th nitrate	Kelley 1951

ATTACHMENT A
CHRONOLOGY OF EVENTS (continued)

Event	Period/ start date	Description	Reference[a]
Process support for U refining, metal production	1952–1959	For AEC, Mallinckrodt, NLO, mostly R&D, process studies	BMI ca. 1987; Author unknown ca. 1975; Ewing, Kiehl, and Bearse 1950; Schwartz and Vaughan 1953; Rengstorff and Lownie 1955
Study of fused salt mixtures as liquid fuel	1953–1954	U salts	Crooks, Snyder, and Clegg 1953; Droege, Snyder, and Filbert 1954
Studies of radioisotope tracers	1954	For industrial control	Kemp et al. 1955
Apparent first use of EU at Battelle	1947	Development of EU fuel elements and assemblies	Saller 1947
King Avenue Buildings 2 through 7 and 10 through 13 constructed	Completed 1954–1955	Buildings A and 1 were already in existence	BMI 1977
Extraction of Th from Brazilian monazite sludge	1954–1955	Extraction of Th from Brazilian monazite sludge	Wallo 1981; Meeley, Snyder, and Filbert 1954
Development of APPR	1954–late 1960s	Development of APPR	Peters 2001; Gallagher, Blosser, and Mann 1955; Cunningham, Seaver, and Edgar 1959; Peters and Harrison 1998
Extraction of U from various ores and concentrates	1950–1957	Washer slimes/slurries (1950); Western ores (1952–1955); low-grade sulfide concentrate (1953); other U ores (ammonium carbonate leaching) (1955–1957)	Meaders et al. 1950; Ewing et al. 1950b; Bearse et al. 1953; Ewing, Kiehl, and Bearse 1955; Van Cleek, Macdonald, and Stephens 1956; Langston, Macdonald, and Stephens 1957
Start of operation of JN-1	1955	Start of operation of JN-1	BCL ca. 1974; BCL undated a, DOE 2003a
Start of operation of JN-2	1955	Started as a critical assembly facility	BMI ca. 1987; DOE 2000a; Layendecker 1996; Peters 2001; Jensen 2003; BCL undated a; Kirsch 2000a
Experiments, other work in JN-1	1955–1960	Mostly small-scale experiments on irradiated capsules	BMI ca. 1987; BCL ca. 1974
Work at the critical assembly facility (JN-2)	1955–1963	Critical assembly and vault work only	Peters 2001; DOE 2003b; Gallagher, Blosser, and Mann 1955; BCL undated a; Kirsch 2000a
Continued work in defense	1955–1965	More Naval Reactors research; work on water-cooled reactor and the Aircraft Nuclear Propulsion Program	BMI ca. 1987; BCL 1997a
Subcontract weapons research for Sandia, etc.	1955–1965	Also for Los Alamos, Rocky Flats, LLNL	BMI ca. 1987
Postirradiation examination, testing of materials in JN-1	1955–1983	Including materials from BRR, other research, defense, and commercial power reactors	None

ATTACHMENT A
CHRONOLOGY OF EVENTS (continued)

Event	Period/ start date	Description	Reference[a]
Startup of JN-3 (BRR)	Oct 1956	Startup of JN-3 (BRR)	DOE 2003a; Peters 2001; Anno, Plummer, and Chastain 1958; Plummer, Anno, and Chastain 1960; BCL undated a; Kirsch 2000b
BRR, ETR loop program work in JN-1 and JN-3	1956–1962	Including assay of fission gases during irradiation of fuel, other materials; BRR upgrade to 2 MW, in-pile loop in 1959	Bodnar et al. 1958; Basham and Rieder 1960
Miscellaneous instrument/detector/dosimetry work	1956–1967	E.g., reactor neutron flux; leak detection; gamma, neutron effects on semiconductors; gamma, neutron dosimeters	Diethorne at al. 1958; Smith et al. 1956; Klickman et al. 1956; Moody, Kendall, and Willardson 1958; Howes, Ellerman, and Sunderman 1960; Hedden, Kircher, and King 1960; Farkas 1966; Kramer, Closser, and Mengali 1966; Kramer 1967
Evaluation and assays for recovery processes for spent fuel elements	1957–1964	Zircex, Darex, Thorex, Zirflex, fluoride volatility, Niflex processes	Ewing, Brugger, and Sunderman 1961; Peterson et al. 1959
BRR upgraded from 1 to 2 MW	1958		Anno, Plummer, and Chastain 1958
R&D on methods of measuring content in industrial materials, rates of biological/catalytic processes	1958–1963	E.g., Mg, Al, Ca, Fe in cement; investigated P-32 labeling, various radiotracers, activation analysis	Calkins and Pobereskin 1954; Dayton and Dickerson 1962; McFarling and Kircher 1963
Defense work in JN-1and JS-1 (Hot Isostatic Process Facility)	1958–1984	Naval reactors, R&D for Army and other defense programs	Peters 2001
BRR upgraded from 1 to 2 MW	Mar 1959	BRR upgraded from 1 to 2 MW	Plummer, Anno, and Chastain 1960; Anno, Plummer, and Chastain 1958
BRR beam tube shut down for furnace repair	Jun–Dec 1960	BRR beam tube shut down for furnace repair	None
Construction of Building JN-4 (old Plutonium Laboratory)	1960	Construction of Building JN-4 (old Plutonium Laboratory)	Peters 2001; Flynn et al. 1987
R&D: coating of U oxide, new fuel forms and materials, containment methods	1960s	R&D: coating of U oxide, new fuel forms and materials, containment methods	Author unknown ca. 1987; Chastain 1961
AEC license to fabricate U fuel elements	1960s	—	Author unknown ca. 1987

ATTACHMENT A
CHRONOLOGY OF EVENTS (continued)

Event	Period/ start date	Description	Reference[a]
Studies involving Pu properties, processing, and fabrication; Np target rod fabrication for Pu production; fabrication of RTGs using Pu and Cm	1960–1977	U recovery process for Pu spent fuel; Ta corrosion by liquid Pu alloys; UN-PuN fuel materials, cladding for fast reactors; PuO2 and other fuel studies; Pu, Np recovery from commercially irradiated targets; Pu aerosols; fabrication of RTGs with Pu-238 (MHW) or Cm-244	Author unknown ca. 1987; DOE 2000b; Swigert 1963; Taylor 1983
Addition of alpha-gamma test cells to JN-1	1964	Addition of alpha-gamma test cells to JN-1	Peters 2001; Martin, Storhok, and Gates 1966; BCL ca. 1974
Addition to JN-4	1964	Addition to JN-4	Flynn at al. 1987
RTG work with Pm sources	1964–1968	RTG work with Pm sources	Ritzman et al. 1966
King Avenue: continued defense program work	1965–1975	Included fabrication of HEU fuel elements	BMI ca. 1987
West Jefferson site: materials mockup experiments, advanced fuels research	1965–1975	The advanced fuels work was done with U nitrides and oxides.	BMI ca. 1987
Further addition to JN-4	1967	Further addition to JN-4	Flynn at al. 1987
Addition of Mechanical Test Cell to JN-1	Apr 1967	Addition of Mechanical Test Cell to JN-1	Peters 2001, BCL ca. 1974
Research on bioeffects of underground explosions at Amchitka Island, Alaska	1967–1974	During/after the Milrow and Cannikin tests	Kirkwood 1971, 1974; Burgner, Isakson, and Lebednik 1971; Stephan and Mercier 1972; Williamson and White 1974
JN-2 converted to a Plutonium Laboratory	Late 1960s–1970	Work done for LLNL; handled encapsulated Pu only. Vault still used	Peters 2001; Kirsch 2000a
Work for NLO under prime AEC contract	~1970?	Involved test quantities of radioactive metal	Author unknown ca. 1975
End of active nuclear work in JN-2	1970	—	Peters 2001
Added High-Energy Cell to JN-1	1972–1973	Allowed for examination of full-sized fuel assemblies	Peters 2001; BCL ca. 1974, 1997a
Neutron radiography of reactor fuel	1971–1975	Done in reactor pool and an out-of-pool facility	Ray 1972
JN-2 converted to a Plutonium Laboratory	Late 1960s–1970	Work done for LLNL; handled encapsulated Pu only. Vault still used	Peters 2001; Kirsch 2000a

ATTACHMENT A
CHRONOLOGY OF EVENTS (continued)

Event	Period/ start date	Description	Reference[a]
Production of Cf-252 sources	1972–1980s?	Sources made from Cf wire in two alpha-gamma cells; unclear where wire came from or for whom work was done	Peters 2001; Scotti and Martin 1972
Final shutdown of the Battelle Research Reactor	1974	Final shutdown of the Battelle Research Reactor	DOE 2003a; Layendecker 1996; BCL undated a; Kirsch 2000b
Added fuel pool, Washdown Room to JN-1	1975	Associated with the High Energy Cell	BCL 1997a
Partial decommissioning of the BRR to NRC criteria, with final survey	1975	Partial decommissioning of the BRR to NRC criteria, with final survey	Rubadue 2000; DOE 2003a, 2003b; Layendecker 1996; Peters 2001, BCL undated a; Kirsch 1975, 2000b; Kok 1975a
Removal of last reactor fuel elements from BRR (JN-3), shipment to Savannah River Plant	March 1975	Removal of last reactor fuel elements from BRR (JN-3), shipment to Savannah River Plant	Layendecker 1996; Kok 1975b
Decontamination of CAL/Pu Laboratory (JN-2)	1975	Decontamination of CAL/Pu Laboratory (JN-2)	Peters 2001; BCL undated a
Work for various defense programs, Naval Reactors, DOD (Army, Air Force), and NASA	1975–1987	Source materials processed and fabricated for DOD activities	BMI ca. 1987
Visit by AEC, ANL personnel to assess the need for a full survey and decontamination	Jan 1976	Visit by AEC, ANL personnel to assess the need for a full survey and decontamination	DOE ca. 1987
Battelle radiological survey of King Avenue facilities used for MED projects	1977	Found to be at background levels, except for Buildings A and 1	DOE ca. 1987, DOE 2000c
Decontamination of JN-4 and Buildings A and 1	1978–1982	Battelle itself performed the JN-4 decontamination, using contractors for specialty work.	DOE ca. 1987, DOE 2000b; Wissinger 1978; Freas and Madia 1982; Rudolph, Kirsch, and Toy 1984; Clements 1981; Stellrecht, Freas, and Dettorre 1979; Madia et al. 1979; Flynn at al. 1987; BCL ca. 1982
Cask sabotage program work in JN-1	1979–1983	In the pool and the Controlled Area; Myers et al. (1994) says 1981–1983.	Peters 2001; Author unknown 2003; Myers et al. 1994a; BCL 1997a
Further radiation surveys	1980–1982	Probably performed by Battelle	DOE 2000b
Research on destruction of chemical agents using high-level gamma fields	1981–1982	JN-1: irradiating capsules of chemical agents using Co-60 sources, using a closed beam port (not in the hot cells)	Peters 2001

ATTACHMENT A
CHRONOLOGY OF EVENTS (continued)

Event	Period/ start date	Description	Reference[a]
Agreement to form the BCLDP	1984	Apparently not formally implemented until 1986	Rubadue 2000
End of nuclear research for DOE; formation of CEMP; formal start of BCLDP	1986	CEMP	2003a; Langsam and Carter 1986
Removal of last reactor fuel elements from JN-1, shipment to INL	1986	Peters (2001) says 1987. Actual removal was apparently in 1986, license termination in 1987.	Peters 2001
Decontamination of some King Ave areas by Battelle, S&M program for King Ave/West Jefferson areas with residual contamination	1986–1988	Buildings A and 1 were decontaminated under CEMP.	DOE ca. 1987, DOE 2000c; Langsam and Carter 1986; Author unknown 1992
Termination of reactor license	1987	To possess and store fuel elements and similar materials	Layendecker 1996
End of use of Building JN-1, shipment of last of fuel in JN-1 pool	1987	BCL (undated a) says 1988, but this appears to be in error.	Peters 1998
Preliminary characterization survey of 15 buildings at the King Avenue, West Jefferson sites	1987	By Argonne National Laboratory	Smith et al. 1988
D&D begins under the BCLDP	1989	D&D begins under the BCLDP	Weaver et al. 2003; Peters 2001
Verification surveys of Buildings JS-1, JS-10, and JS-11 by ORAU, with release by DOE/NRC	1989–1990	Verification surveys of Buildings JS-1, JS-10, and JS-11 by ORAU, with release by DOE/NRC	DOE 2003a; ORAU 2006a; Peters 2001
Further King Avenue D&D (by Battelle?)	1990	Done under DOE's Surplus Facilities Management Program	DOE 2000c
Final verification survey of Building 7A	1993	Final verification survey of Building 7A	ORAU 2006a
Final verification survey of Buildings 3 and 4	1995	Final verification survey of Buildings 3 and 4	ORAU 2006a
Final verification survey of Building 5 by ORAU	1996	Final verification survey of Building 5 by ORAU	ORAU 2006a
Final verification survey of Buildings 1, 6, 7 by ORAU	1997	Final verification survey of Buildings 1, 6, 7 by ORAU	ORAU 2006a

ATTACHMENT A
CHRONOLOGY OF EVENTS (continued)

Event	Period/ start date	Description	Reference[a]
Completion of D&D of King Avenue site	1998	Except for Building 2? See 1999 entry below.	Peters 2001
Beginning of D&D of JN-1	1998	Beginning of D&D of JN-1	Weaver et al. 2003
Final verification survey of Building 2 by ORAU	1999	Final verification survey of Building 2 by ORAU	ORAU 2006a
Emptying, removal of the JN-1 alpha-gamma cells	1999–2000	Emptying, removal of the JN-1 alpha-gamma cells	DOE 2003a
Completion of remediation of King Avenue XP pad	Oct 2000	The pad was unrelated to D&D activities (BCLDP), but full RCRA closure was contingent on closure of the pad.	BCL 2002a
Completion of D&D of King Avenue facilities, final verification survey of Building A by ORAU	2000	Completion of D&D of King Avenue facilities, final verification survey of Building A by ORAU	DOE 2003a, ORAU 2006a
Removal of the BRR bioshield	FY 2001	Removal of the BRR bioshield	DOE 2003a
Nuclear support, surveillance/ maintenance activities; some decontamination conducted at West Jefferson	2001	JN-1, JN-2, and JN-3 were involved.	BCL 2002a
Completion of RCRA closure requirements for the King Avenue site	25 Jul 2001?	Completion of RCRA closure requirements for the King Avenue site	BCL 2002a
Surveys of JN-2 and JN-3 by ORAU	2004	Surveys of JN-2 and JN-3 by ORAU	ORAU 2006b

a. In addition to the references listed, information was obtained from site descriptions and abstracts given on the OSTI Web site for various other Battelle reports.

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES

LIST OF TABLES

Table B-1. King Avenue site building descriptions.

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
A		Entrance was through Building 4 or Building 11 (BMI 1977)	Thorium solvent extraction pilot plant (DOE ca. 1987, Jensen 2003), ore processing (including carnotite), metallurgical studies (DOE ca. 1987); encapsulation of HEU for ATR fuel elements (pilot plant) (Jensen 2003)	First part erected in 1929 as "the King Avenue Building" (Boehm and Groner 1986); added onto before World War II (Boehm and Groner 1986); corporate offices (Langsam and Carter 1986, Jensen 2003). The Safety Office (including the health physics base of operations) was here in at least 1962 (Sunderman and Dickerson 1962)	1942–1986
A	E basement, E wing	53 through 57A	Not clear that this area was involved in MED projects; Jensen (2003) lists only the 1st and 4th floors	Creep Laboratory (BMI 1977)	Not applicable
A	Basement, E wing, S end	58A through 58L, 67	Ore processing, monazite sand solvent extraction	After MED/AEC projects, equipment dismantled, removed (BMI 1977). Unoccupied in 1977 (BMI 1977). Was Laser Laboratory for several years. Residual activity found in Room 58 complex in late 1980s (DOE ca. 1987)	Unknown
A	1st floor	Not applicable	Ore processing, monazite sand solvent extraction (Jensen 2003)	Unknown	Unknown
A	S end of 1st floor in E wing	163 through 163F	Upper part of high bay area of 58 complex	Offices	Unknown
A	NW basement of W wing	6, 6A through 6D, 7	Machine shop operations (BMI 1977)	Converted to library facility by 1977 (BMI 1977)	Unknown
A	North basement, S of E-W corridor	17	Mechanical Test area (BMI 1977)	Laboratory record storage	Unknown
A	2nd floor	Not applicable	Laboratory used for tracer work (Sunderman and Dickerson 1962)	Unknown	Unknown

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES (continued)

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
A	N end, 2nd floor, S of E-W corridor	219, 219C, 219D	Metallography sample examination (BMI 1977), storage	Reprint and brochure distribution offices	Unknown
A	4th floor	Not applicable	Pilot plant encapsulation of HEU for ATR fuel elements (Jensen 2003, DOE 1999), presumably in the Encapsulation Laboratory; Nuclear & Flow Systems Shop, Room A-467, supported the Encapsulation Laboratory (Browne 1974). Tracer-level laboratory (<100 µCi) also located on this floor (Sunderman and Dickerson 1962)	Little radioactive material handled (Browne 1974). The cafeteria was located on the 4th floor also (Boehm and Groner 1986)	Unknown
1	Not applicable	Foundry	Foundry (Langsam and Carter 1986, Jensen 2003); melting, cutting, grinding facilities (Jensen 2003). NU, Th extraction (DOE ca. 1987, Jensen 2003); ore beneficiation studies (Jensen 2003)	Constructed in about 1937. Source material fabrication for DOD, 1975 on (BMI ca. 1987)	1942–1986
1	1st floor	(the) Foundry	Uranium processing (BMI 1977; DOE 1999) in the tons (Jensen 2003), ore beneficiation studies (Jensen 2003)	Unknown	Unknown
1	High bay at S end of 2nd floor, adjacent area	1223, 1223A, 1223B	Uranium production from ore (BMI 1977, DOE 1999); beneficiation operations with ores such as carnotite (BMI 1977). Miscellaneous physical handling of materials in 1223; equipment storage in 1223A, miscellaneous laboratory work in 1223B	Room 1223 complex found to contain residual activity in ~1987 (DOE ca. 1987), after cleanup	?–1986
1	Near 2nd floor high bay area	1219 and 1219A	Ore processing	Ore laboratory	Unknown
1	4th floor	Not applicable	Laboratory used for tracer-level (<100 µCi) work (Sunderman and Dickerson 1962)	Unknown	Unknown
2	Not applicable	Metalworking Building (Langsam and Carter 1986, Jensen 2003); “Tisdale Building”	AEC research, including electroplating Hanford reactor slugs (Jensen 2003, DOE 1999); heat treatment, fabrication of U, Th alloys (Jensen 2003); fabrication of U, Th slugs (DOE 1999). Also rolling studies, alloy development, and fuel element fabrication (Jensen 2003).	Metal fabricating machinery (hydraulic presses, rolling mills, furnaces) removed from the building in 1993 (Jensen 2003)	Unknown
2	Not applicable	Metalworking Laboratory	Ore processing (Jensen 2003 says “tons”); metalworking (Jensen 2003)	Unknown	Unknown

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES (continued)

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
2	Not applicable	Welding Laboratory	NU, DU, EU welding and fabrication (Jensen 2003)	Unknown	Unknown
2	2nd floor	Not applicable	Miscellaneous R&D, heat treatment, electroplating (Jensen 2003)	Unknown	1954–1986
3	Not applicable	U-235 Processing Buildings; Materials Building (Langsam and Carter 1986, Jensen 2003)	Research activities using NU, DU, EU (Jensen 2003, DOE 1999). Powder metallurgy, melting, metallographic, ceramics research facilities (Jensen 2003). Miscellaneous R&D: ceramics research; metallographic research with tons of source material (Jensen 2003)	Used as an SNM storage and handling point since early times. Source material fabrication for DOD, 1975 on (BMI ca. 1987).	
3	1st floor	Room 3161 complex, the "U-235 area"; "25 Area/Room"	U-235 processing facility; receiving/distribution point for unirradiated sources, SNM, including enriched material and thorium (BCL 1975, Jensen 2003, Langsam and Carter 1986); also for occasional special samples (Stickel 1993). Pickling of SNM to remove oxide for use in Building 3, other Battelle areas; solidification of pickling solutions by evaporation; salts stored here for later recovery. U reduction by calcining for reprocessing; hydrides, carbides, nitrides also made for use elsewhere at Battelle; packaging, storing of wastes from other areas. Metallography laboratory, ceramics research facility (DOE 1999)	Glovebox, hoods, shower, calciner, cutting wheel, prefilter, HEPA filter. Decommissioning activities completed in 1994, building returned to Battelle	Unknown
3	1st floor	Room 3110	Storage area (AEC 1950)	Unknown	1954–1975
3	2nd floor or S end of 1st floor	Melting Laboratory/ Facility, Room 3178	Melting Laboratory/Facility (Langsam and Carter 1986, BCL 1975, Jensen 2003, Kizer 1971). U-235 materials used, 1954–1975; tons of source material (Jensen 2003). Metallographic research (Jensen 2003)	Had hoods	Unknown
3	N end of basement	Powder Metallurgy Laboratory	Powder Metallurgy Laboratory (Jensen 2003, DOE 1999, BCL 1975). NU storage, source processing (Jensen 2003, BCL 1975)	Unknown	1954–1986

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES (continued)

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
4	1st floor	Radiochemistry (Radioisotope) Laboratory (Langsam and Carter 1986, Jensen 2003, Sunderman and Gates 1965)	Entrance to Building A was through Building 4 (BMI 1977). Radiochemical procedures, radiotracer studies, industrial applications of radioisotopes, neutron dosimetry, other operations with radioactive materials (Sunderman and Gates 1965). Radioisotope Laboratory main facility was divided into low-level, medium-level, and high-level rooms; had fission gas laboratory, special radiochemistry hoods, junior cave, gloveboxes, shielded operating and storage areas, vault, counting room, offices, waste disposal facility (Peters 1998, Sunderman and Dickerson 1962, Sunderman and Gates 1965). Also had Radiography Laboratory with X-ray machine, Cs-137 source (Selander 1958c)	The main facilities in Building 4 and the associated facilities in Buildings A and 1 were together called the "Radioisotope Laboratory" (Sunderman and Gates 1965). Hot material transported in cask by truck between JN-1 and Building 4 (High Level Laboratory in the Radiochemistry Laboratory) (Sunderman and Dickerson 1962). Decommissioning completed in 1994	Unknown
4	2nd, 4th floors	Not applicable	Miscellaneous labs on 2nd, 4th floors, including encapsulation facility for HEU, metallography facilities (Jensen 2003, DOE 1999). Had Tribology Shop in Rooms 4022 and 2023 during some period before 1994	Unknown	1954–1986
5	Not applicable	Machine Shop	Work for AEC/ERDA/DOE programs: machining (Jensen 2003, DOE 1999), grinding (Jensen 2003, DOE 1999), milling (Jensen 2003, DOE 1999). Be machining (Jensen 2003)	Source material fab for DOD, 1975 on (BMI ca. 1987). D&D completed in 1995 (DOE 1999)	Unknown
5	Not applicable	Machine shop (Jensen 2003; DOE 1999; Langsam and Carter 1986)	Machining of DU, NU, and EU and some Th (Jensen 2003)	Part of the floor of the machine shop was replaced at some point to remove contamination	Unknown
5	Not applicable	Coating Laboratory	Coating (of materials) (Jensen 2003)	Unknown	Unknown
5	Miscellaneous R&D on 1st and 2nd floors	Not applicable	Not applicable	Unknown	1954–1986

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES (continued)

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
6	Miscellaneous R&D on ground and 1st floors	(one) Chemistry Building (Langsam and Carter 1986, Jensen 2003)	Analytical chemistry activities in support of the DOE/Navy program, including instrumental analyses (Jensen 2003). Alloy studies, chemical analyses, and Co-60 irradiation (Jensen 2003)	Ground floor decontamination was completed in 1993 (DOE 1999)	Unknown
6	1st floor	Gamma Laboratory	Gamma Facility Co-60 source was 1600 Ci in Feb 1962 (Sunderman and Dickerson 1962) or 2100 Ci in Nov 1965 (Sunderman and Gates 1965). Source located in belowground pool with 13 ft of water over source, stored in lead pig on the floor of the pool (Sunderman and Dickerson 1962, Sunderman and Gates 1965).	Source rods were in use from about November 1959 on (Sunderman and Dickerson 1962)	Ten years during 1954–1986 (?)
7	Miscellaneous R&D on 1st, 2nd, and 3rd floors	(another) Chemistry Building (Langsam and Carter 1986, Jensen 2003)	Analytical chemistry activities in support of DOE/Navy program, including instrumental analyses (Jensen 2003), over a (unspecified) period of ten years (DOE 1999). Analytical chemistry on source materials and corrosion studies (Jensen 2003)	Annex (Building 7A) was decontaminated, released to Battelle in 1993 (DOE 1999); the significant radioactive operation in 7A was a uranium fluoridation pilot plant (Basham 1993)	1954–1986
9	Miscellaneous R&D on ground floor	Mechanical Engineering Building (Langsam and Carter 1986, Jensen 2003)	Research programs for AEC/ERDA/DOE: handling NU and DU uranium, krypton studies (Jensen 2003)	Hoods in laboratory area. The building was decontaminated and released to Battelle in 1991 (DOE 1999)	1954–1986

Table B-2. West Jefferson site building descriptions

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
JN-1	Not applicable	Hot Cell Facility/ Laboratory; JN-1A was the original building, added to in the 1960s; JN-1B was major addition of 1970–1971	Fuel development research support for AEC (Jensen 2003). Spent fuel examination; postirradiation examination, studies of fuel specimens, reactor components, other mats/structures; source encapsulation; shipping container sabotage (BCL 2002a, DOE 1999, Rubadue 2000, Jensen 2003); some criticality experiments (Jensen 2003); later, hot cell examination of fuel from commercial power reactors "in support of DOE programs." All cells had remote manipulation, several operating stations, each with lead-glass oil-filled viewing window (DOE 2003a). Support facilities: cask handling, solid and liquid waste disposal, contamination control, equipment decontamination, ventilation mezzanines (DOE 2003a, 2003b; Norris 1961; Sunderman and Dickerson 1962)	Extensively modified in 1960 and 1979 (DOE 2003b). All cells in JN-1 have very high residual radiation fields (DOE 2000a). Water in the fuel transfer pool was removed in 1996 and packaged TRU waste was then stored there (DOE 2003a). Main offices were on the first floor near the laboratories (Peters 2001).	1955–1986
JN-1	Section JN-1A, 1st floor	Controlled Access Area (CAA), with small spent fuel pool	Support area for hot cell work: provided access to the cells through their doors, facilities for moving large items into cells, and space for special projects (DOE 2003b). Heat treatment furnace in CAA (Selander 1959b); ventilated cubicles for work with small shielded specimens (Sunderman and Dickerson 1962); small hot room with pit, sinks, and hot drain for equipment assembly/decontamination, manipulator repair bench/glovebox (Sunderman and Dickerson 1962, Myers et al. 1994b). Enclosed drum compactor (Myers et al. 1994b). Spent fuel pool used to 1972 (Peters 2001, Myers et al. 1994b). Sabotage Program (1981–1983): apparatus bolted on top of spent fuel pool, shaped charge shot at small model cask; gases collected, analyzed (Peters 2001, Myers et al. 1994b)	Unknown	1955–1986

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES (continued)

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
JN-1	Section JN-1A, ground and 1st floor (cells), basement (subcells)	Two large cells with their subcells: High Level Cell, Low Level Cell	The original two large cells (DOE 2000a, 2003a, 2003b; Langsam and Carter 1986; Rubadue 2000) designed to provide shielding for 10,000 Ci and 10 million Ci of 1-MeV gamma emitter, respectively (Peters 2001, BCL undated b, Sunderman and Dickerson 1962). High-density concrete walls (ferrophosphorus aggregate at 293 lb/ft^2) up to top of the manipulators, ordinary concrete after that (DOE 2003b). Three operating stations each (BCL 1977a, Sunderman and Dickerson 1962). Metmounts cut and ground in the LLC (Peters 2001), which was also used for tensile tests of Co-60 samples (Author unknown 2003). LLC also used near end of operations to gamma-scan cans of waste (Myers at al. 1994c). Subcells used for creep testing (Author unknown 2003, Sunderman and Dickerson 1962)	The LLC was used to declad fuel, section Pu-containing fuel rods, and do tensile tests on irradiated cobalt specimens (Myers at al. 1994c). The High Level Cell's primary purpose was to cut irradiated fuel sections and defuel fuel rod sections (Myers at al. 1994d), i.e., the most destructive activities (BCL 1997a). Irradiated Co-60 rods were also cut up here (Myers at al. 1994d, BCL 1997a)	1955–1986
JN-1	Section JN-1A, 1st floor	Charpy Cell/Room; Dry/Radioactive Storage Area	Shear test area for nonfuel specimens, using Charpy apparatus (Myers, Redd, and Berchtold 1994a); 74 vertical pipe wells in floor, additional 12 capped storage holes in movable concrete block for canned specimens (Peters 2001, DOE 2003b, Sunderman and Dickerson 1962, Myers, Berchtold, and Sands 1995, Myers, Redd, and Berchtold 1994a); radioactive materials stored in steel cans in wells, transported to/from other cells in portable lead shield (Sunderman and Dickerson 1962)	Charpy apparatus removed and sent to King Avenue in 1988–1989 (Myers, Redd, and Berchtold 1994a)	1955–1986
JN-1	Section JN-1A, 1st floor	Transuranic vault	Storage	Unknown	1955–1986
JN-1	Section JN-1A, 1st floor	Evaporator Room	Open evaporator tank with exhaust hood and fill pipe coming from Evaporator Storage Tank; sludge periodically put into drums (Myers, Berchtold, and Tomlin 1994)	Unknown	1961–1986
JN-1	Section JN-1A, 1st floor	Laboratories	Chemistry Laboratory: two hoods, hot sink, hot drain to sump. Counting Room: X-ray diffraction, gamma/alpha spectroscopy, radiochemistry studies; hot sinks, hood, cave, hot drain to sump. Microprobe Room: microprobe equipment, Cs-137 source; isotopic analyses of Metmount samples, using X-ray diffraction. (Peters 2001, Myers et al. 1994e)	Unknown	1964–1986

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES (continued)

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
JN-1	Section JN-1A, 1st floor	East machine shop	(Peters 2001, Myers et al. 1994e)	Unknown	1964–1986
JN-1	Section JN-1A, basement	10 small segmented alpha-gamma cells (Peters 2001, DOE 2003b, Rubadue 2000, DOE 2003a)	Cells 1-9 constructed in 1964 for metallography testing of fuel rod specimens; Cell 10 added in early 1970s (Peters 2001, BCL ca. 1974, Myers et al. 1994f). Shared working/access area (BCL undated b), individual working box in each cell (Myers et al. 1994f). Cells 1, 2, 3, 10 for metallography operations: Metmounts ground down in Cell 3; polished, washed, acid-etched in Cell 2; hardness-tested, photographed with metallograph in Cell 1. Cell 4 used to port materials to/from CAA, other cells; never used for project work. Cells 5, 6 used only to prepare Cf-252 sources, mid-1970s on, Cf wire cut, welded into containers, leak-tested. Cell 7 used for dissolution, slug solidification of burnup fuel specimens. Cell 8: unclad fuel samples heated in furnace, exposed to laser beam/detector for analysis. Cell 9: X-ray diffraction testing of Metmounts (e.g., Myers, Berchtold, and Sands 1995, Myers et al. 1994f, Sunderman and Gates 1965, BCL 1997). Shielding added to wall between working area and evaporator room, 1964 (Gates 1964)	For handling irradiated fuel metallography specimens (BCL undated b, 1997a). When Cell 10 was built, Cell 1 equipment abandoned. Cell 3 was the most contaminated alpha-gamma cell because Metmounts were ground there; two in-cell grinders were the most contaminated equipment (Peters 2001, Myers, Berchtold, and Sands 1995). Some spills in Cell 5 (Myers et al. 1994f). Fumes from acid-fuel solutions in Cell 7 contaminated the rear of Cells 7, 8 via a faulty connection (Peters 2001, Myers, Berchtold, and Sands 1995).	1967–1986
JN-1	Section JN-1A, 1st floor	Mechanical Test Cell	Main mechanical test cell (Rubadue 2000, DOE 2003b, Myers, Redd, and Berchtold 1994b) added in 1967 (BCL ca. 1974). Three operating stations (BCL undated b). High-density concrete front wall(s) (ferrophosphorus aggregate at 293 lb/ft^2) to the top of the manipulators, ordinary concrete above that (DOE 2003b, DeMastry, Lusk, and Gates 1967). Unspecified thickness of iron plate belatedly added to one wall for shielding reasons (DOE 2003b)	Mainly for tensile tests, but also for creep, vacuum fusion, burst, radial burnup, density, and expanding mandrel tests (Myers, Redd, and Berchtold 1994b, BCL 1997a)	1967– 1986
JN-1	Section JN-1B, 1st floor	Cask Washdown Room	For cleaning newly arrived casks, etc. (DOE 2003b, Myers et al. 1994b, Myers, Berchtold, and Stickel 1995); also entry-exit area for High-Energy Cell (DOE 2003b, Myers, Berchtold, and Stickel 1995). Walls open at top for crane access (DOE 2003b, Myers, Berchtold, and Stickel 1995). Compactor under hood (Myers, Berchtold, and Sands 1995)	Unknown	1970–1986

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES (continued)

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
JN-1	Section JN-1B, 1st floor	High Bay/ Operations Floor	Connected to the High Energy Cell, the Fuel Storage Pool, and associated facilities; designed for handling large casks (DOE 2003b)	Unknown	1970–1986
JN-1	Section JN-1B, 1st floor	Fuel Storage Pool	Storage of fuel assemblies, rod bundles, rod holders, tools (DOE 2003b, Myers, Berchtold, and Stickel 1995). Inside and outside sumps to collect any pool leakage (Myers, Berchtold, and Stickel 1995). Cask brought in, lowered into pool; fuel assemblies or bundles taken out, transferred to High Energy Cell through fuel channel running from pool to under the High Energy Cell floor (Jensen 2003)	Unknown	Unknown
JN-1	Section JN-1B, 1st floor	Pump Room	Held wastewater storage tank, deionized water tank, and ion exchange tanks used to process the liquid radioactive waste (Myers, Berchtold, and Sands 1995, Myers, Berchtold, and Stickel 1995)	Unknown	1972–1986
JN-1	Section JN-1B, 1st floor	High Energy Cell	Largest hot cell (DOE 2003b, DOE 2003a), five operating stations (BCL undated b). Power plant fuel element testing, evaluation (DOE 2000a). Access via door, ceiling plugs, or floor trap door to transfer canal (DOE 2003b), then to fuel transfer pool (Rubadue 2000, DOE 2003a). Seven storage holes in floor (DOE 2003b). Two walls barytes (at 220 lb/ft^2), two ordinary concrete (DOE 2003b). Cask washdown room at entryway (BCL undated b, Myers et al. 1994b). Had beam port for Co-60 source inside the cell to irradiate test instruments (Myers et al. 1994b), some storage holes in floor (Myers et al. 1994b)	Designed to handle entire high-burnup fuel assembly (BCL undated b, 1997a); nondestructive inspection and testing (BCL 1997); sectioning of rods (BCL 1997a); fission gas measuring tests done here (Myers et al. 1994b, BCL 1997a); studies and characterization of Three Mile Island resins (BCL 1997a); effects of Co-60 on instrumentation (BCL 1997a)	Unknown
JN-1	Section JN-1B, 1st floor	Old Back Dock, Waste Control Area	In early years, used to collect and package waste and as the original JN-1 dock; later, used for storage of a large number of highly contaminated items (Myers at al. 1994g)	Unknown	1970–1986
JN-1	Section JN-1B, 1st floor	Old Back Dock, Waste Control Area	In early years, used to collect and package waste and as the original JN-1 dock; later, used for storage of a large number of highly contaminated items (Myers at al. 1994g)	Unknown	1970–1986
JN-1	Attached to the JN-1B section	Waste Storage Shed, Sheep Shed, Equipment Storage Room	Storage of low-level contaminated containerized waste, equipment, other material (DOE 2003b, 2003c; Myers and Berchtold 1994). Two internal rooms/areas were heavily shielded (Myers and Berchtold 1994)	Unknown	Unknown

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES (continued)

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
JN-1	With JN-1, but not attached	Waste Management Shed	Storage of highly contaminated containerized waste, equipment, other material (DOE 2003b, 2003c)	Unknown	JN-2: 1955–1963; 1964–1970. Vault to at least 1996
JN-2	Not applicable	CAL, Critical Assembly Facility, Critical Assembly Building; later also a Plutonium Laboratory	Started operation in 1955 (Peters 2001). 1956–1963: critical assembly studies (DOE 2003b, DOE 2000a, Chamberlain ca. 2003, Jankowski et al. 1957, Jankowski and Chastain 1958, Egen at al. 1960); zero-power organic-moderated critical assembly and other experiments (Jensen 2003). 1964(?)–1970, used for other nuclear projects, including direct conversion concepts/Plutonium Laboratory (DOE 2000a, 2003a, 2003b; Jensen 2003; Chamberlain ca. 2003; Kirsch 2000a), irradiation experiment assembly (DOE 2003b, Chamberlain ca. 2003), special laboratory (Jensen 2003). Had Pu, enriched U storage vault (DOE 2000a, 2003a, 2003b; Jensen 2003; Rubadue 2000; Freas et al. 1971; Evans and Woodward 1978; BCL 1977a); control room (Jankowski et al. 1957, Hogan et al. 1958, Jankowski and Chastain 1958); attached high-bay area (DOE 2000a, 2003a; Rubadue 2000), machine shop (Hogan et al. 1958), offices, labs. Also a fence around the building (separate from the site fence) (Hogan et al. 1958)	Built to accommodate zero/low-power mockups of prototype reactors (Kirsch 2000a). Later, a laboratory installed for work with encapsulated Pu for LLNL. 1970: end of active nuclear work (Chamberlain ca. 2003, Peters 2001). 1975: first decontamination (Peters 2001). Was Admin Building, had Accountability Laboratory to ~1987 (Langsam and Carter 1986). RAL in decontaminated former Pu Laboratory, 1978 to ~2003 (DOE 2000b; Rubadue 2000; DOE 1999, 2000a, 2003a, 2003b; Chamberlain ca. 2003; Peters 2001; Jensen 2003); carpenter shop in bay moved from JN-3, from ~1987 on (Langsam and Carter 1986). Vault still used in 1996 or later (DOE 1999). “Minimally contaminated” at final decontamination (DOE 1999, 2000a; Jensen 2003)	1956–1974

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES (continued)

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
JN-3	Not applicable	BRR Building	Pool-type research reactor (Jensen 2003); operation began in October 1956 at 1 MW (Anno, Plummer, and Chastain 1958), was increased to 2 MW in March 1959 (Plummer, Anno, and Chastain 1960), ended in December 1974 (DOE 1999). Shielding facility located at end of thermal column (Morgan et al. 1958, Plummer, Anno, and Chastain 1960, Myers and Stickel 1995); gas loop facility added 1958 (Bodnar et al. 1958; Plummer, Anno, and Chastain 1960, Basham and Rieder 1960, Anno and Plummer 1962); fast neutron activation facility added ~1960 (Jung, Epstein, and Chastain 1960). Carpenter shop (Langsam and Carter 1986); health physics office and instrument laboratory (Langsam and Carter 1986). JN-3 addition: well/hole storage in 144 wells/holes, primarily for spent fuel assemblies (Myers and Stickel 1995)	Partly decommissioned in 1975 (Rubadue 2000, DOE 2003a, Jensen 2003, DOE 1999). Carpenter shop, Health Physics and Instrument Laboratory to ~1987 (Langsam and Carter 1986). 1987 on: temporary storage of TRU wastes (DOE 2003a, Jensen 2003), for which a special waste storage area was built in the basement (Myers and Stickel 1995)	1960–1977 (DOE 2000b, Rudolph, Kirsch, and Toy 1984)
JN-4	Not applicable	Main Plutonium Laboratory ("old" laboratory section and its "new" additions)	Facility used in part for production of U-Pu nitride pellets, which were packed into fuel pins (Peters and Smith 2000). Development of actinide fabrication and joining techniques (Smith 2000). Had Metallography Laboratory (BCL 1977a). Room 4118 was used for fabrication of Pu-238 heat sources (DOE 2000b, Smith 2000). Backup facility for LLNL weapons assembly (Author unknown 2003, Peters and Smith 2000). Some wet chemistry in a glovebox for Allied Chemical Nuclear Services project (Smith 2000); some nuclear weapons, test shot (cask sabotage) projects (Author unknown 2003). Pu shipped to Battelle was handled here (Author unknown 2003, BCL 1977a), initially in the Downdraft Room (BCL 1977a).	Support of Pu research, processing (Peters 2001, Freas et al. 1971, Rudolph, Kirsch, and Toy 1984, BCL 1977a); added to in 1964, 1967 (Freas et al. 1971, Rudolph, Kirsch, and Toy 1984, BCL 1977a). ~50% of work was Pu alloy R&D to support weapons development for LLNL, Rocky Flats, Army reactors (Author unknown 2003, Smith 2000). Two fume hoods; 43 gloveboxes in 1971, 28 in 1978 (Peters 2001, Freas and Madia 1982, Rudolph, Kirsch, and Toy 1984). Presses, furnaces, grinders, lathes (Clements 1981). Old section removed by 1982 (BCL ca. 1982)	1976–present
JN-6	Not applicable	(New) Guardhouse	Security operations (guardhouse)(Rubadue 2000, DOE 2003b)	None	1965–1975?

ATTACHMENT B
BUILDINGS AND OTHER FACILITIES (continued)

Building	Location or subarea	Room, area, or alternate name	Description and MED/AEC use	Later uses and status	Years of operation
JS-1	Not applicable	Not applicable	Used to fabricate military reactor fuel by hot isostatic pressure bonding technique (DOE 1999, Boehm and Groner 1986). Hot Isostatic Processing Facility (BMI ca. 1987, Langsam and Carter 1986); explosive containment (Langsam and Carter 1986); Ballistics Facility (Langsam and Carter 1986)	Decommissioned in 1990 (DOE 1999); D&D was complete by 2003	?–1988
JS-10	Not applicable	Not applicable	Fabrication of uranium components using explosive forming techniques and ballistic experiments with DU (DOE 1999, Boehm and Groner 1986)	Facility characterization in 1998 showed that no D&D was necessary (DOE 1999)	?–1989
JS-12	Not applicable	Not applicable	Operations with radioactive materials in the ballistic tunnel, ductwork, and hardware using DU (DOE 1999, Boehm and Groner 1986)	Decommissioned in 1989 (DOE 1999)	Unknown

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL

LIST OF TABLES

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Table C-1. History of operations at the King Avenue site.

Building[a]	Period	Isotopes[b]	Form[c]	Type(s) of work[d]
A	1943–1979	U, Th, progeny	DU, NU, EU, Th	6.4 MT (Author unknown 1992) (apparently the total ever worked with in this building)
A	1943–1944; 1956–1956	U, progeny	Uranate; U_3O_8; $U_2O_2(NH_4)_2$	Metallurgical R&D, studies of uranates
A	1943–1944; 1950–1952; 1958–1961; 1972–1975	U, progeny	U carbides, mainly UC (NU; EU 9%–93%)	R&D, studies of U carbides and their alloys
A	1943–1962; 1979–1979	U, progeny	U metal, alloys	R&D, studies of work on U metal and its alloys
A	1946–1957	U, progeny	U_3O_8, UO_2, $(NH_4)_2U_2O_7$, $Na_2U_2O_7$, other forms (UO_3?)	U chemical extraction: phosphate rock (1946–1950); shales (1946–1949, 1952–1954); washer slimes/slurries (1950); carnotite ores (1947–1949); Western ores (1952–1955); low-grade sulfide concentrate (1953); U ores (ammonium carbonate leaching) (1955–1957). Converting pitchblende to UO_2; refining MgX (concentrate), V-20 soda salt to UO_2 (1950–1951)
A	1950–1971	U, progeny	UO_2 (NU, DU)	R&D on UO_2 used in fuel of various types (powder, pellet, plate-type, etc.) (a few plates: DU)
A	1953–1954	U, progeny	U salt solutions (see at right)	Study of fused salt mixtures as liquid fuel: U fluorides, phosphates, sulfates, hydrosulfides; solubility of U in fused salts, O_2 in uranyl sulfate solutions
A	1953–1959	U, progeny	U metal, UO_2, UO_3, UNH	Process support for U metal production for Mallinckrodt, NLO
A	1953–1959	U, progeny	UF_4	Process support for U metal production (some work involved UF_4); study of UF_4 as fused salt mixture for liquid fuel (1953–1954)
A	1956–1959	U, progeny	U compounds (see at right)	Study of U compounds: U beryllides, borides, silicides, sulfides
A	1957–1958	U, progeny	UCl	Study of UCl dispersions (short period only)
A	1957–1961	U, progeny	U nitrides, mainly UN (DU; NU; EU 5%–93%)	R&D, studies of U nitrides and their alloys
A	1958–1960	U, progeny	UO_2 (EU, mostly 93%)	R&D on UO_2 for fuel of various types (powder, pellet, plate-type, etc.); enrichment from few percent to full; bulk of work apparently with 93% EU
A	1944–1961	Th, progeny	Th metal, Th alloys	Work with Th compounds, alloys (except extraction): R&D with Th metal (1944–1946); forging, rolling, machining Th ingots into Hanford slugs (1947)
A	1951–1955	Th, progeny	$Th(NO_3)_4$, other forms	Th extraction from monazite sands (1949–1950). Solvent extraction to purify $Th(NO_3)_4$ (1951–1952). Extraction from Brazilian monazite sludge (1954–1955)
1	1943–1962	U, Th	DU, NU, EU, Th	11.6 MT (Author unknown 1992)

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building[a]	Period	Isotopes[b]	Form[c]	Type(s) of work[d]
1	1943–1944; 1950–1952; 1956–1962	U, progeny	U carbides, mainly UC (NU; EU 9%–93%)	R&D, studies of U carbides and their alloys
1	1943–1944; 1956–1956	U, progeny	Uranate; U_3O_8; $U_2O_2(NH_4)_2$	Metallurgical R&D, studies of uranates
1	1943–1962	U, progeny	U metal, alloys	R&D, studies of work on U metal and its alloys
1	1946–1957	U, progeny	U_3O_8, UO_2, $(NH_4)_2U_2O_7$, $Na_2U_2O_7$, other forms (UO_3?)	U chemical extraction from phosphate rock (1946–1950); shales (1946–1949, 1952–1954); washer slimes/slurries (1950); carnotite ores (1948–1949); Western ores (1952–1955); low-grade sulfide concentrate (1953); U ores (ammonium carbonate leaching) (1955–1957). Converting pitchblende to UO_2 (1950–1951); refining MgX (concentrate), V-20 soda salt to UO_2 (1950–1951)
1	1950–1965	U, progeny	UO_2 (NU, DU)	R&D on UO_2 used in fuel of various types (powder, pellet, plate-type, etc.) (a few plates: DU)
1	1952–1958	U, progeny	U metal, UO_2, UO_3, UNH	Process support for U metal production for MED/AEC, Mallinckrodt, NLO
1	1952–1958	U, progeny	UF_4	Process support for U metal production (some work involved UF_4); study of fused salt mixtures as liquid fuel (1953–1954)
1	1953–1954	U, progeny	U salt solutions (see at right)	Study of fused salt mixtures as liquid fuel: U fluorides, phosphates, sulfates, hydrosulfides; solubility of U in fused salts, O_2 in uranyl sulfate solutions
1	1956–1959	U, progeny	U compounds (see at right)	Study of U compounds: U beryllides, borides, silicides, sulfides
1	1956–1962	U, progeny	UO_2 (EU, mostly at 93%)	R&D on UO_2 for fuel of various types (powder, pellet, plate-type, etc.); enrichment from few percent to full; bulk of work apparently with 93% enrichment
1	1957–1961	U, progeny	U nitrides, mainly UN (DU; NU; EU 5%–93%)	R&D, studies of U nitrides and their alloys
1	1944–1961	Th, progeny; minor U	Th metal, Th alloys; Th phosphate, halide, fluorides, carbides; ThN	Th compounds, alloys (except extraction): R&D with Th metal (1944–1946); forming Th ingots into Hanford slugs (1947); Th hydriding (1951); electroplating metals on Th (with Th phosphate) (1948–1953); arc-welding Ames Th, Th alloys; metallurgical/mechanical study of Th; Th prep by iodide process (1951–1954); bomb reduction of Th halides; Th ingot prod by arc melting, casting (with Th fluorides) (e.g., 1955); Th-5, -10, -15, -20 wt % U alloys; Th-U alloy made with thorium iodide; ingots cast by vacuum induction melting to Th buttons, casting Th-U carbides by arc-melting; ThN prep (1956–1961). Dissolution of Al-canned Th (1955)
1	1951–1955	Th, progeny	$Th(NO_3)_4$, other forms	Th extraction from monazite sands (1949–1950). Solvent extraction to purify $Th(NO_3)_4$ (1951–1952). Th extraction from Brazilian monazite sludge (1954–1955)
2	1954–1972	U, Th	DU, NU, EU, Th	32.6 MT (Author unknown 1992)

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building[a]	Period	Isotopes[b]	Form[c]	Type(s) of work[d]
2	1954–1959	U, progeny	U metal, UO_2, UO_3, UNH	Process support for U metal production for Mallinckrodt, NLO
2	1954–1962	U, progeny	UF_4	Process support for U metal production (some work involved UF_4) (1954–1958); prep of UO_2 crystals from UF_4 by vapor deposition (1959–1962)
2	1954–1962	U, progeny	U metal, alloys	R&D, studies of work on U metal and its alloys
2	1954–1971	U, progeny	UO_2 (NU, DU)	R&D on UO_2 used in fuel of various types (powder, pellet, plate-type, etc.) (a few plates: DU)
2	1955–1957	U, progeny	U_3O_8, $(NH_4)_2U_2O_7$, other forms(?)	U extraction from U ores (ammonium carbonate leaching) (1955–1957)
2	1956–1956	U, progeny	$U_2O_2(NH_4)_2$	Study of prep of ammonium uranate
2	1956–1959	U, progeny	U compounds (see at right)	Study of U compounds: U beryllides, borides, silicides, sulfides
2	1956–1962	U, progeny	UO_2 (EU, mostly at 93%)	R&D on UO_2 for fuel of various types (powder, pellet, plate-type, etc.); enrichment from few percent to full; bulk of work apparently with 93% EU
2	1956–1962	U, progeny	U carbides, mainly UC (NU; EU 9%–93%)	R&D, studies of U carbides and their alloys
2	1957–1958; 1961–1961	U, progeny	UCl (EU; NU?); U_3O_8 (NU?)	Study of UCl dispersions (EU, short period only, 1957–1958); making UCl from U_3O_8 (1961)
2	1957–1961	U, progeny	U nitrides, mainly UN (DU; NU; EU 5%–93%)	R&D, studies of U nitrides and their alloys
2	1961–1961	U, progeny	HEU_3O_8, HEU	Prep of Al-U (HEU) fueled glass fiber fuel plates for irradiation
2	1955–1965; 1971–1972	Th, progeny; minor U	Th metal, alloys; iodide, phosphate, halide, nitride, fluoride, carbide;ThO_2 alloys	Th compounds, alloys (except extraction): bomb reduction of Th halides; prod Th ingots by arc melting, casting (with Th fluorides); Th-5, -10, -15, -20 wt % U alloys; Th-U alloy made with Th iodide; ingots cast by vacuum induction melting as Th buttons, casting Th-U carbides by arc-melting; ThN prep (1956–1961); Ta alloy buttons (trace Th) (1959–1960); Th carbide powder prep by vapor deposition (1960); corrosion of Th under storage, Th compounds in NaK (1960–1961); additives to Th-U (1960–1965); ThO_2 alloys with other mats (1971–1972)
2	1956–1956	U content	Fissium	Fissium (“synthetic fissium alloy”): specimen fab, annealed; X-ray diffraction
3	1954–1972	U, Th	DU, NU, EU, Th	39.5 MT (Author unknown 1992)
3	1954–1959	U, progeny	U metal, UO_2, UO_3, UNH	Process support for U metal production for Mallinckrodt, NLO
3	1954–1960; 1979–1979	U, progeny	U metal, alloys	R&D, studies of work on U metal; its alloys, including electroplating; vapor deposition of metals on U
3	1954–1962	U, progeny	UF_4	Process support for U metal production (some work involved UF_4) (1954–1958); prep of UO_2 crystals from UF_4 by vapor deposition (1959–1962)
3	1954–1971	U, progeny	UO_2 (NU, DU)	R&D on UO_2 used in fuel of various types (powder, pellet, plate-type, etc.) (a few plates: DU)

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building[a]	Period	Isotopes[b]	Form[c]	Type(s) of work[d]
3	1956–1959	U, progeny	U compounds (see next column)	Study of U compounds: U beryllides, borides, silicides, sulfides
3	1956–1962	U, progeny	UO_2 (EU, mostly at 93%)	R&D on UO_2 for fuel of various types (powder, pellet, plate-type, etc.); enrichment from few percent to full; bulk of work apparently with 93% enrichment
3	1956–1962	U, progeny	U carbides, mainly UC (NU; EU 9%–93%)	R&D, studies of U carbides and their alloys
3	1957–1961	U, progeny	U nitrides, mainly UN (DU; NU; EU 5%–93%)	R&D, studies of U nitrides and their alloys
3	1957–1958; 1961–1961	U, progeny	UCl (EU; NU?); U_3O_8 (NU?)	Study of UCl dispersions (EU, short period only, 1957–1958); making UCl from U_3O_8 (1961)
3	1955–1965; 1971–1972	Th, progeny; minor U	Th metal, alloys; iodide, phosphate, halide, nitride, fluoride, carbide;ThO_2 alloys	Th compounds, alloys (except extraction): Th ingot prod by arc melting, casting (Th fluorides); Th-5, -10, -15, -20 wt % U alloys; Th-U alloy with Th iodide; ingots cast by vacuum induction melting as Th buttons, casting Th-U carbides by arc-melting; ThN prep (1956–1961); Ta alloy buttons (trace Th) (1959–1960); Th carbide powder prep by vapor deposition (1960); corrosion of Th under storage, Th compounds in NaK (1960–1961); Th-U additives (1960–1965); ThO_2 alloys with other mats (1971–1972)
3	1956–1956	U content	Fissium	Fissium (“synthetic fissium alloy”): specimen fab, annealed; X-ray diffraction
4	1954–1982	U, Th	DU, NU, EU, Th	12.0 MT (Author unknown 1992)
4	1954–1959	U, progeny	U metal, UO_2, UO_3, UNH	Process support for U metal production for Mallinckrodt, NLO
4	1954–1961; 1979–1979	U, progeny	U metal, alloys (NU, 10% EU)	R&D, studies of work on U metal and its alloys (NU, 10% EU)
4	1954–1962	U, progeny	UF_4	Process support for U metal production (some work involved UF_4) (1954–1958); for preparation of UO_2 crystals from UF_4 by vapor deposition (1959–1962)
4	1954–1971	U, progeny	UO_2 (NU, DU)	R&D on UO_2 used in fuel of various types (powder, pellet, plate-type, etc.) (a few plates: DU)
4	1955–1957	U, progeny	U_3O_8, $(NH_4)_2U_2O_7$, other forms(?)	U extraction from U ores (ammonium carbonate leaching) (1955–1957)
4	1955?–1965?	Fission gases[e]	Fission gases	Fission gas analysis in fission gas laboratory (in Radioisotope Laboratory)
4	1956–1962	U, progeny	UO_2 (EU, mostly at 93%)	R&D on UO_2 for fuel of various types (powder, pellet, plate-type, etc.); enrichment from few percent to full; bulk of work apparently with 93% enrichment
4	1956–1956	U, progeny	$U_2O_2(NH_4)_2$	Study of preparation of ammonium uranates, including X-ray diffraction
4	1956–1959	U, progeny	U compounds (see next column)	Study of U compounds: U beryllides, borides, silicides, sulfides

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building[a]	Period	Isotopes[b]	Form[c]	Type(s) of work[d]
4	1956–1962	U, progeny	U carbides, mainly UC (NU; EU 9%–93%)	R&D, studies of U carbides and their alloys
4	1957–1958; 1961–1961	U, progeny	UCl (EU; NU?); U_3O_8 (NU?)	Study of UCl dispersions (EU, short period only, 1957–1958); making UCl from U_3O_8 (1961)
4	1957–1961	U, progeny	U nitrides, mainly UN (DU; NU; EU 5%–93%)	R&D, studies of U nitrides and their alloys
4	1961–1961	U, progeny	HEU_3O_8, HEU	Preparation of Al-U (HEU) fueled glass fiber fuel plates for irradiation
4	1955–1955	Th, progeny	?	Support for Th extraction activities?
4	1955–1965; 1971–1971	Th, progeny; minor U	Th metal, alloys; iodide, phosphate, halide, nitride, fluoride, carbide;ThO_2 alloys	Th compounds, alloys (except extraction): bomb reduction of Th halides; Th ingot prod by arc melting, casting (with Th fluorides); Th-5, -10, -15, -20 wt % U alloys; Th-U alloy made with Th iodide; ingots cast by vacuum induction melting as Th buttons, casting Th-U carbides by arc-melting; ThN prep (1956–1961); Ta alloy buttons (trace Th) (1959–1960); Th carbide powder prep by vapor deposition (1960); corrosion of Th under storage conditions, Th compounds in NaK (1960–1961); additives to Th-U (1960–1965); ThO_2 alloys (1971–1972)
4	1956–1956	U content	Fissium	Fissium (“synthetic fissium alloy”): specimen fab, annealed; X-ray diffraction
4	1954	Tracers	Tracers	Radioisotopes in industrial control (tracer methods)
4	1958–1960	Not applicable	Not applicable	Radiochemistry of inclusion compounds: nitration of hydrocarbons, urea compounds
4	1967–1974	Unknown	unknown	Research on bioeffects of underground explosions at Amchitka Island, Alaska during/after Milrow, Cannikin tests (samples counted at Battelle?)
4	1982–1982?	Not applicable	Not applicable	Radiation surveillance for Piqua Nuclear Power Facility: annual survey, soil sampling. Apparently done by Battelle (samples counted at Battelle?)
5	1954–1972	U, Th	DU, NU, EU, Th	14.7 MT (Author unknown 1992)
5	1954–1959	U, progeny	Only U metal?	Process support for U metal production for Mallinckrodt, NLO, including dingot metal
5	1954–1960; 1979–1979	U, progeny	U metal, alloys	R&D, studies of work on U metal and its alloys, including extrusion cladding, roll-bonding, fuel element fabrication
5	1954–1971	U, progeny	UO_2 (NU, DU)	R&D on UO_2 used in fuel of various types (powder, pellet, plate-type, etc.) (a few plates: DU)
5	1956–1962	U, progeny	UO_2 (EU, mostly at 93%)	R&D on UO_2 for fuel of various types (powder, pellet, plate-type, etc.); enrichment from few percent to full; bulk of work apparently with 93% enrichment
5	1956–1962	U, progeny	U carbides, mainly UC (NU; EU 9%–93%)	R&D, studies of U carbides and their alloys, including fuel element fab and coating
5	1957–1961	U, progeny	U nitrides, mainly UN (DU; NU; EU 5%–93%)	R&D, studies of U nitrides and their alloys, including bonding, welding

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building[a]	Period	Isotopes[b]	Form[c]	Type(s) of work[d]
5	1956–1959	U, progeny	U compounds (see at right)	Study of U compounds: U beryllides, borides, silicides, sulfides. Included ball-milling, fabrication
5	1955–1965; 1971–1972	Th, progeny; minor U	Th metal, alloys; iodide, phosphate, halide, nitride, fluoride, carbide;ThO_2 alloys	Th compounds, alloys (except extraction work): production of Th ingots; Th-5, -10, -15, -20 wt % U alloys; Th-U alloy made with Th iodide; Th, Th-U carbides; ThN prep (1956–1961); Ta alloy buttons (trace Th) (1959–1960); prep Th carbide powders (1960); additives to Th-U (1960–1965); ThO_2 alloys with other mats (1971–1972)
5	1958–1960	P-32, Ba-140, La-140, Y-91	Tracers in cement	Develop activation analysis, other methods to measure Mg, Ca, Al, Fe in cement; e.g., P-32 labeling, EDTA titration to precipitate tagged Ag*IO
6/7	1954–1982	U, Th	DU, NU, EU, Th	7.9 MT in Building 6, 8.7 MT in Building 7 (Author unknown 1992)
6/7	1954–1959	U, progeny	U metal, UO_2, UO_3, UNH	Process support for U metal production for Mallinckrodt, NLO
6/7	1954–1961; 1979–1979	U, progeny	U metal, alloys	R&D, studies of work on U metal and its alloys, including corrosion studies
6/7	1954–1962	U, progeny	UF_4	Process support for U metal production (some work involved UF_4) (1954–1958); for preparation of UO_2 crystals from UF_4 by vapor deposition (1959–1962)
6/7	1954–1971	U, progeny	UO_2 (NU, DU)	R&D on UO_2 used in fuel of various types (powder, pellet, plate-type, etc.) (a few plates: DU)
6/7	1955–1957	U, progeny	U_3O_8, $(NH_4)_2U_2O_7$, other forms(?)	Chemical extraction of U from U ores (by ammonium carbonate leaching, as feed to Grand Junction pilot plant) (1955–1957)
6/7	1956–1956	U, progeny	$U_2O_2(NH_4)_2$	Study of preparation of ammonium uranates
6/7	1956–1959	U, progeny	U compounds (see at right)	Study of U compounds: U beryllides, borides, silicides, sulfides
6/7	1956–1962	U, progeny	UO_2 (EU, mostly at 93%)	R&D on UO_2 for fuel of various types (powder, pellet, plate-type, etc.); enrichment from few percent to full; bulk of work apparently with 93% enrichment
6/7	1956–1962	U, progeny	U carbides, mainly UC (NU; EU 9%–93%)	R&D, studies of U carbides and their alloys
6/7	1957–1961	U, progeny	U nitrides, mainly UN (DU; NU; EU 5%–93%)	R&D, studies of U nitrides and their alloys
6/7	1957–1958; 1961–1961	U, progeny	UCl (EU, NU); U_3O_8 (NU?)	Study of UCl dispersions (EU, short period only, 1957–1958); making UCl from U_3O_8 (1961)
6/7	1961–1961	U, progeny	U_3O_8	Preparation of single U_3O_8 crystals, examination
6/7	1955–1955	Th, progeny	Unknown	Support for Th extraction activities?

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building[a]	Period	Isotopes[b]	Form[c]	Type(s) of work[d]
6/7	1955–1965; 1971–1972	Th, progeny; minor U	Th metal, ThO_2 alloys; iodide, phosphate, halide, nitride, fluoride, carbide	Th compounds, alloys (except extraction): Th-5, -10, -15, -20 wt % U alloys; Th-U alloy made with Th iodide; ThN prep (1956–1961); solubility of U in Th, corrosion of Th under storage, Th compounds in NaK (1960–1961); Th-U additives (1960–1965); ThO_2 alloys (1971–1972). Dissolution of Al-canned Th (1955)
6/7	1960?–1972?	Pu, U	Spent fuel (small quantities?); Pu alloys, UN-PuN	Study of Pu in pyrometallurgical process for U recovery from spent fuel elements, with Al-10 wt % U-5 wt % Pu; corrosion of Ta, Ta alloys by liquid Pu alloys (1960?). Study of UN-PuN fuel mat, cladding for fast reactors
6/7	1956–1956	U content	Fissium	Fissium (synthetic fissium alloy): specimen fab, annealed; X-ray diffraction
6/7	1954–1954; 1958–1962	Fe-59, Mn-54	Metal	Unknown
6/7	1956–1961	U	$HEUO_2$, DUO_2; other?	Miscellaneous neutron dosimetry/monitor work, including wire-activation method for reactor flux profile data; fast neutron effects on semiconductors;
6/7	—	U	Al_2O_3-Mo-Si_2-UO_2, EU, DU	Neutron activation of ion chamber walls; ceramic tubes of Al_2O_3-Mo-Si_2-UO_2 with EU for fission-power elements, DU for heating elements (1959-60)
6/7	1958–1960	Not applicable	Not applicable	Radiation chemistry of inclusion compounds by nitration of hydrocarbons and their urea (nitration) compounds
6/7	1958–1960	P-32, Ba-140, La-140, Y-91	Tracers in cement	Develop activation analysis, other methods to measure Mg, Ca, Al, and Fe in cement; radiometry of samples, P-32 labeling, EDTA titration with tagged Ag*IO
6/7	1959–1960	Fission gases?[e]	Fission gases?	Isotopic exchange leak detection system (an Ag-Br column in conjunction with delayed-neutron monitor)
6/7	1960–1962	H-3	Tritiated cytidine, trace levels	Miscellaneous biological effects studies, including nucleic acid metabolism in HeLa S3 cells after X-ray-induced mitotic delay (using tritiated cytidine)
6/7	1962–1963	MFP; UO_2	UO_2 particles	Internal radiation effects on catalyst activity: 5 wt % ceramic-coated UO_2 particles added to catalyst; catalyst efficiency tested; neutron irradiation
6/7	1967–1974	Not applicable	Not applicable	Research on bioeffects of underground explosions at Amchitka Island, Alaska during/after Milrow, Cannikin tests (samples counted at Battelle?)
6/7	1982–1982?	Not applicable	Not applicable	Radiation surveillance for Piqua Nuclear Power Facility: annual radiation survey, including soil sampling, by Battelle (samples counted at Battelle?)
9	1958–1962?	U, Th	DU	0.06 MT (Author unknown 1992)
9	1958–1962?	Krypton[e]	Krypton, UO_2	Analysis of gas from SME, OMRE, and OMR fuel specimens; limited laboratory-level work with DU and NU

a. Buildings 4, 6, and 7 might have been used for some work even when not listed because it is not always clear what associated laboratory-type tests might have been done. In addition, X-ray diffraction of most samples and specimens should be assumed to have been performed (usually in Building 4 if the specimen was not hot; however, plutonium was not handled at Building 4).
b. AP = activation products in this table only (for the materials in question).
c. The U form was probably NU when not specified; EU appears not to have been used before 1947. Sometimes only the final form was indicated by the relevant reference, so the possibility of a different initial or intermediate form than listed above must be kept in mind.

d. X-ray diffraction was said always to have been on a powder sample, so diffraction samples or specimens of a solid material had to be ground or milled before the test could be performed. Therefore, at least small amounts of the material should be assumed to have been ground to powder form.
e. Fission gases could be generated in samples or rods irradiated elsewhere or in the Battelle reactor. Fission gases could be counted in the in-reactor loop at the Battelle reactor, in a JN-1 hot cell when a rod or sample was opened, or (if the total FP content was less than 1 Ci) at the fission gas laboratory in the Building 4 Radioisotope Laboratory at King Avenue (Sunderman and Dickerson 1962). If chemical separations or special counting had to be done, the sample was opened in the hot cell or at the fission gas laboratory. In both such cases, the sample could be heated in a quartz tube to drive off the gases, which were then collected. Townley et al. (1960) indicate that in the analysis of fission gases from samples irradiated at the Battelle reactor, some or all of the Kr-89, Xe-137, Xe-140, and Xe-141 FP chains were studied, with the respective down-the-chain progeny S-89, Cs-137, Ba-140, and Ce-141 being analyzed in some studies by being trapped on stainless-steel mesh and with I-131 and I-133 also being analyzed when they deposited on the traps; the traps were dissolved in acid and the various progeny and radioiodine were then chemically separated and counted.

Table C-2. History of operations at the West Jefferson site.

Building	Period	Isotopes[a,b]	Form[c]	Type(s) of work[d]
JN-1 (Hot cells)	1954–1983	U, Th, AP, FP, Pu	DU, NU, EU, Th, AP, FP, Pu	8.9 MT (Author unknown 1992)
JN-1	1954–1962	U, MFP, AP	U metal, alloys (NU, 10% EU)	Postirradiation studies/examinations of clad NU element; fuel elements, specimens irradiated elsewhere; U alloys (NU or 10% EU); U-Nb alloys
JN-1	1954–1972	U, MFP, AP	UO_2 (NU)	Postirradiation studies of corrosion of clad U metal, alloys and elements irradiated elsewhere; uncoated, coated fuel particles; fission gas release from UO_2 crystals, UO_2 in cladding; sweep capsules; cleavage faces for recoils, tracks; cladding contacted with UO_2 foil (MFP penetration analysis). FP diffusion and recoil studies, including sectioning; radiation damage studies in ceramics, Al by contact with UO_2; iodine-retaining coatings
JN-1	1962–1972	U, MFP, AP	UO_2 (NU)	Study of Cr foils impregnated with FP concentration of 2×10^9 recoils/cm^2 to measure Xe-133 release and Ce-141, Ru-103, and Nb-95 distribution
JN-1	1956–1961	Th, U, MFP, Cs-137[e]	U-Th alloys	Studies of U-Th alloys: fission gas retention study, other postirradiation analyses and tests, including burnup disk sectioning, Cs-137 release
JN-1	1956–1962	U, MFP, AP	UO_2 (EU 25%–93%)	BRR, ETR loop programs (25-50% enrichment); postirradiation examination of $HEUO_2$ dispersed in SS; UO_2-SS fuel (mostly at 93%); spherical fueled elements (93% enrichment); high burnup fuel (some at 30% enrichment, some at 93%); encapsulated UO_2-SS fuel (at 93% enrichment), UO_2 pin-type fuel (at 35%); pellets (at 25% to 50% enrichment) and compacts for ML-1. Miscellaneous studies of fuel; UO_2 dispersions in various matrices, claddings (20% enrichment); coated particles, including crushing to release fission gases; high-burnup irradiation stability study of Zr_2O_2-UO_2, similar fuels
JN-1	1957–1961; 1965–1971	U, MFP	U nitrides, mainly UN (DU; NU; EU 5%–93%)	Postirradiation testing of UN dispersion, cermet forms; fission bubble growth kinetics, migration; irradiation behavior of UN fuel elements

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building	Period	Isotopes[a,b]	Form[c]	Type(s) of work[d]
JN-1	1958–1962	U, MFP[e]	U carbides, mainly UC (NU; EU 10%–93%)	Postirradiation examination of UC (some NU, some EU) capsules; dispersed UC, UC_2 (NU); other UC, UC_2 specimens (at 30% enrichment, some at 93%), further specimens (at 10% enrichment) to high burnup; included section/seal/heat in tubes for fission gas examination, postirradiation heat/capture fission gases on charcoal traps or by freezing, e.g., coated NU UC_2 particles; FP deposition in out-of-pile loop system (UC_2); fission bubble growth kinetics, migration; irradiation behavior of UC elements
JN-1	1962–1962; 1965–1966	U, MFP[e]	U silicides, hydrides	Postirradiation studies of U_2Si_2; deposition behavior of FPs in an out-of-pile gas loop system, with U hydride as fuel
JN-1	1957–1964	U, Th; MFP,[f] AP	Spent fuel in solid, liquid form (mostly non-radioactive)	Recovery (dissolution) of spent fuel: Zircex, Darex, Thorex, Zirflex, fluoride volatility, Niflex processes; mostly non-radioactive Zirflex: U fuel elements; Thorex: Th forms. Examine ORNL Fluoride Volatility Pilot Plant vessels, Hanford Purex evaporator steam coil sections. 1961 on: mostly fluoride volatility process studies
JN-1	1958–1966	AP, some incidental MFP	Al plate; Nb, Zr, SS as cladding, matrix, or alloy component	Postirradiation examination of GCRE Mark 1 fuel plate Al; elements with Nb, Zr cladding or alloys; irradiation surveillance program on SS; seals, gaskets, etc.; non-radioactive core, structural materials, especially cladding (SS, Inconel, Zircaloy-2, etc.), sintered Al particles, often with postirradiation heating; Ta alloy damage, high-burnup boron compound dispersions; materials, electrical components for SNAP-8
JN-1	1964–1983	MFP, AP	Al plate; Nb, Zr, SS as cladding, matrix, or alloy component	Postirradiation examination of items from power and defense reactors: Shippingport Atomic Power 1B generator (1964–1965); Fermi Reactor failed fuel (1967?); PM-3A Type I, Serial 2 (defective) core and Serial 3 core (1967-68); Vermont Yankee 304 SS Schedule 80 reducer (1974); Arkansas Nuclear Plant 1 reactor building spray system SS 304 recirculation piping (≤1975); H. B. Robinson, Point Beach Zircaloy (1975–1976); fuel rods (unspecified, 1977); Point Beach 1, Dresden 3 fuel rods, Oskarshamn 1 fuel rod sections (1978); Connecticut Yankee failed fuel assemblies (1981); oxide thickness on spent fuel rods (unspecified, 1983); support of Three Mile Island cleanup: characterization of highly loaded EPICOR II prefilter liners #3 and #16 by sampling, analyzing liner contents, examining liner (1981–1983)
JN-1	1979–1983?	MFP	Spent fuel	Shipping cask sabotage source term investigation; used five irradiated fuel rods from the H. B. Robinson reactor (1979)
JN-1	1964–1968	Pm-147	Pm oxide	RTG work: hot cell examination of SD-4 thermionic diode (1964–1968)
JN-1	1964–1977	Pu-238, U, Th, MFP; Nd-150, N-15 as tracers	PuO_2, UO_2, UO_2-PuO_2, UN-PuN	Radiation effects on irradiated PuO_2 (1967); in-pile creep of PuO_2, UO_2-PuO_2 (1970–1972). UN-PuN capsule postirradiation examination (1970?–1971); PuN-UN burnup with Nd-150 tracer, UN-PuN burnup with N-15 tracer (1970). After Pu-238 prod in commercial power reactor: postirradiation examination, Pu recovery (1974?–1975). Experiments, irradiation effects on fuel mat aerosols (UO_2, PuO_2) (1975?–1977)

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building	Period	Isotopes[a,b]	Form[c]	Type(s) of work[d]
JN-1	1974?–1975	Np, Pu	Pu fuel?	After production of Pu-238 in commercial power reactor: postirradiation examination, Np recovery (1974?–1975)
JN-1	1972–1980s?	Cf-252	Cf-252 wire	Processing and encapsulating Cf-252 (produced elsewhere)
JN-1	1956–1961; 1966–1967		HEUO_2, DU UO_2; other?	Miscellaneous neutron dosimetry and monitoring work, including fast neutron effects on semiconductors; neutron activation of ion chamber walls; proportional flow counter for neutron activation studies of MGCR specimen fission gas release (including in-operation degassing) (1959). Ceramic tubes of Al_2O_3-Mo-Si_2-UO_2 with E U, DU (1959–1960)
JN-1	1958–1960	Not applicable	Not applicable	Develop activation analysis, other methods to measure Ca, Al, Fe in cement
JN-1	1961–1962	Not applicable	Not applicable	Evaluation of stresses, other test parameters on sealed radiation sources
JN-1	1962–1963	MFP; U	UO_2	Internal radiation effects on catalyst activity: 5 without ceramic-coated UO_2 particles added to catalyst; catalyst efficiency measured after neutron irradiation. Postirradiation beta-gamma activity followed as function of decay time. Catalysts with up to 65 mCi/g of FP beta activity used in dehydration of cyclohexanol
JN-2 (CAL, Pu Laboratory)	1955–1978	U, Pu	NU, EU, Pu	1.7 MT (Author unknown 1992)
JN-2	1955–1961	U	U metal foil (NU, DU); UO_2 (2%–93% EU)	APPR criticality experiment (1955, EU foil). Evaluation reflected heterogeneous BWR, including critical assembly experiments (U foil) (1957–1958). Critical experiment with movable water reflector; core: Al-polyethylene-U core (1957-?). Critical assembly experiment for GCRE-1 (UO_2) (1958). Plastic-moderated critical assembly: polyethylene, polystyrene (1958–1961). Mockup ML-1-1B reactor, GCRE critical assembly elements rewrapped to average 303 g U-235 per element, 61 element positions; 19 pins/element, with 7 at 35 vol%, 12 at 70 vol% UO_2 (1959–1960). Ditto, but 48 elements with total of 31.1 kg U-235 (1961). Critical assembly experiments with UO_2 pellets, pins (1.8% enrichment); DU foil (0.04 wt % U-235), NU foil to calibrate Battelle counting facilities in activity per fission in U-235, U-238 (1959–1960)
JN-2	1964–1968	Pm-147	Pm oxide	RTGs: purify multicurie quantities of Pm-147 (oxide) (1964–1966). Fab, test SD-4 thermionic diode (1964–1968), segmented thermoelectric modules (1966)
JN-2	1964–1968; 1974–1978	Pu-238, Pu, U, Th, MFP; Nd-150, N-15 (as tracers)	Pu and U-Pu forms (see at right)	Study PuN as fuel; Pu-Al, Pu-Fe, U-Pu-fissium, U-Pu-Mo, PuO_2-UO_2; solid-solution alloys (Th-Pu, U-Pu-Nb, etc.); liquid-metal fuels (Pu alloys); ceramic fuels (PuC, PuC-UC mixtures, PuN, Pu_2Si_3, $PuSi_2$); cermet fuels (PuC, PuO_2); cast dispersion alloys (Pu-Fe alloys); PuB; fab, casting, etc. (1964–1968). Prepare U, Pu microspheres as intrinsic thermocouple flux probes (PuO_2) (1966–1968). Study UN-PuN fuel mat, cladding for fast reactors: prepare UCN-PuCN by arc melting, casting; vaporization of Pu nitride, carbide fuels; UN-PuN with liquid Na or with SS, Nb-Zr, Inconel

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building	Period	Isotopes[a,b]	Form[c]	Type(s) of work[d]
JN-2	1966–1978	See Types of work at right	See Types of work at right	Incaloy cladding (1966–1968?). Fabricate, examine UN-PuN pellets, EBR II pellet-type fuel, UN-PuN pin-type elements, irradiation capsules (1968–1971). 1970: Study of PuN-UN burnup using Nd-150 tracer, N diffusion in PuN, UN-PuN with N-15 as tracer (1970). Develop Na-bonded SS-clad PuN capsule (1972?). Design, fab RTGs with Pu-238 (PuO_2) (1974–1978)
JN-2	1966–1974?	Np, Pu	NpO_2	Prepare Np microspheres as thermocouple flux probes (Np oxide) (1966–1967?); fab NpO_2-ZrO_2, NpO_2-CaO-ZrO_2 target rods for Pu prod (1971?–1974?)
JN-2	1974–1978	Cm-244	Cm oxide?	Design, fab RTGs with Cm-244 (1974–1978)
JN-3(BRR)	1955–1975	U	NU, EU	0.1 MT (Author unknown 1992)
JN-3	1955–1965	U, MFP, AP[e]	UO_2 (NU)	Irradiation/damage study of: graphite; UO_2 rods in graphite; uncoated, coated fuel particles; ceramics. Foils in recoil studies. Irradiate to study fission gas diffusion/release from foils, coated powders, clad forms; releases from sweep capsules, fueled graphite in in-pile loop. UO_2-contacted Al damage study
JN-3	1956–1962	U, MFP, AP[e]	UO_2 (EU 25%–93%)	BRR, ETR loop programs (25%–50% enrichment); evaluate spherical fueled elements (93% enrichment); fission gas release from high-burnup irradiation (some at 30% enrichment, some at 93%); irradiation of UO_2-SS capsules, other fuels (at 93% enrichment), including gas loop measurements; irradiation coated particles
JN-3	1957–1957	U, MFP, AP	U metal (NU), alloys (NU, 10% EU)	Irradiation of Zr-clad NU element; irradiation study of U alloys (NU or 10% enrichment)
JN-3	1958–1962	U, MFP, AP[e]	U carbides, mainly UC (NU; EU 10%–93%)	Irradiation of UC (NU, EU) capsules; PyC-coated UC_2 particles (NU); UC, UC_2 (NU) dispersion specimens, other UC, UC_2 specimens (at 30% enrichment, some at 93%); high-burnup specimens (at 10% enrichment); coated spheres in sweep capsules, including in-pile fission gas analysis. FP deposition in out-of-pile loop system (UC_2); fission bubble growth kinetics, migration; UC fuel irradiation behavior
JN-3	1958–1972	AP	See at right	Irradiation effects on seals, gaskets, etc.; non-radioactive core, structural mats, especially cladding, sintered Al particles. Irradiation of SNAP-8 mats, electrical components
JN-3	1965–1971	U, MFP	U nitrides, mainly UN (DU; NU; EU 5%–93%)	Postirradiation testing of UN dispersion, cermet forms; fission bubble growth kinetics, migration; irradiation behavior of UN fuel elements
JN-3	1956–1961	U, Th, MFP	U-Th alloys	U-Th alloy studies: irradiation to measure fission gas retention, other postirradiation data
JN-3	1964–1971	Pu, U, MFP; Nd-150, N-15 (tracers)	PuO_2, PuN, UN-PuN; Nd-150, N-15 (as tracers)	Radiation effects on PuO_2 (1967); PuO_2, UO_2-PuO_2 in-pile creep studies (1970–1972). Study of UN-PuN fuel mats: irradiation of capsule(s) (1970–1971). PuN-UN burnup study with Nd-150 tracer; UN-PuN with N-15 tracer (1970). Irradiation, experiments re fuel material aerosol properties (UO_2, PuO_2) (1975?–1977)
JN-3	1964–1968	Pm-147	Pm oxide	RTG work: irradiation, in-pile testing of SD-4 thermionic diode (1964–1968)

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building	Period	Isotopes[a,b]	Form[c]	Type(s) of work[d]
JN-3	1956–1961; 1966–1967	U, MFP, AP	UO_2 (EU, DU), other forms?	Miscellaneous neutron dosimetry/monitor work, including the following. Wire-activation for reactor flux profile data; fast neutron effects on semiconductors; thermal neutron flux meter, sensor; "tactical" neutron dosimetry; fast neutron dosimetry system; neutron activation of ion chamber walls. Proportional flow counter for neutron activation studies of MGCR specimen fission gas release (1959). Thermal neutron flux monitoring system for Hanford; ceramic tubes of $Al2O_3$-Mo-Si_2-UO_2 with EU, DU (1959–1960). Study of 1 to 50,000 rad semiconductor fast neutron dosimeter (1966–1967)
JN-3	1956–1961	Not applicable	Not applicable	Miscellaneous gamma dosimetry/monitor work, including the following. Gamma effects on semiconductors, tin oxide films; effects of high-intensity radiation bursts on electrometer-type ion chambers; gamma testing of photovoltaic gamma dosimeter; high-level gamma dosimetry research, including glasses for high-level gamma dosimeters
JN-3	1958–1960	Ba-140, La-140, Y-91	Tracers in cement	Develop activation analysis to measure Mg, Ca, Al, Fe in cement
JN-3	1958–1961	Fission gases[e]	Fission gases	Installation, use of in-pile apparatus (loop) to study fission gas release (in beam tube facility) (1958–1960). Beam tube furnace for coated particle fuel irradiation, including collection, analysis of released fission products; develop methods to study fission gases with gamma spec analysis, charcoal traps (1961)
JN-3	1959–1960	Fission gases?[e]	Fission gases	Develop fuel element leak detector
JN-3	1960	AP?	AP?	Facility to study fast neutron activation using the BRR spectrum
JN-3	1962–1963	U, progeny	UO_2	Internal radiation effects on catalyst activity: 5 wt % ceramic-coated UO_2 particles added to catalyst, irradiation
JN-3	1964	N-17/O-17	Not applicable	Develop N-17 (O-17) decay neutron monitor to track reactor power
JN-3	1971–1975	AP?	Likely $HEUO_2$	Neutron radiography of reactor fuel: done in reactor pool and in out-of-pool facility
JN-4 (part of this was called the "Old" Pu Laboratory)	1960?–1968	Pu, Th, U	Pu, PuO_2, PuO_3, Th-Pu and U-Pu alloys	Study Pu in pyromet process for U recovery from spent fuel, with Al-10 wt % U-5 wt % Pu; corrosion of Ta, Ta alloys by liquid Pu alloys (1960). Study PuO_2 pellets in SS, Inconel, etc.: 10–15 vol % Pu_2O_3 presintering to PuO_2 (1960–1962). Pu alloys, 5-28 wt % Pu; Th-Pu; heat, hot roll, metallographic examination (1960–1962). Evaluate PuN fuel; Pu-Al, Pu-Fe, U-Pu-fissium, U-Pu-Mo, PuO_2-UO_2; solid-solution alloys (Th-Pu, U-Pu-Nb); liquid-metal fuels (Pu alloys); ceramics (PuC, PuC-UC mixtures, PuN, Pu silicides); cermets (PuC, PuO_2); dispersion Pu-Fe alloys; PuB; fab, cast (1962–1968). Prep U, Pu microspheres (PuO_2) (1967–1968). Study UN-PuN fuel mat: prep UCN-PuCN by arc melting, casting; vaporize Pu nitride, carbide fuels; UN-PuN with liquid Na or with SS, Inconel cladding; Na wetting of UN-PuN fuels (1966–1968?). Fabricate, examine UN-PuN pellets, EBR II pellet fuel, UN-PuN pin-type elements, irradiated capsules (1968–1971)

ATTACHMENT C
HISTORY OF OPERATIONS BY BUILDING AND MATERIAL (continued)

Building	Period	Isotopes[a,b]	Form[c]	Type(s) of work[d]
JN-4 (part of this was called the "Old" Pu Laboratory)	1964–1968	Pm-147	Pm oxide	RTG: purify multicurie quantities of Pm-147 (oxide) (1964–1966)? Fabricate, test SD-4 thermionic diode (1964–1968). Develop segmented thermoelectric modules (1966)
JN-4 (part of this was called the "Old" Pu Laboratory)	1966–1967?	Np	NpO	Prepare Np microspheres as thermocouple flux probes (Np oxide)
JN-4 (part of this was called the "Old" Pu Laboratory)	1978–1980	Pu, Th, U, progeny	Residual radioactivity	Decontamination of JN-4
JS-1	1960s?–Early 1980s?	U, Th	DU, NU, EU, Th	0.9 MT (Author unknown 1992)
JS-10, JS-12	1960s?–Early 1980s?	U	DU, EU	0.3 MT for JS-10, 0.002 MT for JS-12 (Author unknown 1992)
INL	1958–1960	U, MFP, fission gases	UC (NU, EU)	Irradiate UC (NU, EU) FP release; in-pile irradiation, postirradiation examination of UC (SRE specimens): at MTR(?), Xe-133 assay during postirradiation heating with charcoal traps, gamma spec; also Kr-85. Battelle representative observed data-taking at MTR (INL)
Hanford	1961–1962	Not applicable	Not applicable	Research at Hanford (HAPO) on reactor materials in Non-Power Reactor; on PRTR operation
Power plants, ANL, INL	1970–1981	Not applicable	Not applicable	Work at commercial nuclear power plants, etc. The 1970 work at Connecticut Yankee might have been part of program to use commercial reactors to make Pu-238 from Np target rods.

a. AP = activation products in this table only (for the materials in question).
b. The U form was probably NU when not specified; EU appears not to have been used before 1947. Sometimes only the final form was indicated by the relevant reference, so the possibility of a different initial or intermediate form than listed above must be kept in mind.
c. The U form was probably NU when not specified; EU appears not to have been used before 1947. Sometimes only the final form was indicated by the relevant reference, so the possibility of a different initial or intermediate form than listed above must be kept in mind.
d. X-ray diffraction was said always to have been on a powder sample, so diffraction samples or specimens of a solid material had to be ground or milled before the test could be performed. Therefore, at least small amounts of the material should be assumed to have been ground to powder form.
e. Fission gases could be generated in samples or rods irradiated elsewhere or in the Battelle reactor. Fission gases could be counted in the in-reactor loop at the Battelle reactor, in a JN-1 hot cell when a rod or sample was opened, or (if the total FP content was less than 1 Ci) at the fission gas laboratory in the Building 4 Radioisotope Laboratory at King Avenue (Sunderman and Dickerson 1962). If chemical separations or special counting had to be done, the sample was opened in the hot cell or at the fission gas laboratory. In both such cases, the sample could be heated in a quartz tube to drive off the gases, which were then collected. Townley et al. (1960) indicate that in the analysis of fission gases from samples irradiated at the Battelle reactor, some or all of the Kr-89, Xe-137, Xe-140, and

Xe-141 FP chains were studied, with the respective down-the-chain progeny S-89, Cs-137, Ba-140, and Ce-141 being analyzed in some studies by being trapped on stainless-steel mesh and with I-131 and I-133 also being analyzed when they deposited on the traps; the traps were dissolved in acid and the various progeny and radioiodine were then chemically separated and counted.

f. One report stated that the principal radioisotope after a 10-month cooling of the fuel was Co-60 (~100% of gamma activity in the eluted fraction) (Dayton and Tipton 1960a). However, another report stated that, at high burnup, Co-60 was a minor contributor to the total activity of decladding solutions and that the MFPs were in the same proportion as in the fuel dissolved except for cesium (Ewing, Brugger, and Sunderman 1961).

ATTACHMENT D
TYPES AND QUANTITIES OF MATERIAL

LIST OF TABLES

ATTACHMENT D
TYPES AND QUANTITIES OF MATERIAL (continued)

Table D-1. Source and production materials.

Material	Process or operation	Content and form notes[a]	Period and amount[b]	Reference
U metal, NU	Production from phosphate, etc. (see other entries below); metal refining process support; development of fuel rods as metal in a matrix; e.g., U in graphite; studies, fabrication of U metal, U alloys	None	1943–1979 overall. 1946–1957 (extraction). 1952–1958 (metal refining process support, e.g., casting large ingots); largest ingot melt ~1,200 lb, 170 lb typical (Rengstorff and Lownie 1955)]. 1951 [10 kg of U ingots to ORNL (Author unknown ca. 1987)]	Author unknown ca. 1987; DOE 2000c, Ferry 1943, DOE 2000a, Dayton and Tipton 1959a, Rengstorff and Lownie 1955
U metal, DU	Used as foil in critical assembly work; machined, rolled	None	1957–1960, possibly other years also	DOE 2000c, DOE 2000a
U metal, EU	R&D on fuel elements; processing and machining	10%–93% enrichment	1947–1962; 1979. One critical experiment: critical mass of HEU foil was 4.935 kg (Dayton and Tipton 1957).	Author unknown ca. 1987, DOE 2000c, DOE 2000a, Dayton and Tipton 1957; Saller 1947
U as recycle uranium	Apparently used for testing of recovery methods, assay	DU, NU, or EU	1962–1980, sent by Battelle to Feed Materials Production Center or Rocy Flats Plant: 168 kg DU, 5,135 kg NU, and 29 kg EU (including 1.45 kg U-235 for enriched). 1962–1982, received by Battelle: 1,616 kg DU, 2846 kg normal U, and 0.06 kg EU (including 1 g U-235 in the EU). More than half of the total kg U transfers were in 1964	DOE 2000d
UO_3	Small amount in metal refining process support; some made from UNH (1957)	NU	1954–1959	Vaughan, Bridge, and Schwartz 1957

ATTACHMENT D
TYPES AND QUANTITIES OF MATERIAL (continued)

Material	Process or operation	Content and form notes[a]	Period and amount[b]	Reference
UO_2	Extensive research, testing: production from pitchblende, MgX, soda salt; metallurgical, fab operations, including fuel element production; metal refining process support; other uses, e.g., in instruments	DU, NU, EU; much as $HEUO_2$; apparently all NU for refining process support. As powder, pellets, or plates. For 1.7 g UO_2 powder, standard capsule: activation to 1.5 Ci at 10 s postirradiation, 0.23 Ci at 2 d	1946–1972 overall. 1952–1958 (metal refining process support). 1956–1961 (instrument work). Neutron radiography of $HEUO_2$ reactor fuel in the BRR in 1971–1975. Not enriched before about 1956	Author unknown ca. 1987, DOE 2000c, Chastain 1961
U carbides	R&D, study of U carbides and alloys, mainly for fuel apps	NU and EU, up to 93%	1943–1944; 1950–1952; 1956–1961; 1972–1975	Litton 1960, Hanson 1961
U nitrides	R&D, study of U nitrides and alloys, mainly for fuel apps	DU, NU, and EU, up to 93%	1945; 1957–1961	BMI 1945b
Uranate, UNH forms	Uranate studies; metal refining support; extraction products, byproducts; producing fueled glass fiber plates; producing UCl from U_3O_8	U_3O_8; HEU_3O_8 (fueled glass); $U_2O_2(NH_4)_2$; $Na_2U_2O_7$. Possibly some enriched forms	1943–1944, 1956 (uranate studies). 1952–1959 (metal refining process support: UNH, uranates). 1946–1957 (U extraction from various forms). 1961 (fueled glass, UCl prod)	None
U salts, including UF_4	Study fused salt mixtures as liquid fuel; refining process support (UF_4); vapor deposition conversion of UO_2 to UF_4	U fluorides, hydrosulfides, phosphates, sulfates	1953–1954 (fused salt study). 1952–1959 (metal refining process support). 1959–1962 (conversion of UO_2 to UF_4)	None
Other U compounds	Studies of miscellaneous U compounds; postirradiation studies of some; study, production of UCl	NU and EU, up to 93%. U beryllides, borides, UCl (dispersion), silicides	1945: Al with 1%–30% U, 10-15-lb melts forming ingots ~6 in. × 10 in. × 1.5 in. (BMI 1945b). Feb 1947: 14 lb "Tu-Be alloy" at 90% Be; 13 lb at 30% Be; 72 lb Tu-Be dross at 2% Be; 59 lb TuO_2 +BeO at 33% Be (Grenell 1947). 1956–1959 (miscellaneous compounds). 1962, 1965–1966 (postirradiation studies). 1957–1958 (UCl study); 1961 (UCl production)	BMI 1945b, Grenell 1947

ATTACHMENT D
TYPES AND QUANTITIES OF MATERIAL (continued)

Material	Process or operation	Content and form notes[a]	Period and amount[b]	Reference
Thorium metal and alloys	R&D with Th metal; forging, rolling, machining ingots into slugs for Hanford; arc-welding Ames Th, Th alloys; mechanical/metallurgical study of Th; making Th-U alloy with Th iodide; casting Th buttons; casting Th-U carbides by arc-melting	None	1944–1961 overall. 1944–1946 (R&D). 1947 (slugs for Hanford)	BMI 1945a, 1945b; Author unknown ca. 1987
Th nitrate and other Th salts; Th-containing materials	Production of Th nitrate from various feed materials (see below); Th solvent extraction pilot plant, to purify $Th(NO_3)_4$ and produce Th metal	November 1949: Th product with 33 parts U_3O_8 per million of ThO_2 produced from carbonate solution initially having 38,000 parts U_3O_8 per million of ThO_2	1949–1950 (extraction from monazite sands). 1951–1952 (R&D, pilot plant). 1954–1955 (extraction from Brazilian monazite sludge)	DOE ca. 1987, DOE 2000c; Author unknown ca. 1987; Brown 1951; Hunter 1949, Kelley 1949, 1951; DOE 2000a
Other Th compounds	Th hydriding; electroplating metals on Th (Th phosphate); Th prep by iodide process; bomb reduction of Th halides; Th ingot prod by arc melting, casting (with Th fluorides); making Th-U alloy with Th iodide; ThN prep; Th carbide powder prep by vapor deposition; prep of Th-U, ThO_2 with various additives	Th carbides, fluorides, halides, iodide, nitride, phosphate; ThO_2. Most often to produce Th metal or a Th alloy. Th-U alloys: 5-10 without U	1951 (hydriding). 1948–1953 (electroplating metals on Th). 1951–1954 (Th prep by iodide process). 1954–1955 (electrodeposition of Th to form ingots). 1956–1961 (ThN prep).	Gallagher, Blosser, and Mann 1955

ATTACHMENT D
TYPES AND QUANTITIES OF MATERIAL (continued)

Material	Process or operation	Content and form notes[a]	Period and amount[b]	Reference
Plutonium and other TRU elements	Study of Pu in U spent fuel recovery; Ta corrosion by Pu alloys; studies of UN-PuN fuel, PuO_2 pellets, Th-Pu and U-Pu alloys, liquid metal fuels, ceramic/cermet fuels; R&D on Pu processing, fabrication; fabrication of Pu-238 heat sources; breeder experiments; Pu production from Np targets fabricated at Battelle (irradiated elsewhere), Np/Pu recovery; Pu microspheres; aerosol experiments	Some TRU elements as a result of breeder experiments. Usually Pu-238 (e.g., for RTGs: powder transferred, coated, pressed, sintered to pellets, welded into capsules), but Pu-239/240 also handled; Np-237 in microspheres. U-Pu spent fuel: 10 without U, 5 without Pu. Ceramic, cermet, and alloy fuels included Pu carbides, nitrides, oxide, and silicides and various U-Pu compounds. Some vaporization was done for fuel production. RTGs, aerosols: as PuO_2	Work apparently began in 1960 (recovery, corrosion studies). 1960–1962 (PuO2 pellets; e.g., ~1 kg UO2-PuO2 fuel pellets fabricated into test fuel in one campaign); nitride fuel studies. 1962–1968 (various fuel studies; >20 kg of Pu metal fabricated into classified shapes). 1964–1971 (test fuel irradiations). 1966–1968 (Pu microspheres as probes). 1968–1971 (fabrication of UN-PuN pellets, EBR II pellet fuel, UN-PuN pins, irradiation specimens; ~20 kg of Pu metal powder processed into PuN or PuN-UN pellets/test fuel)). 1974–1975 (Pu recovery from targets). 1974–1978 (RTG work: ~4 kg of Pu-238-O2 handled in fuel development studies, heat source fabrication. 1975–1977 (aerosol experiments). Some Pu sent from INL in 1969-70 (form unknown).	DOE 2000b, Toy 1986, DOE 2000a, Freas and Madia 1982, BCL (ca. 1983)
Phosphate rock, superphosphate material	Extraction of U via leaching and other chemical processes	Florida phosphate rock, 0.009% U, extracted at 87%–94% efficiency; superphosphate, 0.006% U, at 55% efficiency	1946–1950	Beyer 1948a, Blatz 1949
Shales	Extraction of U via leaching and other chemical processes	Tennessee shales: 0.005%–0.009% U	1946–1949, 1952–1954	Beyer 1948b, Blatz 1949, Bearse et al. 1948

ATTACHMENT D
TYPES AND QUANTITIES OF MATERIAL (continued)

Material	Process or operation	Content and form notes[a]	Period and amount[b]	Reference
Monazite sands	Extraction of U via leaching, other chemical processes; extraction of Th via solvent extraction, with production of ThO_2 or Th metal	5%–7% ThO_2 for Brazilian, Indian sand concentrates, 3%–4% for Idaho sand concentrates. 1949: 6 lb U + 100 lb Th from each ton of sand, i.e., 5%–6% Th and 0.3% U. 5-10 mR/hr contact with sack	1948–1950. In 1949, expected to process 30 to 40 lb of sand per week eventually (Blatz 1949)	Hunter 1949, Wallo 1981, Blatz 1949
Colorado carnotite ore; "Western" ores; other ores	Carnotite ore: ground to minus 10 mesh; separation of minus 325 mesh fraction; mixing with 10% salt; roasting. No details: Colorado, Western ores. Other ores: ammonium carbon leaching	Colorado carnotite: 0.25%–0.3% U as ore; after grinding, the 5% that was the minus 325 fraction assayed at 0.7%–1% U	1948–1949 (carnotite). 1952–1955 (Western ores). 1955–1957 (other ores)	Willigman 1949, Author unknown ca. 1987, DOE 2000c, Blatz 1949
Pitchblende ore	Probably same as Mallinckrodt process, including grinding; converted to UO_2	Pitchblende ore, July 1944 on, at 300 mg Ra per ton (Mason 1977); to 65% (DOE 1997, AEC 1949) or 70% (Dupree-Ellis et al. 2000) U_3O_8 by weight; to 100 mg Ra/ton (Dupree-Ellis et al. 2000); average 135 mg Ra/ton (AEC 1949); 0.185 ppm equilibrium Ra for Q-11 (60%) ores (AEC 1949); 0.1 Ci/ton, to 70% U, average U concentration >25%, ~100 mg Ra/t ore for 25% U (Eisenbud 1975)	1950–1951	Wesner et al. 1950, Langston, Tangel, and Richardson 1950, Mason 1977, DOE 1997, AEC 1949, Dupree-Ellis et al. 2000, Eisenbud 1975

ATTACHMENT D
TYPES AND QUANTITIES OF MATERIAL (continued)

Material	Process or operation	Content and form notes[a]	Period and amount[b]	Reference
MgX, other sludge forms	MgX concentrate (Mg uranate prepared in Africa from low-grade ore tailings) converted to UO_2. Other forms: washer slimes/slurries, low-grade sulfide concentrate	Pitchblende ore tailings, from which MgX was made, were 30–50% U_3O_8 (AEC 1951b).	1950 (washer slimes). 1950–1951 (MgX). 1953 (sulfide concentrate)	Meaders et al. 1950; Ewing, Kiehl, and Bearse 1950; Ewing et al. 1950a, 1952; AEC 1951b; Wheeler, Langston, and Stephens 1955; Van Kleeck, Macdonald, and Stephens 1956; Langston, Macdonald, and Stephens 1957
V-20 soda salt	Converted to UO_2	Probably sodium uranate ($Na_2U_2O_7$). 1952: Vitro soda salt assayed at 75% U_3O_8, Colorado soda salt at 60%–72% U_3O_8 (Termini 1952)	1950–1951	Termini 1952
Intact fuel elements, assemblies; fuel samples	Unirradiated, irradiated, or spent fuel material: postirradiation studies, including fission gases (assayed in an in-pile loop)	Irradiated/spent fuel specimens (usually enriched), MFPs; when cladding/additives present, could also have substantial level of activation products. Fission progeny on one loop filter: Ba-140, Ce-141, Cs-137, Sr-89 (I already decayed)	1955 to about 1983	DOE 2000a

ATTACHMENT D
TYPES AND QUANTITIES OF MATERIAL (continued)

Material	Process or operation	Content and form notes[a]	Period and amount[b]	Reference
Fuel elements and dissolution products (Th, U, activation products, FPs)	Dissolution of simulated and actual spent fuel elements to test recovery processes: Sulfex, Zircex, Darex, Thorex, Zirflex, fluoride volatility, Niflex processes. Con Ed pins were dissolved one at a time, with the solids from previous dissolutions added to each new dissolution.	Co-60 was principal isotope after 10-month fuel cooling (~100% of gamma activity in eluted fraction) (Dayton and Tipton 1960a); pin core had 44 g of 95.5% ThO_2-UO_2 by weight, cladding had 25 g of Type 304 SS (Ewing, Brugger, and Sunderman 1961). But: at high burnup, Co-60 a minor contributor to total activity of decladding solutions, MFPs in same proportion as in fuel, except for Cs (Ewing, Brugger, and Sunderman 1961).	1957–1964 overall. 1955 (dissolution of Al-canned Th); 1959 (dissolution of 25 ThO_2-UO_2 Con Ed pins, each 7.4 in. long by 0.31 in. in diameter; dissolution products were returned to ORNL for analysis)	Ewing, Brugger, and Sunderman 1961, Peterson et al. 1959
Miscellaneous activated material	Material irradiated at BRR or elsewhere, studied in hot cells during and after irradiation	Typically Al, SS, Nb, Zr. Often highly activated	1955 to about 1983	None
Fission plate	Used in the BRR shielding facility for fission spectrum/energy studies, etc.	Plate: 3.5 kg U-235 (as HEU), clad with ~25 mils 2S Al (Anno and Plummer 1962, Morgan et al. 1958). Ratio of thermal to epithermal neutrons at plate was 67; average gamma 2 MeV; thermal neutron/gamma flux ratio 16 (Morgan et al. 1958)	Plate was 28 in. in diameter (Morgan et al. 1958)	Morgan et al. 1958
FP-containing material	Used to dope catalysts to test efficacy of catalysts	Up to 65 mCi per gram of catalyst	At least 1962	Dayton and Dickerson 1962
Pm oxide	Purification of multicurie quantities of Pm-147; fabrication of diodes, RTGs	Pm-147	1964–1966 (purification). 1964–1968 (diodes). 1966–1968 (RTGs)	Ritzman et al. 1966

ATTACHMENT D
TYPES AND QUANTITIES OF MATERIAL (continued)

Material	Process or operation	Content and form notes[a]	Period and amount[b]	Reference
Np	Production of Pu from Np targets fabricated at Battelle (irradiation elsewhere), Np recovery; study, preparation of U, Pu, Np microspheres as flux probes	Np-237 in microspheres as Np oxide	1960–1974 overall? 1960 (recovery, corrosion studies). 1966–1967 (microsphere work). 1974–1975 (Np recovery from irradiated targets)	DOE 2000b
Cm	RTG work	Cm-244 (Cm oxides?)	1974–1978	None
Cf-252 material	Cf wire ported into alpha-gamma cell, cut, weighed, put into containers. These were sealed by welding, leak-tested, washed. In second cell, welded into larger container, leak-tested, washed	Cf-252	Started in the mid-1970s	Peters 2001, Scotti and Martin 1972
Miscellaneous radioisotopes	Studies using radiotracers or labeled compounds; hydrocarbon radiation chemistry studies; study using Cr foil impregnated with FPs to study Xe-133 release, other FP distribution; alpha cell direct-conversion generator develop	P-32, Ag-?, H-3, I-125 as labels; Nd-150, N-15 as tracers for burnup. Cr foil impregnated with 2E9 recoils/cm^2 of unspecified FPs (to see Ce-141, Ru-103, and Nb-95 distribution). Po-210 layer as cathode in alpha cell direct conversion generator	1954 (tracers for industrial process controls). 1958–1960 (P-32, Ag used for labeling). 1958–1960 (radiation chemistry of hydrocarbons). 1960–1962 (tritiated cytidine for bio studies). 1962–1970 (Cr foil during small part of this period). 1963–1964 (alpha cell generator development). 1970 (burnup study). 1976 (at least) (I-125 labeling)	Calkins and Pobereskin 1954; McFarling et al. 1962; Plummer et al. 1964
“Fissium” (faux irradiated fuel material)	Used to study dissolution, interactions of fuel materials	Radioactive content unclear: faux FP mix, but might have had actual U content	1956	Dayton and Tipton 1956

a. X-ray diffraction was done on samples of many and probably most specimens. For this test, the material had to be in powdered form so even larger solid specimens would have a small amount of powder associated with them, probably produced by grinding of the specimen or a specimen from the same lot. The age of forms obtained from elsewhere (e.g., metal from a refinery) is generally not available, so the material might have had significant progeny.

b. Information on quantities is not available in most cases. However, except for fuel production contracts and fuel handled in the BRR and the hot cells, the amounts handled at any given time appear to have been relatively small (laboratory or pilot-plant scale).

Table D-2. Sealed source materials.

Material	Process or operation	Content and form notes[a]	Period and amount[b]	Reference
Gamma sources	Used in nondestructive testing laboratory for radiography; used in Co-60 facility for irradiation effects research, destruction of chemical agents. Some sources stress-tested for standards development	Radiography: Co-60, Cs-137; at hot cells, 100 mCi Co-60 source in well, other large sealed Co-60 sources, 100 mCi Ra-226 source; Ir-192 (to 30 Rhm), Tm-170 (to 8 Rhm) studied at King Ave (Rhoten 1958)	All of operating period (radiography). 1961–1962 (stress evaluation). 1981–1982 (chemical agent destruction). Large Co-60, Ra-226 sources: instrument calibration. ~1982: Co-60 pellets put in spent fuel cask to make 10K-Ci source for calibrating high-level detectors for Three Mile Island	BMI 1977, Rhoten 1958, ORAUT 2006b
Beta sources	Studies of reactor coolant channel voids; U penetration into Al cladding; Po-210 alpha cell generators	Sr-90 source for uranium penetration into Al cladding (backscattering method); Ce-Pr source for alpha cell generator voltage checks	1960 (reactor coolant channel voids, U penetration into cladding); 1964 (Ce-Pr source)	Dayton and Tipton 1960b, 1961; Plummer et al. 1964
Neutron sources	Radiographic source(s) in testing laboratory; startup source in critical assembly facility (JN-2); activation source for production of short-lived isotopes (unspecified location	Critical assembly: Po-Be source ("small") (Hogan et al. 1958, Jankowski and Chastain 1958). Activation source: 10-Ci Po-Be source (Sunderman 1962, Dayton and Tipton 1961)	All of operating period for radiography (?); 1956–1970 for the startup source; Sept 1961 on for the activation source	Hogan et al. 1958, Jankowski and Chastain 1958, Ray 1972, Dayton and Tipton 1961, Sunderman 1962
U_3O_8	Small amount present with the metal in most processes	None	Unknown	Ferry 1943
Residue, scrap from carnotite ore processing	Not applicable	70–200 mg U/L of liquid residue; 0.34% U in cake residue; some scrap at 0.2% U	Unknown	Blatz 1949

a. X-ray diffraction was done on samples of many and probably most specimens. For this test, the material had to be in powdered form; therefore even larger solid specimens would have a small amount of powder associated with them, probably produced by grinding of the specimen or a specimen from the same lot. The age of forms obtained from elsewhere (e.g., metal from a refinery) is generally not available, so the material might have had significant progeny.

b. Information on quantities is not available in most cases. However, except for fuel production contracts and fuel handled in the BRR and the hot cells, the amounts handled at any given time appear to have been relatively small (laboratory or pilot-plant scale).

Table D-3. Waste and remainder materials.

Material	Process or operation	Content and form notes[a]	Period and amount[b]	Reference
Leftover samples, specimens	Not applicable	Some were highly irradiated and activated	Unknown	None
Activated parts, mats	Not applicable	Activation products	Unknown	DOE 2000a

a. X-ray diffraction was done on samples of many and probably most specimens. For this test, the material had to be in powdered form; therefore even larger solid specimens would have a small amount of powder associated with them, probably produced by grinding of the specimen or a specimen from the same lot. The age of forms obtained from elsewhere (e.g., metal from a refinery) is generally not available, so the material might have had significant progeny.

b. Information on quantities is not available in most cases. However, except for fuel production contracts and fuel handled in the BRR and the hot cells, the amounts handled at any given time appear to have been relatively small (laboratory or pilot-plant scale).

ATTACHMENT E
REPORTED INCIDENTS

Table E-1. Incidents reported at the King Avenue and West Jefferson sites.

Description of incident	Location	Date	Reference
When Co-60 wires were being cut and cast in the High Level Cell, the dose rates were so high that two windows cracked. (The dust level was high, so it is not clear if the dose rate was from the sources themselves or from surface dust on the windows.)	JN-1	Between 1955 and 1987	BCL 1997a
A worker was preparing to expose a solution using a 5-Ci Cs-137 source. The arm of the camera (which locks the source in place when the top plug is screwed into its lowest position) was placed in backwards; this released the source and allowed it to fall out of the camera and into the solution when the worker opened the lower plug to create the beam. The worker tried to recover the source and replace in the camera using a threaded rod, but was unable to do so after several attempts. The worker left the laboratory and sat in the office (where films in a drawer registered the dose rate there from the loose source) until a coworker came in from lunch. The coworker helped replace the source in the camera. The source was later transferred from the defective camera to another camera by the [Title Redacted]. The first worker's film badge registered 9,600 mR; the second worker forgot to put on the film badge before attending to the source, so second workers dose was estimated to be 3,200 mR; the [Title Redacted] also forgot to put on a badge, but no estimate of the dose is available. The workers were allowed to continue to work in the laboratory, but were restricted to non-radioactive work.	Redacted	1956	Selander 1956, Peterseim 1956
Minor corrosion of the heat exchanger tube bundle resulted in contamination of the primary demineralized-water system by the undemineralized secondary-system water. Activation of the dissolved salts in the pool water caused radioactive contamination of the water. The tube bundle and the primary water were replaced. No exposures noted.	BRR	Late October 1957	Anno and Plummer 1962
In studies of the thermal conductivity of U and UO_2, a leak developed in the hot cell measuring apparatus during analysis of hot uranium.	JN-1	March 1958	None
During SRE fuel materials program postirradiation studies, for one capsule, there was a significant air leak at the capsule-punch seal, so the capsule puncturing apparatus had to be modified. No exposures noted.	JN-1	June 1958	Dayton and Tipton 1958
Irradiated capsule containing graphite-coated UO_2 spheres was opened with power hacksaw (at ends) and a remote milling machine (lengthwise split). "During milling, the capsule slipped in the vise, allowing the middle tool to penetrate the middle can containing the two spheres fabricated at Battelle. The halves of the capsule pulled apart, freeing the inner cans ..., invalidating the fission gas measurements." After the canister gases were sampled, the canisters were opened by slitting the sides and ends with a milling machine. The spheres were recovered along with the graphite "flour" used to fill the annulus around the spheres and neutron dosimeter wires.	JN-1	~12/01/1958	Dayton and Tipton 1959b

ATTACHMENT E
REPORTED INCIDENTS (continued)

Description of incident	Location	Date	Reference
A worker punctured his finger with contaminated [Item Redacted] while cleaning the low-level cell. After he removed his outer clothing, he tried to clean the wound, but after several attempts, the dose rate at the wound was 0.4 mR/hr. He was taken to the emergency room at [Location Redacted, where the doctor was not able to clean the contamination out of the wound. The surface of the skin around the wound was excised, which reduced the reading to background level. The HP accompanied him to the hospital and performed the radiation monitoring; the HP checked all surgical equipment and saw that all materials used in the decontamination were properly disposed of. (FPs)	Redacted	1959	Selander 1959c
A worker punctured his finger with a piece of metal containing radioactive dust, while cutting up the pan from beneath the [Redacted] in the high-level cell at the [Redacted]. This operation also caused a dust cloud that highly contaminated his face and clothing. After he removed his outer clothing, the radiation reading at 6 inches in front of his face was > 5 mR/hr. After decontamination, a survey showed that he had contamination in his nose and around his eyes. Contamination in his nose was 6,000 cpm beta-gamma on a Q-Tip. Readings around his eyes were about 0.5 mR/hr. Readings at the puncture location on the back of his fourth finger, right hand, were 5 mR/hr. (FPs)	Redacted	1959	Selander 1959a
A worker technician was overexposed (greater than 3 rem in a quarter) while removing a [Redacted] from the [Redacted]. It was shipped from [Redacted], via the Battelle mail car to the [Redacted]. It had ~1 g of alumina powder and 1.7 g of UO_2 powder packed into three quartz tubes, which in turn were in a sealed hollow aluminum container inside the [Redacted]. The [Redacted] was being irradiated in the [Redacted] at the end of a long wire. The technician was raising the wire when the CAM alarmed. He and a researcher left the area. FP activity was calculated to be 1.5 Ci at 10 s after exposure of the [Redacted], 0.23 Ci 2 d later. The [Redacted] was placed in a cask. See Table 5-6 for dose rates related to this. The technician's dose to the [Redacted] was 7.75 rem (beta plus gamma); to the [Redacted], 13.55 rem; to the [Redacted], "within tolerance limits," and the whole-body dose, 13.2 rem gamma. (Probably the same incident as the entry immediately above.)	Redacted	1961	Chastain 1961
A worker got a [Redacted] overexposure in cleaning highly contaminated equipment in a hot cell. He did not wear leaded [Redacted] during many parts of the operation because they were too cumbersome and he thought he would not go over the limit.	Redacted	1962	Saling 1962
A worker opened each of four small lead containers, removed the X-ray diffraction mounts from the lead plugs, and placed each mount on a length of angle iron. A second worker pushed the angle iron with the mount through a port into the hot cell, removed the mount, and extracted the angle iron for the next mount. After all four mounts had been transferred into the cell, the first worker discovered that he was contaminated. The insides of both his lead [Redacted] were contaminated, indicating that the contamination was on the outside of the lead containers because these had been handled with [Redacted] and all the work after the small containers were opened were done with the lead [Redacted] on. The second worker checked himself and found that he too was contaminated; some contamination appeared to be inside his [Redacted].	Redacted	1963	Saling 1963

ATTACHMENT E
REPORTED INCIDENTS (continued)

Description of incident	Location	Date	Reference
A technician was overexposed (greater than 3 rem in a quarter). The overexposure was discovered when the film badge reports came back from Landauer. The cause was said to be the failure of a supervisor to keep proper track of radiation exposure.	JN-1	First quarter 1964	Saling 1964
A worker's film badge read 1,800 mrem. This was thought to have resulted from maintenance work in the [Location and Date Redacted. Although the dose rate in that general area, where he spent 5 minutes, was 2.5 R/hr before the reactor flanges were removed, the radiation level immediately in front of the [Redacted] was 10-20 times higher and he might have passed close to the [Redacted]. The worker also handled a [Redacted] reading 10 R/hr for about half a minute. His pocket dosimeter had been dropped before this and had been thought to be in working order, but on [Date Redacted] it was discovered to be malfunctioning, so it did not register the exposure.	Redacted	Redacted	None
An unspecified spill occurred in the Plutonium Laboratory. During the cleanup, coveralls, rubber gloves, booties, paper suits, masks, and both body and ring film badges were worn. Nevertheless, one worker's badges were contaminated and his dose had to be estimated from past experience.	JN-4	09/09/1968	Ottman 1968
A worker exceeded 3 rem for the second quarter of 1973 by the 6/3/1973 film badge report. The exposure was primarily associated with control area and cell entries that involved cask unloading operations and experiment setups. (Typical of various such reports from 1960 on.)	JN-1	Second quarter 1973	Lowry and Kirsch 1973
Over many years, about 55 gal annually of hydraulic fluid leaked from the glands/ seals of the rams onto the Hydraulic Door Room floor to a depth of ~5 in. The fluid was pumped out and the seal gaskets were replaced in about 1974, but the leakage continued. At one time, the walls of the room were sandblasted and some sand was left in the room, so it was suspected that the rams were scarred and that was why leakage continued even after the seal gasket replacement. Wash water from many CAA cleanings was also thought to have drained into the opening between the door and the CAA floor, adding 1-3 in. of wash water. There was also an opening between the door and the floor of the cell interior. Fuel/cladding dust and fragments, etc., and even small items fell through the opening between the door and the High Level Cell, producing a layer of sludge.	JN-1	Before 1974	Myers and Berchtold 1994
A worker was exposed to Pu-239 via [Redacted]. Urine samples collected during the first week after exposure indicated excretion rates from 0.09 to 0.21 dpm/24 hr. Urine samples collected during the fifth and sixth weeks after exposure indicated excretion rates below 0.03 dpm/24 hr. A sample collected [Date Redacted] with an unusually high 4,000 mL of urine showed a rate of 0.25 ±0.11 dpm/24 hr; however, this was considered a dubious result because of the volume. Urine, fecal, and whole-body counting data are also available for [Dates Redacted] for this worker, who was involved in another unspecified potential intake incident during the week of [Date Redacted]. Fecal samples indicated that the lung burden of Pu-239 was unlikely to be more than a few percent of the MPBB. Two other workers exposed in the same incident showed some positive Pu-239 results, but the urine and feces indicated an even lower intake for them, likely less than 1%–2% of the MPBB.	Redacted	Redacted	Geiger 1974
A worker was exposed to an average airborne concentration of 1.0E-11 µCi/mL of Pu-239, 7.17E-10 µCi/mL of Sr-90, and 7.37E-12 µCi/mL of U-238 while engaged in operations associated with [Redacted] in the [Redacted] area.	Redacted	1974	Kirsch 1974

ATTACHMENT E
REPORTED INCIDENTS (continued)

Description of incident	Location	Date	Reference
One worker was exposed to an airborne concentration of 3.60E-10 µCi/mL of soluble Pu-239 and 4.31E-10 µCi/mL of soluble Sr-90 for a 90-min period while engaged in transferring [Redacted]. Data were obtained from a lapel BZ sample. This was a planned operation: airborne contamination was anticipated and the worker was wearing protective apparel, including a full-face respirator. The respirator smears were taken immediately and the results indicated that contamination did not enter the respirator. Urine and fecal samples were also negative. It was concluded that the worker "did not inhale any of the contaminant."	Redacted	1975	Toy 1975b
Two [Redacted] were tearing out [Redacted] connected to the ducting in the old part of the building. One apparently thought that airborne levels were high, so they both took nose swabs. The swabs were negative, but the second worker's lapel air filter read positive (80 dpm of mostly Pu-239, 0.13% soluble). His gloves had been found to be contaminated and had been changed several times during the work, but his coveralls read clean. He submitted 24-hour urine and fecal specimens over the next 2 days and a smear survey of his work area was done.	Redacted	1977	BCL 1977b; Langendorfer 1977; Wissinger and Swindall 1977 (air sample report), Wissinger 1977 (smear survey report); Farmelo and Helwagen 1977 (lapel sampler report)
Core samples taken around the holding tanks and autoclave during the 1978–1982 decontamination showed that contamination had occurred "sometime during the 16-year use."	JN-4	Between 1960 and 1977	Rudolph, Kirsch, and Toy 1984
Several minor spills occurred in the Plutonium Laboratory.	JN-4	Between 1960 and 1977	Rudolph, Kirsch, and Toy 1984
At the end of the working day on [Redacted], it was found that during the removal of [Redacted] from the [Redacted], the lapel sampler filter of a technician exceeded the 40-hr MPC for soluble Pu-239, 2.0E-12 µCi/mL. The concentration was calculated at 3.51E-12 µCi/mL; after a 64-hr decay it showed 2.84E-12. The technician was removed from [Redacted] activities and placed on a bioassay program. His nasal swabs indicated no activity. A smear survey of the areas in which he had been working indicated activity; it was cleaned up.	Redacted	Redacted	Langendorfer 1977, Farmelo and Helwagen 1977, BCL 1977a, BCL 1977b, Wissinger and Swindall 1977
A release took place during a manipulator removal operation. The highest measurable contamination was 138 dpm/100 cm^2 beta-gamma; no airborne contamination was detected.	JN-1	04/07/1977	Peters and Harrison 1998
A worker was repairing a contaminated [Redacted], wearing what was later found to be a contaminated [Redacted]. The alpha air concentration inferred from his lapel air sampler was found to be 10% greater than the 40-hr control guide for soluble Pu-239 (the soluble control guide was applied until a solubility analysis was done). Subsequent analyses of the air filter showed that the material was insoluble Pu-239, giving a concentration of 5.5% of the insoluble Pu-239 control guide. Nasal swabs results were negative, as were bioassays for Pu, U, and Sr-90 in the urine and feces. Gamma isotopic analyses were also performed on the bioassay specimens.	Redacted	Redacted	Kirsch 1978a (includes most of the bioassay data)

ATTACHMENT E
REPORTED INCIDENTS (continued)

Description of incident	Location	Date	Reference
A worker entered the [Redacted] laboratory with other workers; they wore lapel samplers but he did not because he intended to stay only a minute or two. However, he became involved in moving [Redacted] with them over a 2-hr work period. His involvement became known only several days later. Because of the elevated readings from the lapel samplers, urine and fecal specimens were collected from him. Because of the delay, the readings were essentially negative and he was assigned the same dose as that to another of the workers, 196 mrem as the 50-yr accumulated lung dose.	Redacted	Redacted	Kirsch 1978b
A routine count of a [Redacted] for a worker after a control area entry prompted follow-up bioassay sampling. Counts for both a follow-up [Redacted] and a smear of the worker's respirator interior were negative. [Redacted] samples were collected 1, 2, and 3 days later to assess the early clearance phase; a urine sample was collected. Results for all samples were negative for Pu, Am, and FPs, while the [Redacted] also showed normal concentrations of uranium. It was concluded that an inhalation intake had not occurred and that the original [Redacted] probably was contaminated by contact with a contaminated surface in the [Redacted].	Redacted	Redacted	Kirsch 1978c (includes most bioassay data)
The Hot Lab received an empty NAC shipping cask under the HEDL program. Workers were not aware that the cask was pressurized when the lid was removed in the JN-1N High Bay area in preparation for loading an HEDL fuel element. Airborne radioactivity was released. Workers involved in the cask operation were wearing respiratory equipment, but another worker not involved but in the area was not. All workers evacuated immediately. There were no CAM alarms. The immediate surrounding area was contaminated by airborne radioactive debris. Bioassay data was normal for the worker not wearing a respirator. Calculations indicated that DOE and NRC airborne concentration guides were not exceeded; concentrations measured in the area were an alpha max of 3.14E-14 µCi/mL and a beta-gamma max of 1.36E-11 µCi/mL. The floor was contaminated to 5 to t10 dpm/100 cm^2 alpha and 153,000 dpm/100 cm^2 beta-gamma; equipment surfaces were contaminated to <5 dpm/100 cm^2 alpha and 42.3K dpm/100 cm^2 beta-gamma (BCL 1979; timeline and air sampling results).	JN-1	01/16/1979	BCL 1979 (timeline and air sampling results), Peters and Harrison 1998, Ray 1990
A DU fire occurred in an autoclave in Building JS-1, where military fuel was fabricated. The fire was brought under control and moved to another part of the building where a controlled burn continued until it self-extinguished. Then removable contamination was cleaned up. Personnel were not exposed to airborne contaminations above [NRC?] guidelines.	JS-1	09/22/1979	Smith et al. 1988, Peters and Harrison 1998, Ray 1990
In the Hot Cell Facility (JN-1), during unloading operations of a failed spent fuel assembly from the Connecticut Yankee Atomic Power Company, there was a release of airborne radioactivity and subsequent surface contamination from what one source described as an explosion when unloading a "superheated" fuel element. Exposure was said to be for from one to several minutes. At least one worker was not wearing a respirator. This was reported to NRC. Bioassay procedures, including in vivo counting, showed that exposures were below 10 CFR Part 20 limits and that pulmonary depositions were <1% of the permissible body burden. [Probably same incident as entry below]	JN-1	05/02/1980 or 05/03/1980	Jensen 2003, DOE 2003b, Ray 1990, Kirsch 1987

ATTACHMENT E
REPORTED INCIDENTS (continued)

Description of incident	Location	Date	Reference
In a second NAC cask incident, a cask was opened above the storage pool, which resulted in the release of widespread radioactive contamination within the high bay area up to the ceiling. There was no release from the building; personnel exposures were well below 10 CFR Part 20 limits. The cleanup and disposal of radioactive waste were completed.	JN-1	05/02/1980	Peters and Harrison 1998, Pasupathi and Toy 1990, Kirsch 2000b
A worker received a [Redacted] of 22.48 rem in the second quarter of 1980. Of this, 20.560 rem were received during an operation that involved the collection of [Redacted]. The personnel monitoring device was a finger ring TLD. The operation involved two [Redacted] tasks. There was a deviation [unspecified] in how this was done from how it was planned to be done. [Apparently it was flashed, unknown to HP.]	Redacted	Redacted	Toy 1981
A worker was [Redacted] in an unspecified fashion. The wound was smeared at the exit from the contaminated area and again several times after decontamination of the wound; all smears were negative. The Battelle wound monitor (detector) was also used to check the cut, again with negative results. It was concluded that no uptake had likely occurred. (FPs)	Redacted	1981	Swindall 1981
A fire that resulted in $100,000 of damage occurred at Building 7. It is not clear if any radioactive material was involved.	Building 7	07/15/1982	BCL 1966–1993
A worker was [Redacted] in an unspecified fashion at [Redacted]. The material was DU. Special urine monitoring was done every 2 weeks (see undated entry below) in [Redacted] due to what was said to be a potential inhalation in [Redacted] and that could be related to this [Redacted] incident. (Note that [Redacted] was not found to be contaminated in characterization surveys (e.g., Jensen 2003) and was not listed in any document as being the site of radioactive work. Because [Redacted] adjoined [Redacted], it is likely that the incident actually occurred in [Redacted].)	Redacted	Redacted	Swindall 1984 (includes wound survey results); dosimetry records
A worker was subject to special [Redacted] monitoring in [Redacted] 1984 due to a potential inhalation of [Redacted] in [Redacted]. At this time, it was recommended that he discontinue radiation work. When he was found still to be excreting low levels of [Redacted] in [Redacted] 1984, it was recommended that [Redacted] be used if he were needed to perform work rolling [Redacted] or the like.	Redacted	1984	McKown 1984a, 1984b; dosimetry records
Co-60 wires were cut and cast into slugs in the High Level Cell, creating radioactive Co-60 dust. The radiation levels in the cell were so high that two windows cracked.	JN-1	Not stated	Sunderman and Gates 1965
Air currents from a heating fan in the ceiling of the HEC fuel pool (high bay) disturbed the surface currents in the Washdown Room and caused a release into the pool area.	JN-1	Not stated	Myers et al. 1994b

ATTACHMENT F
ANALYSIS OF MEASURED NEUTRON AND PHOTON DOSE RADIATION SURVEY DATA

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RADIATION SURVEY DATA (continued)

F.1 INTRODUCTION

This analysis examines radiation survey data from the West Jefferson site, which contains several major buildings with the potential for workplace radiation fields (Amstein 2010), in relation to neutron and photon dose.

F.1.1 Building JN-1, Hot Cell Building

This facility contains hot cells that were used to provide research and technical assistance in the areas of power reactor fuel performance evaluations, postirradiation examination of nuclear materials and components, radiation source encapsulation, and physical and mechanical property studies on irradiated materials and structures.

The original section of this facility, referred to as JN-1A since 1973, was built in 1955. JN-1A contained offices, laboratory space, hot cells, and subterranean alpha gamma cells. In 1973, JN-1B was added to the building as a high-bay structure containing a large hot cell, adjacent fuel storage pool, and truck bay designed to off-load casks containing irradiated fuel assemblies into the pool for subsequent transfer underwater in to the cell. Both JN-1A and JN-1B were in operation until 1983. In 1983 both JN-1A and JN-1B were shut down and put under a surveillance and maintenance program before initiation of decontamination and decommissioning activities.

The primary purpose of the entire facility was for destructive and nondestructive examination of irradiated reactor fuel, cladding materials and associated reactor components. The facility was never a source of routine neutron radiation exposures. Occasionally, over the years, several ad hoc research experiments using sealed neutron sources (e.g., usually ^{252}Cf) were performed for relatively short periods. This facility was used for these experiments because of its inherent shielding and radiation source handling capabilities. The only other neutron exposure potential is associated with storage of these sealed sources and also a small number of the kinds of neutron sources for instrument calibration that were stored periodically for subsequent transfer off site.

F.1.2 Building JN-2, Critical Assembly Laboratory

JN-2 was designed and constructed for use as a CAL and was used for critical experiments. This building was constructed in 1955 and housed a vault used to store special nuclear materials, a radioanalytical laboratory used to assay routine health physics samples, and capabilities to perform low-level radioactivity and environmental sample analyses. From inception of use, the building housed administrative offices and a radiation detection instrument calibration laboratory that used various radiation sources including at least one or more sealed neutron calibration sources. The high bay of the building housed a facility for reactor-related critical assembly experiments. The experiments were conducted from about 1956 through about 1960 and involved the use of very low levels of uranium foils. After the cessation of critical assembly experiments, the building was used for several nuclear projects, including direct conversion concepts, irradiation experiment assemblies, and special nuclear materials storage and dispensing. The storage area contained a thick-walled vault on the ground floor.

> *A small plutonium laboratory constructed in 1966 (noted in survey records as JN-2 Pu Lab) was used for research conducted for what became known as the "Lawrence Livermore National Laboratory" involving only sealed sources of plutonium contained in glove boxes. This lab was decommissioned in 1975. After decommissioning, this small plutonium lab was converted into a radioanalytical laboratory (RAL) dedicated to analyses of low-level*

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ANALYSIS OF MEASURED NEUTRON AND PHOTON DOSE
RADIATION SURVEY DATA (continued)

radiological samples associated with onsite facilities, including analyses of various health physics samples and environmental media (BCL ca. 2010).

F.1.3 Building JN-3, Battelle Research Reactor

BRR operations began in October 1956 and ended December 1974. Based on Battelle licensing submittals to the NRC (Dingee and Chastain 1961), several critical assembly systems have been studied including beryllium-moderated, a plastic-moderated, two gas-cooled water-moderated, and a uranium dioxide-fueled, organic-and-water moderated reactors. This facility was selected because the shielding research area was equipped with instruments to measure fast neutron, thermal neutron, and gamma radiation dose rates and spectra (Klingensmith, Epstein, and Chastain 1959). BRR defueling and partial dismantlement was completed in 1975.

F.1.4 Building JN-4, Plutonium Laboratory

Built in 1960 with additions in 1964 and 1967, the Plutonium Laboratory housed activities for plutonium research and processing. The facility was operated from 1960 to 1978 when it was shut down for D&D. The facility contained one small laboratory known as the ^{238}Pu laboratory with a neutron exposure potential. It contained several gloveboxes for research studies involving the use of ^{238}Pu. The laboratory portion of the facility was dismantled in 1985. Thereafter, a hazardous materials research facility was located in Building JN-4 that involved only nonradioactive hazardous materials.

F.1.5 Building JS-1

This facility was used to develop alloy and fabrication processes, for corrosion chemistry studies, and to conduct engineering analyses for the U.S. Navy nuclear reactor program. Records indicate that uranium (depleted in ^{235}U) was used in this facility (Cotten and Payne 1990). The building was decommissioned in 1990.

F.2 AVAILABLE DATA COLLECTIONS FROM BATTELLE

The Project has the following data:

- Buildings JN-2, JN-3, and JN-4. The Project obtained 25 documents with results of radiological surveys during the period from 1957 to 1977, primarily associated with operation of the BRR in Building JN-3. The pre-1970 neutron dose survey data were in units of neutron flux (i.e., n/cm^2/s). Often survey results were provided for thermal and fast neutrons. The instruments for the surveys were typically listed on the respective documents. Selander (1957b) identified the MPL (i.e., assumed to be 5 rem/yr) for fast neutrons as being 50 n/cm^2/s for 40 hours for the JN-2 critical assembly. The primary focus of this effort was to collect and analyze paired photon and neutron radiation measurements for use in analyzing the NP ratio. Twenty of the files were identified as containing useful survey data to analyze the NP dose ratio and these were selected for data entry. The files along with their SRDB Reference Identification (Ref ID) numbers are listed in Table F-1.

- Buildings JN-3 and JN-4. The Project obtained 16 documents of supplemental information with results of radiological surveys during the period from 1971 to 1974, primarily associated with operation of the JN-2 and JN-4 facilities but including information from JN-3. Fourteen of the files were identified as containing useful survey data to analyze the NP dose ratio and these were selected for data entry. As noted in Amstein (2010), the Battelle monthly health

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physics program reports for plutonium facilities contain a section for neutron flux surveys on. The reference to the "Old Lab" refers to the original section of the JN-4 plutonium facility. The "New Lab" refers to the plutonium facility in the section of JN-4 constructed in 1967. The "Office Area" refers the office area in JN-4. The JN-2 "Pu Lab" refers to the small Plutonium Laboratory in JN-2. Note that the reference to the "238 lab" under the same section was the ^{238}Pu laboratory in the JN-4 "New Lab." The New Lab was also referred to periodically as the "Large Lab." The respective files are listed in Table F-2 with the associated SRDB reference.

- Building JN-2. Two hazard summary reports were found that described the relative neutron and photon hazards at the JN-2 facility (Table F-3).

Table F-1. Buildings JN-2, JN-3, and JN-4 neutron and photon radiation survey data.

SRDB Ref ID[a]	File
76975	1960 JN-3 redacted
76976	1961 JN-3 redacted
76977	1962 JN-3 redacted
76978	1963 JN-3 redacted
76979	1964 JN-3 redacted
76980	1965 JN-3 redacted
76982	1966 JN-3 redacted
76983	1967 JN-2 redacted
76984	1967 JN-3 redacted
76985	1968 JN-3 redacted
76986	1968 JN-4 redacted
76987	1969 JN-3 redacted
76988	1969 JN-4 redacted
76989	1970 JN-3 redacted
76990	1970 JN-4 redacted
76991	1971 JN-3 redacted
76993	1973 JN-3 redacted
76994	1974 JN-4 redacted
76995	1975 JN-4 redacted
76996	1976 JN-4 redacted

Table F-2. Buildings JN-2 and JN-4 neutron and photon radiation survey data.

SRDB Ref ID	File
79055	1971 JN-3 Reactor
79067	1972 (Apr) JN-4
79071	1972 (Aug) JN-4
79076	1974 (Jan) JN-4
79079	1972 (Jun) JN-4
79081	1974 (Mar) JN-4
79083	1972 (Oct) JN-4
79057	1974
79069	1974 (Apr) JN-4
79074	1974 (Feb) JN-4
79078	1972 (Jul) JN-4
79080	1972 (Mar) JN-4
79082	1972 (May) JN-4
79084	1972 (Sep) JN-4

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Table F-3. Building JN-2 hazards summary reports.

Reference	Title	Document number	Year
Jankowski et al. 1956a	Hazards Summary Report for the PWAR-3 Critical Assembly Experiments	BMI-ACRS-611	1956
Jankowski et al. 1956b	Hazards Summary Report for the APB Reflector-Control Critical-Assembly Experiments	BMI-ACRS-615	1956

F.3 METHODS

The respective records of photon and neutron dose measurement data were categorized according to the general workplace or type of exposure radiation from information in the survey documentation.

The data were analyzed to eliminate incomplete data in terms of dates, missing photon or missing neutron measured doses, less-than photon doses and those results with a calculated NP ratio of zero. Beta/gamma survey instrument data were typically recorded in units of milliroentgen per hour. The results were typically for photon radiation only but when it was clear that beta radiation significantly contributed to the reading, the gamma component only was obtained if available for use in the analysis. Neutron survey results were recorded during earlier years in terms of flux (n/cm^2-s) and beginning in 1970 as a measured dose rate in millirem per hour depending on the instrument. Thermal and fast neutron components were identified although most data used in the analysis were identified as fast neutron doses only. The thermal and fast neutron flux was converted to a neutron dose rate using Equation I-1 as follows:

$$neutron\ dose = 2.5\left(\frac{fast\ neutron\ flux}{20} + \frac{thermal\ neutron\ flux}{680}\right)\ \text{mrem/hr} \qquad \text{(I-1)}$$

This equation is based on NCRP Report 38 guidance on values to use to convert neutron flux to dose rate (NCRP 1971) for identified neutron energies as shown in Table F-4. The selection of 20 (i.e., neutron energy of approximately 1 MeV) to convert the workplace measured fast neutron flux to dose equivalent tends to increase the measured neutron dose rate and to result in a higher NP ratio, which is favorable to claimants.

F.4 RESULTS

The available data for each building have been analyzed and presented in the following sections.

F.4.1 Building JN-1, Hot Cell Laboratory

The primary mission of the JN-1 Hot Cell facility was nondestructive and destructive testing of irradiated reactor fuels. Most of the cells appear to have been used for these testing operations. There was one area that contained alpha-gamma cells that were specifically designed for work with transuranic materials that have a potential for neutron exposure. These cells were added to the basement of JN-1 in 1964 (Peters 2001). Some of these cells contained water windows for neutron shielding. Cells 5 and 6 were primarily used beginning in 1971 for work with ^{252}Cf. From 1971 through 1974, potential for neutron exposure was quite large as the quantities of ^{252}Cf being worked with were in the milligram range. Other operations involving transuranic isotopes appear to have occurred periodically from startup in 1955 through 1988, but the potential for neutron exposure would be much lower than during the ^{252}Cf operations.

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Table F-4. Mean neutron quality factor and flux corresponding to maximum dose equivalent of 1 mSv (100 mrem) per 40-hour week (NCRP 1971).

Neutron energy (MeV)	Quality factor	Neutron flux density ($n/cm^2/s$)
2.5×10^{-3} (thermal)	2	680
1×10^{-7}	2	680
1×10^{-6}	2	560
1×10^{-5}	2	560
1×10^{-4}	2	580
1×10^{-3}	2	680
1×10^{-2}	2.5	700
1×10^{-1}	7.5	115
5×10^{-1}	11	27
1	11	19
2.5	9	20
5	8	16
7	7	17
10	6.5	17
14	7.5	12
20	8	11

For the development of an NP ratio, several radiological surveys during the ^{252}Cf operations from 1971 through 1974 were collected from a small sampling of records boxes. In total, 49 paired neutron and photon dose rate measurements were evaluated. The NP ratio can be described as a lognormal distribution with a geometric mean (GM) of 2.4 (2.42) and a geometric standard deviation (GSD) of 3.9 (3.93) as shown in Figure F-1. The R^2 value of the fitted distribution was 0.97 indicating a robust fit to the dataset. The paired neutron and photon measurements also exhibited a reasonable degree of correlation with a Pearson correlation coefficient of 0.54 and a Spearman rank correlation coefficient of 0.56.

Because NTA film would have adequately monitored neutron energies associated with ^{252}Cf, for dose reconstruction if the worker was monitored for neutron exposure and worked in the JN-1 facility, the recorded neutron doses should be used with the appropriate adjustments (NIOSH 2007) for missed dose. If the worker worked in the JN-1 facility and was not monitored for neutron exposure, consideration should be given as to the potential for neutron exposure before assigning a neutron dose based on the NP ratio listed in Table F-5 and illustrated in Figure F-1. The data in Figure F-1 are considered the upper bound of the NP ratio because the NP ratio for ^{252}Cf is probably higher than other possible exposure scenarios. However, it is probably not appropriate to assign this ratio for a best-estimate case for a long-term employee who was not monitored for neutron exposure. For example, assigning the 95th-percentile ratio of 20:1, as listed in Table F-5, is certainly bounding, but the worker could not have worked with ^{252}Cf until at least 1971.

F.4.2 Building JN-2, Critical Assembly Laboratory

According to Battelle (BCL ca. 2010), the JN-2 CAL was built in 1955. From 1956 through the early 1960s, the facility conducted various critical experiments using EU. Hazards summary reports were

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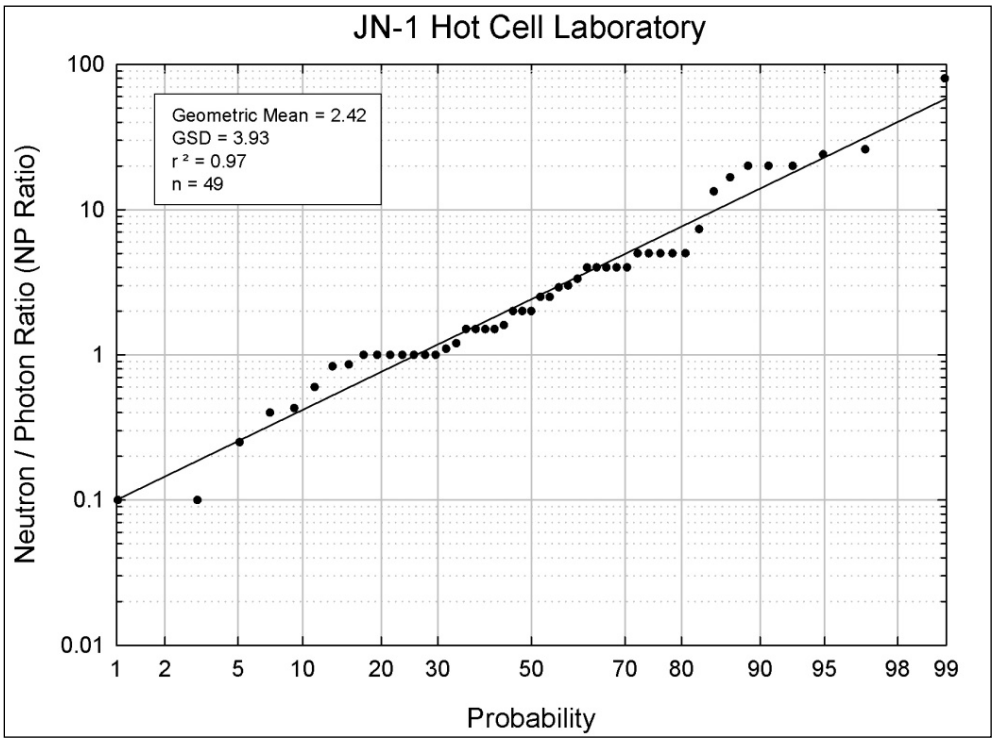

Figure F-1. JN-1 hot cell laboratory NP ratio during ^{252}Cf operations.

Table F-5. Recommended lognormal parameters of NP ratios for Building JN-1.

Workplace location	Years	n	GM	GSD	95th%
Non-^{252}Cf Operations	1955–1983[a]	(b)	(b)	(b)	(b)
^{252}Cf Operations	1971–1974	49	2.4	3.9	22.5

a. Neutron exposures probably very limited before construction and operation of alpha/gamma cells in 1964.
b. No paired survey data received.

associated with operation of a critical assembly. These reports indicated that radiation surveys through the building would be conducted during the early stages of operations. To date, records of these early surveys have not been located.

For interim use, an NP ratio was developed based on the hazards summary reports. According to Jankowski et al. (1956a), the calculated maximum neutron dose rate in the hallway outside the critical assembly during operations with power levels in the 1- to 10-W range would result in a photon dose that was one-eighth the tolerance dose of 300 mR/wk. This corresponds to a photon dose rate of approximately 0.94 mR/hr. In addition, the calculated fast neutron dose would be about one-quarter of the weekly tolerance dose of 300mrem/wk or 75 mrem/wk. This fast neutron dose corresponds to a dose rate of approximately 1.88 mrem/hr. Based on the available information (i.e., same time and location), the expected ratio for the JN-2 facility is approximately 2:1.

Based on a review of the radiological records provided by Battelle for claimants, multiple workers at the West Jefferson facility were monitored for neutron exposure as early as 1956. Unfortunately, to date all of the claimant data is for individuals who worked in JN-1 Hot Cell Facility and the JN-3 BRR. The dosimetry reports have been redacted for other workers at the facility, but at least 30 workers per 2-week cycle were monitored for neutron exposure. Most of the fast neutron results were below the

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30-mrem detection limit, but occasionally the dose would be positive, usually less than 50 mrem. Assuming that some of the neutron badges were issued to workers in the JN-2 facility, the measured neutron dose rates are well below the conservative dose estimates from the hazards summary report. As a result the dose rates in the hazards summary report are considered reasonable and bounding.

According to Amstein (2010) and BCL (ca. 2010), in 1966 a small Plutonium Laboratory was constructed at the JN-2 facility. Radiological surveys for this laboratory have been obtained. The earliest survey obtained to date was conducted in September 1967. Additional surveys from 1972 through 1974 have been added to the analysis for the development of an NP ratio for the facility during the Plutonium Laboratory operations starting around 1967. The NP ratio can be described as a lognormal distribution with a GM of 1.0 (0.97) and a GSD of 1.3 (1.28) (Figure F-2). The arithmetic mean of the distribution was 1.01. Although the R^2 value for the regression was only 0.73, this ratio is considered reasonable given the limited data and the relatively low GSD. The Pearson correlation coefficient for the 14 paired neutron and photon dose rate measurements was 0.94 and the Spearman rank correlation coefficient was 0.91, indicating a very high degree of correlation between the measurements.

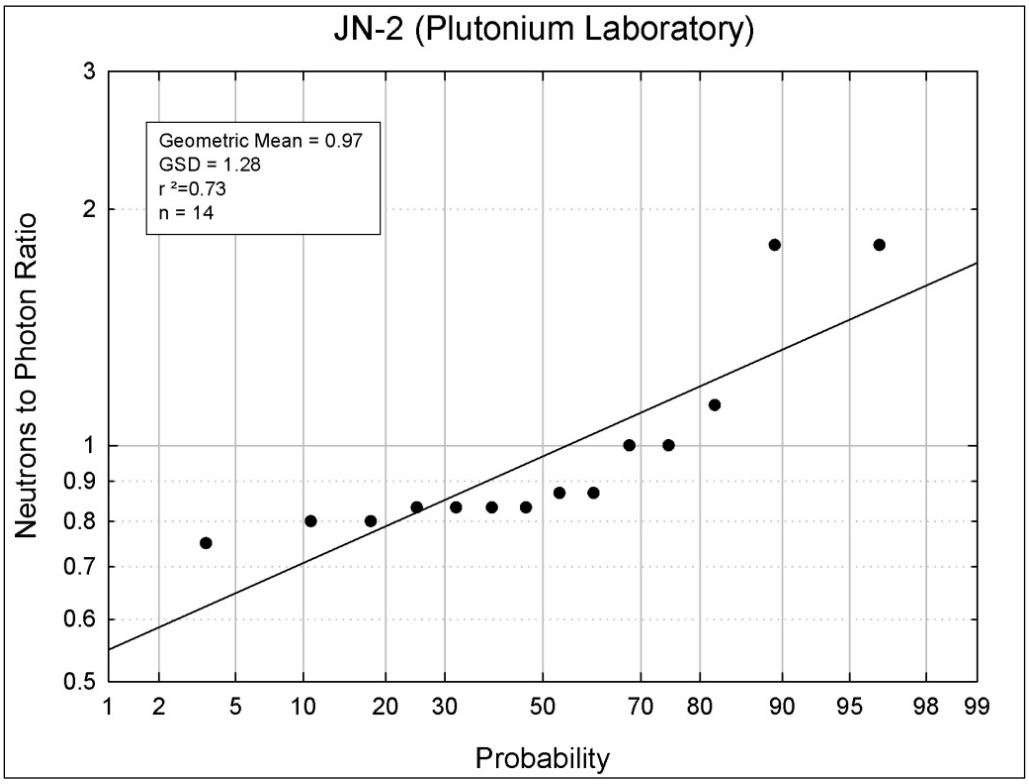

Figure F-2. JN-2 Plutonium Laboratory NP ratio.

The NP ratio for the vault area of the JN-2 facility was significantly lower than that for the Plutonium Laboratory. The arithmetic mean of the NP ratio for the vault area was calculated from nine data points to be 0.29 or approximately one-third of the NP ratio for the Plutonium Laboratory.

Because it is virtually impossible to place individual workers within the JN-2 facility (i.e. vault or Plutonium Laboratory), the approach most favorable to claimants is to use the higher of the two NP ratios for the JN-2 facility. Therefore, for the early period during the critical assembly experiments

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from 1956 through 1966, an NP ratio of 2:1 should be used. For the later years when the JN-2 Plutonium Laboratory was in, an NP ratio of 1:1 should be used. Analysis results of the JN-2 survey data were best described by a lognormal distribution with a GM of 1.0 (0.97) and a GSD of 1.3 (1.28) as shown in Table F-6.

Table F-6. Recommended lognormal parameters of NP ratios for Building JN-2.

Workplace location	Years	n	GM	GSD	95th%
Critical Experiments	1956–1960	(a)	(a)	(a)	(a)
Plutonium Laboratory	1967–1975	14	1.0	1.3	1.5

a. No paired survey data received.

F.4.3 Building JN-3, Battelle Research Reactor

A total of 3,188 NP ratios were obtained from the Battelle documents. These were categorized according to general work location or type of exposure from information in the respective survey documentation. For example, all of the vacuum cleaner samples, whether for dust or sludge, were combined. Similarly, all survey data associated with evaporators were combined. After categorization, data associated with ^{252}Cf was eliminated based on the understanding that the ^{252}Cf source was present in the reactor for about 1 year. The neutron dose rates associated with ^{252}Cf were all less than detectable (i.e., 1 mrem/hr). The data with less-than photon doses were also eliminated. This reduced the data set to 2,487 values. A scatter plot of the paired neutron and photon dose rate measurements is presented in Figure F-3. The data illustrate a higher relative measured photon dose in comparison with the measured neutron dose rate.

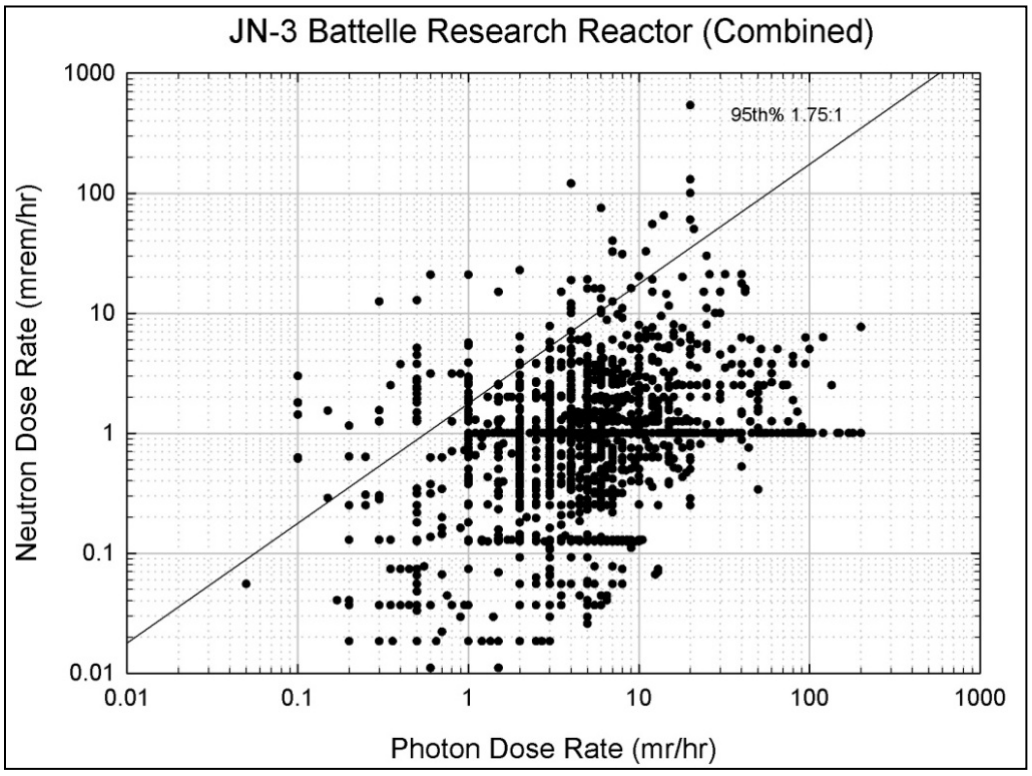

Figure F-3. JN-3 paired neutron and photon dose rate measurements.

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A lognormal probability plot of the data in Figure F-3 is presented in Figure F-4. The data are best represented by a lognormal distribution with a GM of 0.2, a GSD of 3.3 (3.34) and a 95th percentile of 1.4 (1.383).

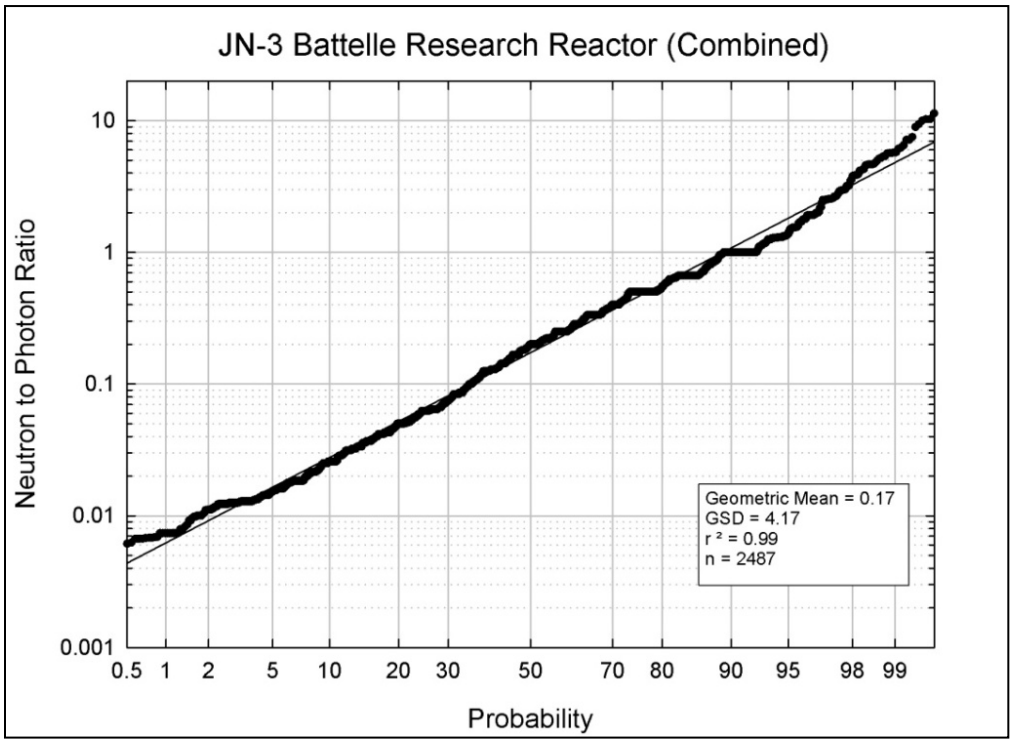

Figure F-4. JN-3 lognormal probability plot of the measured NP ratio.

Each of the workplace categories was analyzed individually; the results in Table F-7 illustrate the variation in the NP ratio. The recommended option is to use the lognormal parameters for all of these work areas collectively, as shown in the last line of Table F-7, because exposed workers would be expected to move throughout the reactor facility.

Table F-7. Statistical parameters of JN-3 survey record NP ratio analysis.

General location	No.	Median	84th %	GSD	95th %
Radioactive dust and sludge	251	0.125	0.050	4.00	1.000
Evaporator	290	0.050	0.167	3.34	0.500
Hot area (3rd floor)	174	0.250	0.667	2.67	1.000
Hot area (basement)	144	0.153	0.667	4.36	1.000
Reactor (beam tubes)	1,116	0.250	0.876	3.50	2.527
Reactor (thermal column)	366	0.222	0.667	3.00	1.028
Shielding tank	146	0.250	0.501	2.00	4.717
Combination of all areas	**2,487**	**0.2**	**0.8**	**4.2**	**2.1**

F.4.4 Building JN-4, Plutonium Laboratory

The JN-4 facility (Plutonium Laboratory) began operations in the early 1960s. According to Battelle (BCL ca. 2010), the facility was modified in 1964 and 1967. A sampling of radiological survey records for the entire West Jefferson plant resulted in a total of 93 records of paired neutron and photon dose rate measurements from 1968 through 1976 for the JN-4 facility. These were obtained from less than

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RADIATION SURVEY DATA (continued)

50 boxes of records. It is understood from Battelle that as many as 700 boxes of similar records are available. Of the 93 records of paired measurements, the neutron measurements were greater than the detection limit in 41 records. The neutron dose rate detection limit varied somewhat from 1968 through 1976 but was generally in the range of 0.2 to 0.3 mrem/hr. The data by year in Figure F-5 illustrate that the NP ratio remained relatively constant over the operational period.

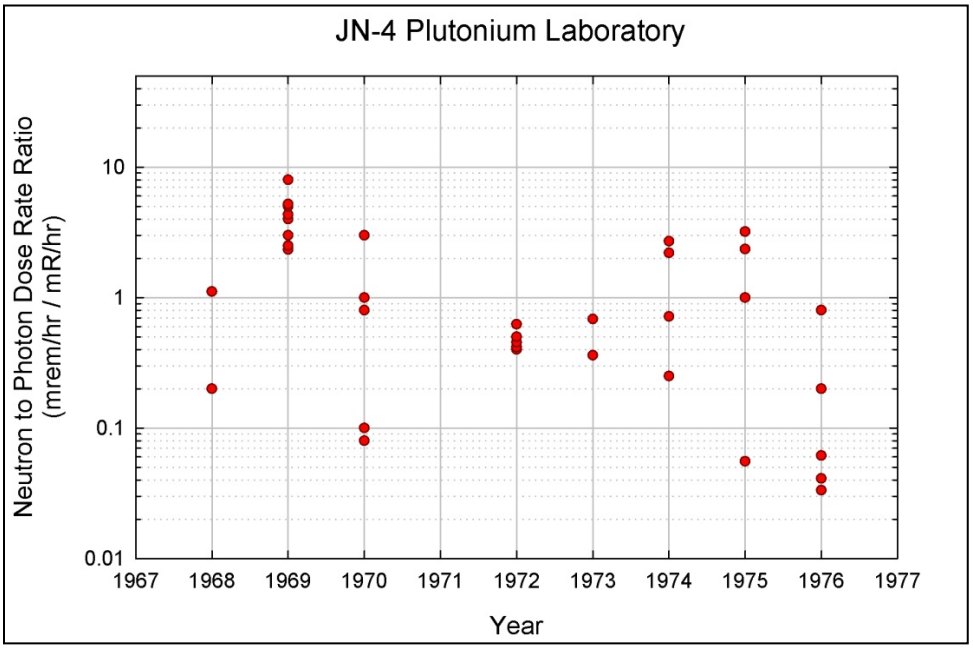

Figure F-5. JN-4 historical trend in NP ratio.

The 41 records of paired measurements were analyzed and are best described by a lognormal distribution with a GM of 0.7 (0.71) and a GSD of 5.0 (4.99) (Figure F-6). The 95th percentile of the fitted distribution was 5.0 (5.1) and the R^2 value was 0.94 indicating a strong lognormal distribution. The arithmetic mean of the distribution was 1.71, and the Pearson correlation coefficient was 0.68 while the Spearman rank correlation coefficient was 0.73. The NP ratio is summarized in Table F-8.

Table F-8. Lognormal parameters of NP ratios for Building JN-4.

Workplace location	Years	n	GM	GSD	95th%
Combined areas	1960–1978	41	0.7	5.0	9.9

The relatively large GSD is attributed to the different operations within the facility. Many of the paired NP ratio measurements greater than 1 were from the ^{238}Pu laboratory. This is likely due to the moderately high neutron production rates due to the high specific activity and the alpha interactions with light materials including oxygen. Due to the relatively large GSD, uncertainty should be considered in most dose reconstructions. A scatter plot of the paired measurements illustrates that the measured neutron and photon dose rates are highly correlated (Figure F-7). The NP ratio 95th percentile of 9.9 is shown on the graph and illustrates that this bounds the majority of the measured data as would be expected.

ATTACHMENT F
ANALYSIS OF MEASURED NEUTRON AND PHOTON DOSE
RADIATION SURVEY DATA (continued)

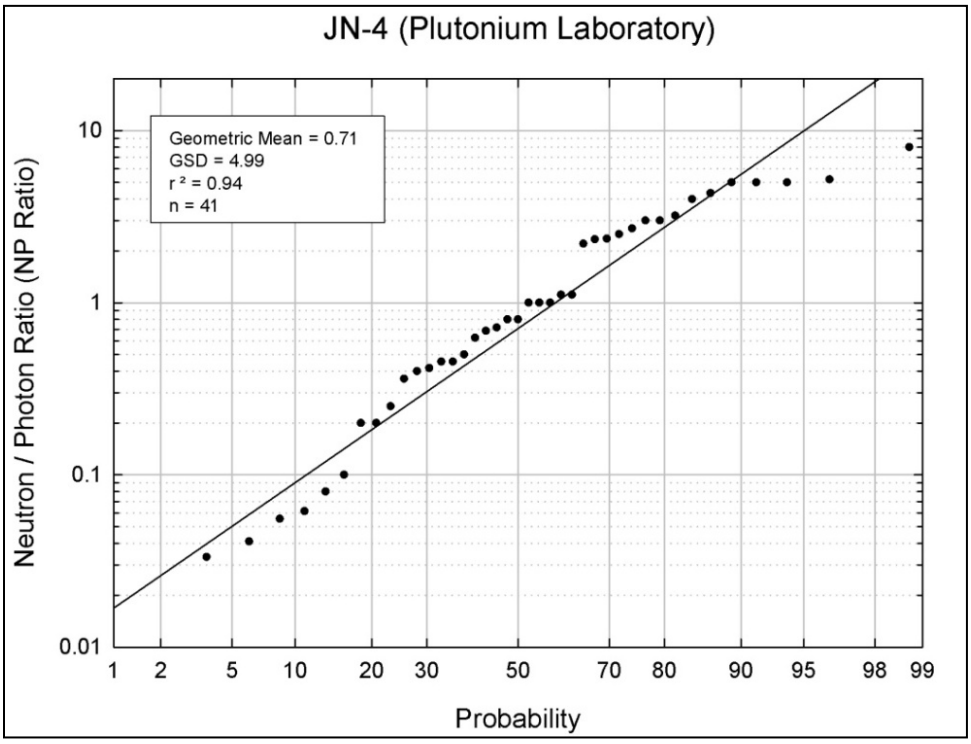

Figure F-6. JN-4 Plutonium Laboratory NP ratio.

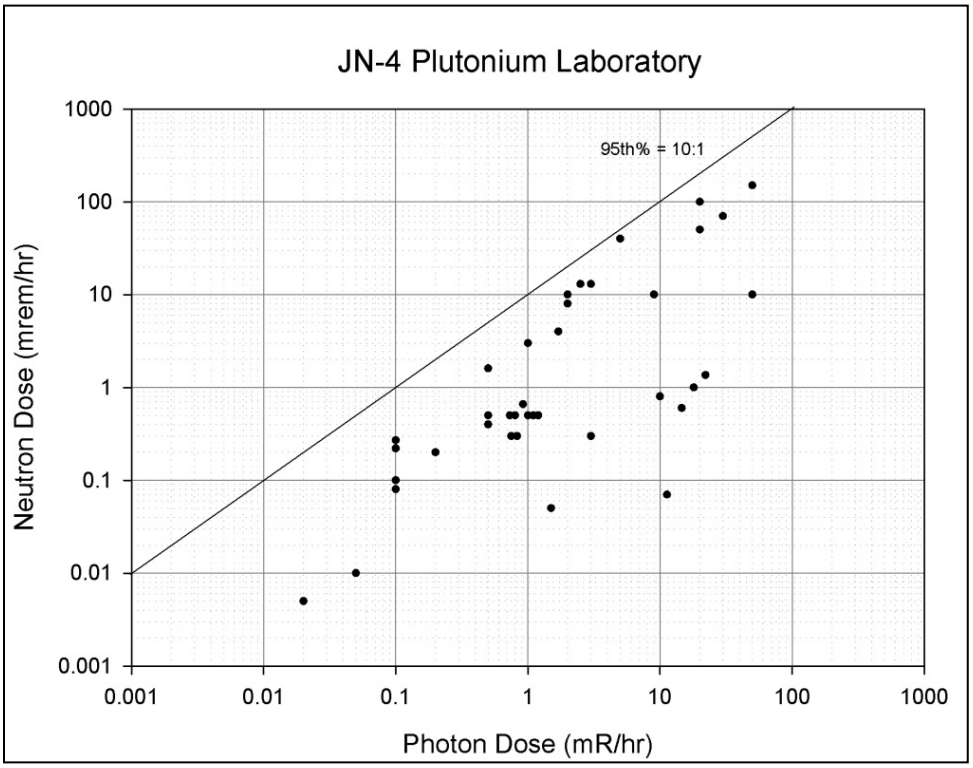

Figure F-7. JN-4 paired neutron and photon dose rate measurements.

ATTACHMENT G
ESTIMATING ONSITE AIR CONCENTRATIONS

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ATTACHMENT G
ESTIMATING ONSITE AIR CONCENTRATIONS (continued)

An atmospheric dispersion factor is a measure of the average atmospheric dispersion for long-term (chronic) atmospheric releases using Gaussian dispersion plume modeling, with units of s/m^3. For a given point or location at some distance from the source, it represents the average air concentration in µCi/m^3 divided by the release rate (µCi/s):

$$\bar{\chi} \text{ over } Q' = \frac{c}{Q'} \qquad \text{(J-1)}$$

where

c = annual average concentration at some location (µCi/m^3)
Q' = radionuclide release rate (µCi/s)

Radionuclide release rate information is available or can be readily determined for the West Jefferson facilities, but onsite air concentration data are not. However, air concentrations or atmospheric dispersion factors can be estimated using conservative atmospheric dispersion assumptions. The approach taken was to use Equation 2.9 from NCRP Report 123, *Screening Models for Release of Radionuclides to Atmosphere, Surface Water, and Ground* (NCRP 1996). This equation is applicable for situations described in Section 2.2.4 of Report 123: "Source and Receptor Not on Same Building Surface, $x \leq 2{,}5\ (A_G)^{0.5}$ or $x \leq 100$m." This equation is:

$$C = \frac{fQ}{\pi uhK} \qquad \text{(J-2)}$$

where

C = concentration (µCi/m$^{3)}$
f = frequency the wind blows toward the receptor (default = 0.25)
Q = release rate (µCi/s$^{)}$
u = wind speed (assumed at 2 m/s)
h = smaller of building height or width (assumed at 10 m)
K = constant (1 m)

Based on a unit release rate of 1 µCi/s, the concentration at some location less than 100-m distance is calculated using this equation as 0.004 µCi/m^3. Substituting into the first equation above, the representative onsite atmospheric dispersion factor is 0.004 s/m^3.

This value would decrease for greater wind speeds and larger buildings. For comparison, the 1973 annual environmental report (BCL 1974) states a dispersion factor of 2.87×10^{-5} s/m^3 for the maximally exposed individual, and the 1986 annual report (BCL 1987) states a factor of 3.35×10^{-5} s/m^3 for the MEI at a distance 500 meters southeast. These comparisons make the onsite factor seem reasonably conservative.

The National Oceanic and Atmospheric Administration reports the annual average wind speed for Columbus, Ohio to be 8.3 mph, or 3.7 m/s, based on 53 years' worth of data (http://lwf.ncdc.noaa.gov/oa/climate/online/ccd/avgwind.html), accessed 20 January 2012. The 1986 annual report documents an average wind speed of 4.5 m/s, so use of the 2 m/s wind speed in the second equation above is conservative.

ATTACHMENT G
ESTIMATING ONSITE AIR CONCENTRATIONS (continued)

Ten meters was used as a representative industrial building height; because emissions are from stacks, the heights could be even greater. The buildings at the West Jefferson and King Avenue sites are clearly longer than 10 m, as can be seen in Figures G-1 and G-2. The 1986 annual report uses a stack height of 18 m and an effective release height of 24.2 m, so 10 m is conservative. Because total emissions were considered but actually come from four different stacks on JN-1 and nine stacks total, the frequency of 0.25 is probably even more conservative than intended as a default value. A dispersion factor of 0.004 s/m^3 also compares reasonably and conservatively to the site boundary dispersion factor of 0.0000287 s/m^3, being 140 times larger.

Therefore, an atmospheric dispersion factor of 0.004 s/m^3 is considered a reasonably conservative representation of onsite atmospheric dispersion at the West Jefferson site, and is applicable and conservative at the King Avenue site as well. The atmospheric dispersion factor can be used with site-specific radionuclide release data and worker breathing rates to make estimates of inhaled radionuclide activity, as shown in Tables G-1 and G-2.

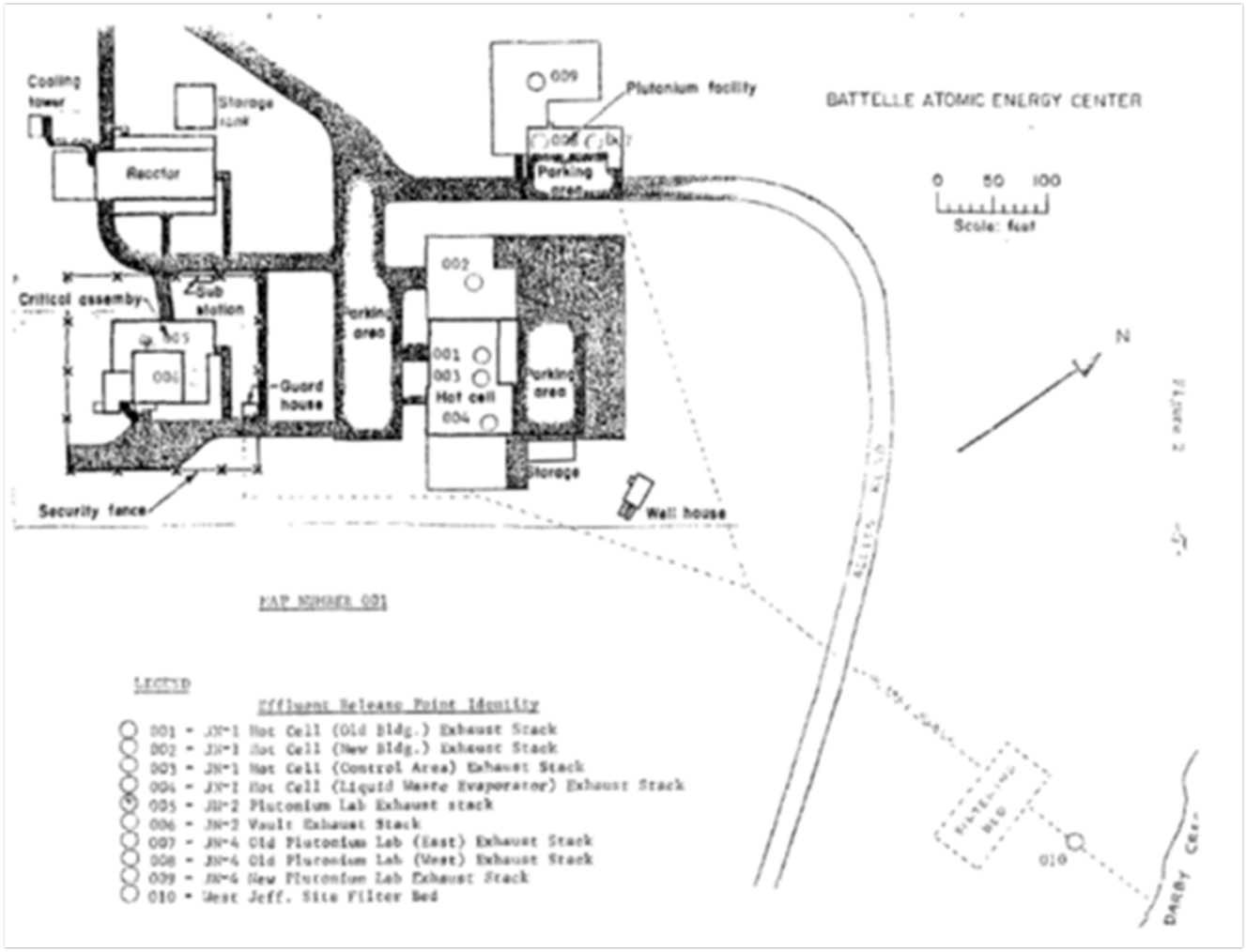

Figure G-1. Layout of the West Jefferson site, 1973.

ATTACHMENT G
ESTIMATING ONSITE AIR CONCENTRATIONS (continued)

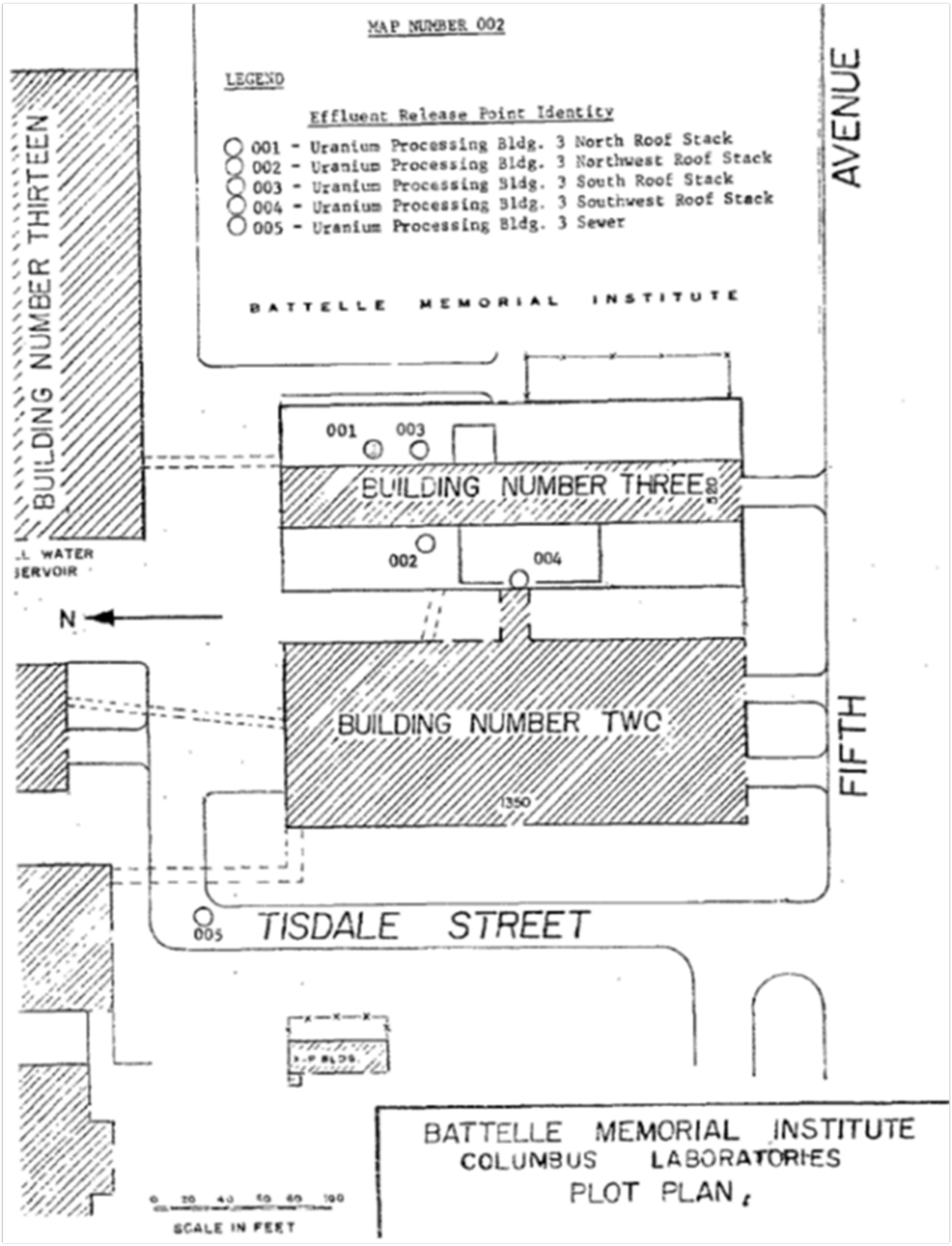

Figure G-2. Layout of the King Avenue facilities that had stack emission points, 1973.

ATTACHMENT G
ESTIMATING ONSITE AIR CONCENTRATIONS (continued)

Table G-1. Intakes of alpha-emitting radionuclides from the West Jefferson site by year (Bq).

Year	U-238	U-235	Pu-238	Pu-239	Am-241	U-234
1973	Not detected	Not detected	Not detected	3.96E-03	Not detected	Not detected
1974	Not detected	Not detected	Not detected	4.27E-03	Not detected	Not detected
1975	Not detected	Not detected	Not detected	3.96E-03	Not detected	Not detected
1976	Not detected	Not detected	Not detected	7.01E-03	Not detected	Not detected
1977	Not detected	2.13E-03	Not detected	6.70E-03	Not detected	Not detected
1978	Not detected	Not detected	Not detected	1.28E-02	Not detected	Not detected
1979	Not detected	2.26E-04	Not detected	3.96E-03	Not detected	Not detected
1980	Not detected	2.04E-03	Not detected	8.23E-03	Not detected	Not detected
1981	Not detected	1.13E-02	Not detected	1.80E-03	Not detected	Not detected
1982	Not detected	5.79E-03	Not detected	5.79E-04	Not detected	Not detected
1983	Not detected	2.83E-03	Not detected	9.45E-03	Not detected	Not detected
1984	Not detected	1.80E-03	Not detected	4.88E-03	Not detected	Not detected
1985	Not detected	2.07E-02	Not detected	6.70E-03	Not detected	Not detected
1986	Not detected	2.99E-02	Not detected	4.57E-03	Not detected	Not detected
1987	Not detected	Not detected	Not detected	6.70E-03	Not detected	Not detected
1988	1.80E-01	1.86E-02	1.13E-04	4.57E-04	2.96E-02	Not detected
1989	1.55E-01	1.13E-02	1.13E-04	4.57E-04	1.92E-02	Not detected
1990	1.01E-01	1.98E-02	1.13E-04	Not detected	1.43E-02	Not detected
1991	1.95E-02	3.35E-02	4.57E-04	1.13E-04	2.01E-02	Not detected
1992	1.49E+00	1.28E-01	3.35E-04	1.13E-04	2.38E-01	Not detected
1993	2.77E+00	2.47E-01	2.26E-04	1.13E-04	1.80E-01	Not detected
1994	1.22E-01	5.79E-02	1.22E-04	5.49E-05	1.37E-02	Not detected
1995	1.04E-01	5.18E-02	Not detected	Not detected	1.55E-02	Not detected
1996	1.10E-04	8.23E-06	3.02E-05	5.18E-05	1.10E-02	1.22E-04
1997	1.68E-06	1.43E-05	1.65E-05	2.32E-05	1.46E-02	1.86E-04
1998	2.29E-04	1.62E-05	1.52E-05	1.40E-05	1.68E-02	2.47E-04
1999[a]	2.00E-04	3.17E-05	1.10E-05	9.60E-06	1.60E-02	2.10E-04
2000	1.71E-04	1.55E-05	6.70E-06	5.18E-06	1.52E-02	1.74E-04
2001	1.65E-04	1.25E-05	1.10E-05	7.62E-06	1.49E-02	1.68E-04
2002	1.52E-04	1.28E-05	1.43E-05	8.53E-06	1.65E-02	1.62E-04
2003	1.40E-04	6.70E-05	2.44E-05	2.26E-05	1.40E-02	1.37E-04
2004	1.31E-04	1.01E-05	1.22E-04	4.27E-05	7.92E-02	1.34E-04
2005[b]	2.53E-04	2.26E-05	4.57E-04	1.34E-04	9.45E-03	2.29E-04

a. No report was located for 1999; values are based on average of adjacent years.
b. Intakes for 2005 may be applied to subsequent years through the present.

ATTACHMENT G
ESTIMATING ONSITE AIR CONCENTRATIONS (continued)

Table G-2. Intakes of mixed fission product radionuclides from the West Jefferson site by year (Bq).

Year	Co-57	Co-60	Sr-90	Sb-125	Cs-134	Cs-137	Eu-152	Eu-154
1975	Not detected	4.27E-02	Not detected	7.92E-03	5.79E-03	1.22E-02	Not detected	Not detected
1976	1.01E-03	5.79E-02	Not detected	1.43E+00	5.49E-03	5.49E-03	Not detected	Not detected
1977	4.57E-04	1.58E+00	Not detected	3.96E-01	1.46E-03	1.46E-02	9.14E-03	5.18E-03
1978	2.41E-02	3.02E-01	Not detected	1.10E-01	2.83E-03	4.88E-02	Not detected	Not detected
1979	1.25E-03	2.83E-02	Not detected	6.10E-03	3.35E-04	1.01E-03	Not detected	1.13E-04
1980	5.79E-04	5.49E-02	Not detected	5.18E-01	1.01E-04	7.01E-03	Not detected	1.13E-05
1981	1.13E-04	3.35E-03	3.35E-02	9.75E-02	2.86E-02	8.53E-02	1.25E-03	Not detected
1982	1.13E-04	5.79E-02	3.35E-06	5.18E-02	7.01E-03	4.27E-02	4.88E-03	5.18E-03
1983	Not detected	1.98E-02	5.79E-07	5.18E-02	1.13E-04	2.44E-02	Not detected	Not detected
1984	Not detected	8.84E-03	3.35E-07	2.59E-02	Not detected	3.35E-03	Not detected	Not detected
1985	Not detected	Not detected	3.96E-03	5.79E-02	Not detected	2.65E-02	Not detected	Not detected
1986	Not detected	Not detected	4.88E-03	5.79E-02	1.65E-02	3.35E-02	Not detected	Not detected
1987	Not detected	2.26E-02	1.46E-03	4.27E-02	1.95E-02	4.57E-02	4.27E-02	5.79E-02
1988	1.74E-02	3.66E-02	1.34E-03	7.31E-02	2.59E-02	3.35E-02	7.31E-02	9.75E-02
1989	5.49E-03	1.52E-02	1.71E-03	2.56E-02	9.75E-03	1.46E-02	2.56E-02	3.96E-02
1990	4.88E-03	1.40E-02	2.38E-03	2.86E-02	1.77E-02	2.77E-02	2.44E-02	3.66E-02
1991	8.23E-03	1.71E-02	1.46E-03	3.66E-02	1.19E-02	1.65E-02	3.05E-02	4.88E-02
1992	3.02E-01	9.14E-02	3.96E-03	1.62E-01	7.92E-02	1.25E-01	4.88E-01	3.05E-02
1993	2.38E-02	6.40E-02	2.26E-04	1.46E-01	5.18E-02	5.79E-02	2.13E-01	4.88E-02
1994	7.31E-03	1.49E-02	7.31E-03	3.96E-02	1.31E-02	1.43E-02	2.04E-02	1.43E-02
1995	5.79E-03	1.28E-02	1.71E-04	3.66E-02	1.13E-02	1.19E-02	1.65E-02	1.16E-02
1996	5.79E-03	1.16E-02	1.19E-04	3.05E-02	1.04E-02	1.16E-02	1.68E-02	1.16E-02
1997	6.10E-03	1.31E-02	1.10E-04	3.35E-02	1.16E-02	1.25E-02	1.77E-02	1.25E-02
1998	Not detected	1.40E-02	1.40E-04	3.05E-02	1.22E-02	1.34E-02	1.77E-02	1.25E-02
1999[a]	No report	1.33E-02	1.87E-04	3.20E-02	1.17E-02	1.30E-02	1.71E-02	8.63E-03
2000	Not detected	1.25E-02	2.35E-04	3.35E-02	1.13E-02	1.25E-02	1.65E-02	1.16E-03
2001	Not detected	1.31E-02	3.05E-04	3.05E-02	1.10E-02	1.22E-02	1.65E-02	1.16E-02
2002	Not detected	1.28E-02	2.90E-04	3.35E-02	1.13E-02	1.25E-02	1.68E-02	1.83E-02
2003	Not detected	1.25E-02	3.05E-04	3.35E-02	1.10E-02	1.16E-02	1.62E-02	1.13E-02
2004	Not detected	8.84E-02	8.53E-04	2.59E-02	8.84E-03	1.98E-01	1.25E-02	8.84E-03
2005[b]	Not detected	1.13E-02	5.49E-03	3.05E-02	1.07E-02	1.01E+00	1.62E-02	1.13E-02

a. No report was located for 1999; values are based on average of adjacent years.
b. Intakes for 2005 may be applied to subsequent years through the present.